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April 5, 2012
L-12-077

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

SUBJECT:
Beaver Valley Power Station, Unit Nos. 1 and 2
Docket No. 50-334, License No. DPR-66
Docket No. 50-412, License No. NPF-73
Pressure and Temperature Limits Report Revision

Pursuant to the requirements of Beaver Valley Power Station, Unit Nos. 1 (BVPS-1) and 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-1 PTLR, Revision 5 and the BVPS-2 PTLR, Revision 4. 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-1 PTLR was revised on March 16, 2012 to include the following changes:

1. Section 5.2.2, "Reactor Vessel Material Surveillance Program," was revised to correct a cross reference error to the Updated Final Safety Analysis Report (UFSAR). The reference to UFSAR Table 5.5-3 was incorrect and has been revised to reference UFSAR Table 4.5-3.

2. Section 5.2.2, "Reactor Vessel Material Surveillance Program," was revised to more closely reflect the language in the 1982 edition of ASTM-E185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," regarding the removal of reactor vessel material specimens.

3. Section 5.2.4, "References," was revised to correct a typographical error.

The BVPS-2 PTLR was revised on March 16, 2012 to include the following change:

1. Section 5.2.2, "Reactor Vessel Material Surveillance Program," was revised to more closely reflect the language in the 1982 edition of ASTM-E185, "Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," regarding the removal of reactor vessel material specimens.

There are no regulatory commitments contained in this letter. If there are any questions, or if additional information is required, please contact Mr. Phil H. Lashley, Supervisor – Fleet Licensing, at (330) 315-6808.

Sincerely,

Paul A. Harden

Enclosures:
A Beaver Valley Power Station, Unit No. 1, Pressure and Temperature Limits Report, Revision 5
B Beaver Valley Power Station, Unit No. 2, Pressure and Temperature Limits Report, Revision 4

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

Enclosure A
L-12-077

Beaver Valley Power Station, Unit No. 1
Pressure and Temperature Limits Report, Revision 5
(23 Pages Follow)

5.0 ADMINISTRATIVE CONTROLS

5.2 Pressure and Temperature Limits Report

BVPS-1 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	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	N/A	5.2-3
3.5.2	N/A	N/A	5.2-3

BVPS-1 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 1 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 Liquid Water Discharge.

5.2.1 Operating Limits

The PTLR limits for Beaver Valley Power Station (BVPS) Unit 1 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 2 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

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 Figure 5.2-2 and Table 5.2-2. These limits are defined in Reference 2. 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 and 5.2-2 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.

Pressure-temperature limit curves shown in Figure 5.2-3 were 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.

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 and enable temperature in accordance with Table 5.2-3. The lift setting provided does not impose any reactor coolant pump restrictions.

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 Table 5.2-3 accounts for appropriate instrument error.

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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 is 347°F.

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 318°F.

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

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 keylock switches (one in each train) into their “automatic” position. Once OPPS is activated (armed/enabled) reactor coolant system pressure transmitters will signal a rise in system pressure above the 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 4.5-3 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 3 and 4) 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 1. This commitment is a condition of license Amendment 256 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 Pressure and Temperature Limits Report

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

Table 5.2-4a, taken from Reference 2, shows the Calculation of Chemistry Factors based on St. Lucie and Fort Calhoun Surveillance Capsule Data.

Table 5.2-4b, taken from Reference 3, shows the St. Lucie and Fort Calhoun Surveillance Weld Data.

Table 5.2-5, taken from Reference 2, provides the reactor vessel beltline material property table.

Table 5.2-6, taken from Reference 12, provides a summary of the Adjusted Reference Temperature (ARTs) for 30 EFPY.

Table 5.2-7, taken from Reference 12, shows the calculation of ARTs for 30 EFPY.

Table 5.2-8 shows the Reactor Vessel Toughness Data (Unirradiated).

Table 5.2-9, taken from Reference 5, provides RT_{PTS} values for 28 EFPY.

Table 5.2-10, taken from Reference 11, provides RT_{PTS} values for 54 EFPY.

5.2 Pressure and Temperature Limits Report

5.2.4 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

2. WCAP-15570, Revision 2, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, April 2001.

3. WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," C. Brown, et. al., November 2000.

4. WCAP-8457, "Duquesne Light Company, Beaver Valley Unit No. 1 Reactor Vessel Radiation Surveillance Program," J. A. Davidson, October 1974.

5. WCAP-15569, "Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1," C. Brown, et al., November 2000.

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. Deleted

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

11. WCAP-15571, Supplement 1, "Analysis of Capsule Y from FirstEnergy Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," B. N. Burgos, June 2007.

12. WCAP-16799-NP, Revision 1, "Beaver Valley Power Station Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," B. N. Burgos, June 2007.

13. FENOC-07-120, Transmittal of LTOPS Setpoint Analysis Report, July 26, 2007.

14. Westinghouse Calculation CN-SCS-07-27, Rev. 0, LTOPS Setpoint Evaluation for Beaver Valley Unit 1 at 30 EFPY.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	LOWER SHELL PLATE
LIMITING ART VALUES AT 30 EFPY:	1/4T, 245.7°F
	3/4T, 207.6°F

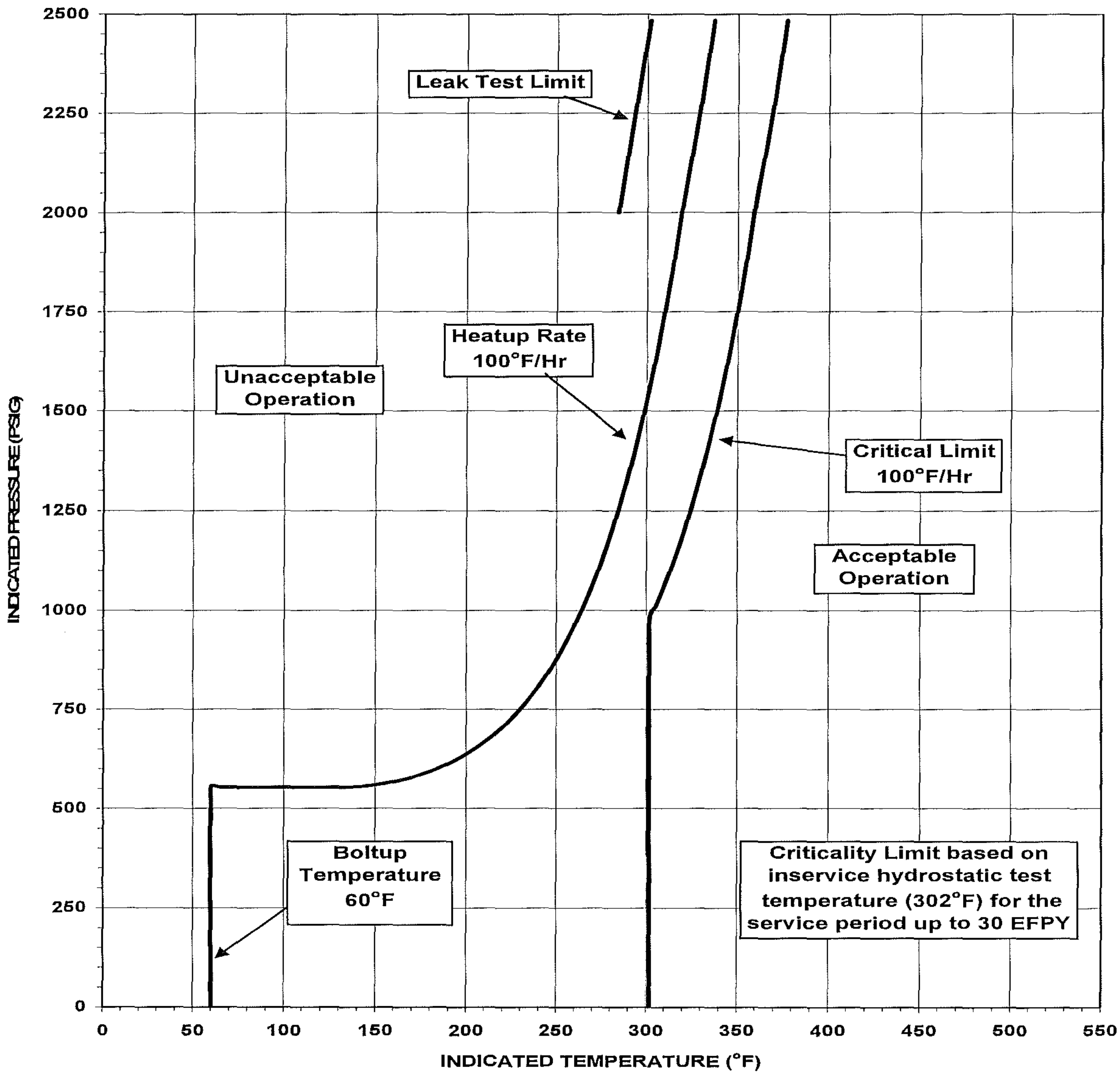

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: LOWER SHELL PLATE
LIMITING ART VALUES AT 30 EFPY: 1/4T, 245.7°F
3/4T, 207.6°F

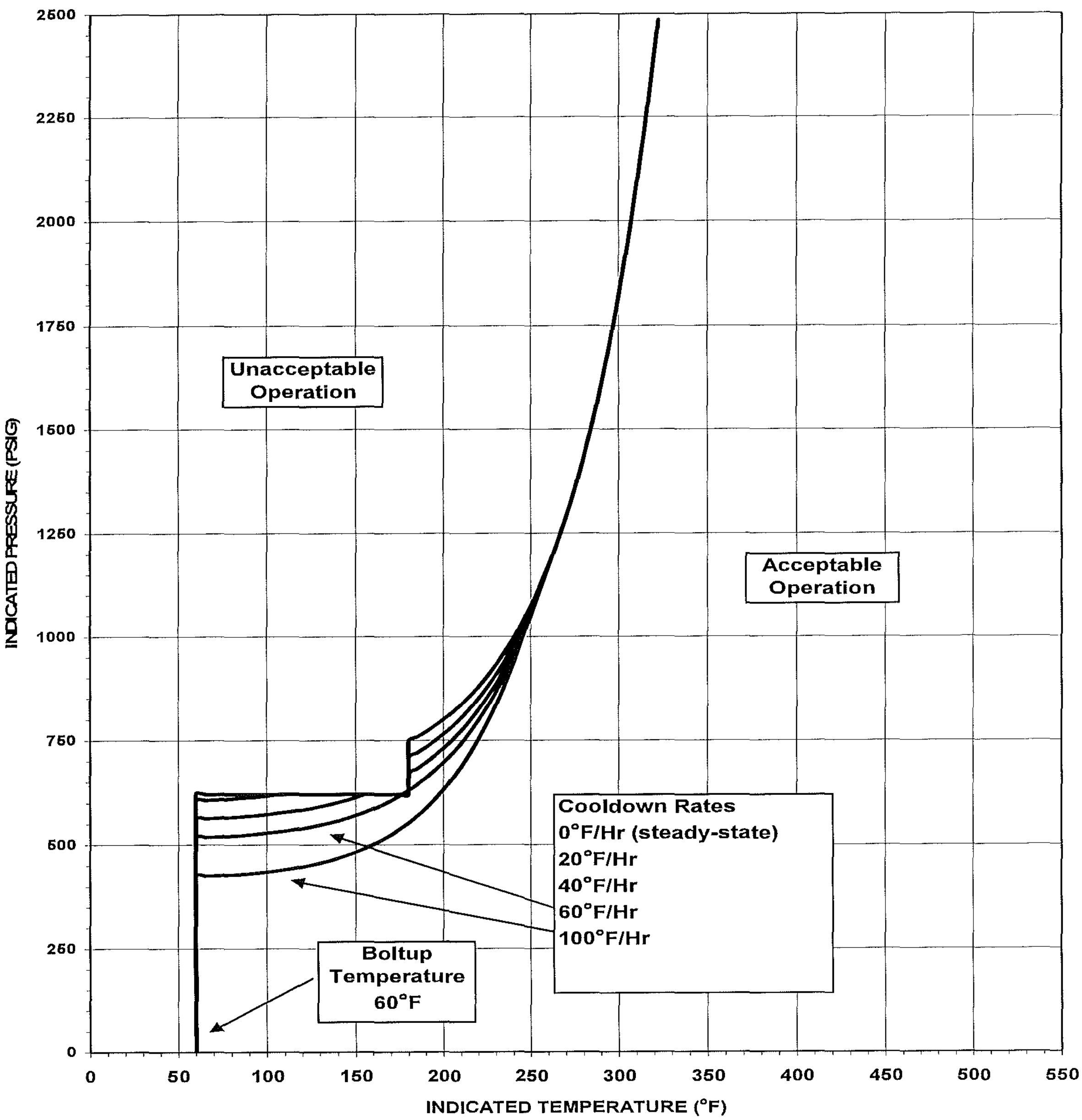

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown
Limitations Applicable for the First 30 EFPY (LCO 3.4.3)

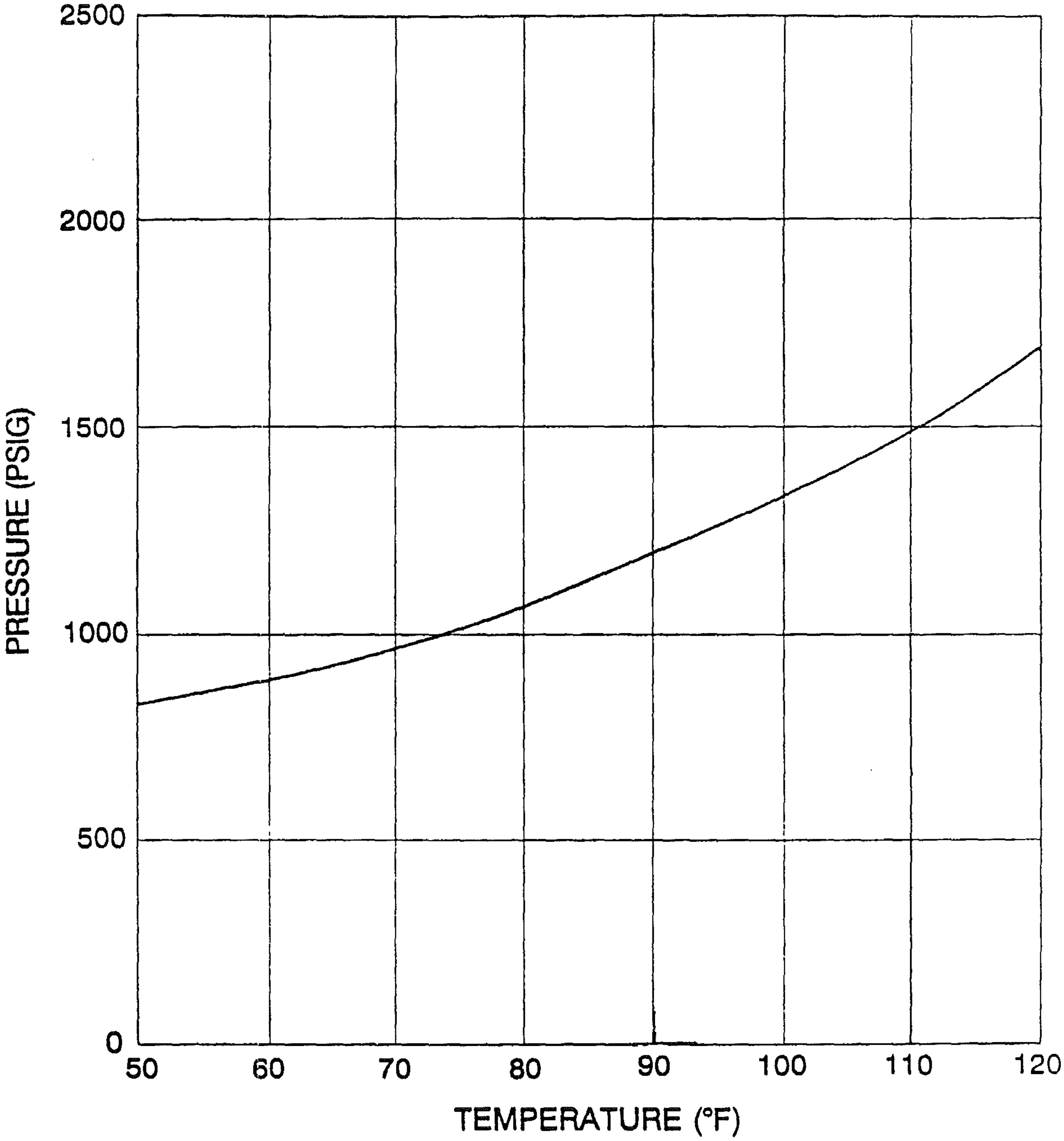

Figure 5.2-3 (Page 1 of 1)
Isolated Loop Pressure - Temperature Limit Curve (LCO 3.4.3)

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

100°F/hr Heatup		100°F/hr Heatup	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**
60	0	245	840
60	554	250	876
65	554	255	917
70	554	260	961
75	554	265	1010
80	554	270	1064
85	554	275	1124
90	554	280	1189
95	554	285	1262
100	554	290	1342
105	554	295	1431
110	554	300	1528
115	554	305	1636
120	554	310	1754
125	554	315	1885
130	554	320	2029
135	554	325	2151
140	555	330	2282
145	557	335	2426
150	560	336.8	2485
155	563		
160	567		
165	573		
170	579		
175	585		
180	593		
185	602		
190	613		
195	624		
200	637		
205	651		
210	667		
215	685		
220	705		
225	727		
230	751		
235	778		
240	807		

100°F/hr Criticality	
T (°F)	**P (psig)**
302	0
302	981
305	1010
310	1064
315	1124
320	1189
325	1262
330	1342
335	1431
340	1528
345	1636
350	1754
355	1885
360	2029
365	2151
370	2282
375	2426
376.8	2485

Leak Test Limit	Temperature (°F)	284	302
	Pressure (psig)	2000	2485

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

Steady State		20°F/hr		40°F/hr		60°F/hr		100°F/hr	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	606	60	563	60	518	60	425
65	621	65	607	65	563	65	519	65	426
70	621	70	608	70	564	70	519	70	427
75	621	75	609	75	565	75	520	75	428
80	621	80	611	80	567	80	522	80	429
85	621	85	612	85	568	85	523	85	430
90	621	90	614	90	570	90	525	90	432
95	621	95	616	95	571	95	526	95	433
100	621	100	618	100	574	100	528	100	435
105	621	105	620	105	576	105	531	105	438
110	621	110	621	110	578	110	533	110	441
115	621	115	621	115	581	115	536	115	444
120	621	120	621	120	585	120	540	120	448
125	621	125	621	125	588	125	544	125	452
130	621	130	621	130	592	130	548	130	457
135	621	135	621	135	597	135	553	135	462
140	621	140	621	140	602	140	558	140	468
145	621	145	621	145	607	145	564	145	475
150	621	150	621	150	614	150	571	150	483
155	621	155	621	155	621	155	578	155	491
160	621	160	621	160	621	160	586	160	501
165	621	165	621	165	621	165	595	165	512
170	621	170	621	170	621	170	606	170	524
175	621	175	621	175	621	175	617	175	537
180	621	180	621	180	621	180	621	180	552
180	747	180	708	180	669	180	630	185	569
185	758	185	720	185	682	185	644	190	588
190	771	190	733	190	696	190	660	195	608
195	784	195	748	195	713	195	677	200	631
200	800	200	765	200	730	200	697	205	657
205	816	205	783	205	750	205	718	210	685
210	835	210	803	210	772	210	742	215	717
215	856	215	825	215	796	215	768	220	752
220	878	220	850	220	823	220	797	225	791
225	903	225	877	225	853	225	830	230	835
230	931	230	908	230	886	230	866	235	883
235	962	235	941	235	922	235	906	240	936
240	996	240	978	240	962	240	950	245	995
245	1033	245	1019	245	1007	245	999	250	1053
250	1075	250	1064	250	1056	250	1053	255	1111

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

Steady State		**20°F/hr**		**40°F/hr**		**60°F/hr**		**100°F/hr**	
T (°F)	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**	**T (°F)**	**P (psig)**
255	1121	255	1114	255	1111	255	1111	260	1169
260	1171	260	1169	260	1169	260	1169	265	1227
265	1227	265	1227	265	1227	265	1227	270	1289
270	1289	270	1289	270	1289	270	1289	275	1357
275	1357	275	1357	275	1357	275	1357	280	1433
280	1433	280	1433	280	1433	280	1433	285	1516
285	1516	285	1516	285	1516	285	1516	290	1608
290	1608	290	1608	290	1608	290	1608	295	1710
295	1710	295	1710	295	1710	295	1710	300	1823
300	1823	300	1823	300	1823	300	1823	305	1947
305	1947	305	1947	305	1947	305	1947	310	2085
310	2085	310	2085	310	2085	310	2085	315	2237
315	2237	315	2237	315	2237	315	2237	320	2405
320	2405	320	2405	320	2405	320	2405	322.1	2485
322.1	2485	322.1	2485	322.1	2485	322.1	2485		

Table 5.2-3 (Page 1 of 1)

Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)

FUNCTION	SETPOINT
OPPS Enable Temperature	347°F
PORV Setpoint	≤ 397 psig

Table 5.2-4 (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
Lower Shell Plate B6903-1[(d)] (Longitudinal)	V	.323	.689	128.49	88.53	.475
	U	.646	.878	118.93	104.42	.771
	W	.986	.996	148.52	147.93	.992
	Y	2.15	1.21	142.18	172.04	1.464
Lower Shell Plate B6903-1[(d)] (Transverse)	V	.323	.689	137.81	94.95	.475
	U	.646	.878	131.84	115.76	.771
	W	.986	.996	179.99	179.27	.992
	Y	2.15	1.21	166.93	201.99	1.464
				SUM:	1104.89	7.404
	$CF = \Sigma(FF^*RT_{NDT}) \div \Sigma(FF^2) = (1104.89) \div (7.404) = 149.2°F$					
Beaver Valley	V	.323	.689	169.30	116.65	.475
Surv. Weld Material 305424[(d)]	U	.646	.878	176.30	154.79	.771
	W	.986	.996	198.99	198.19	.992
	Y	2.15	1.21	189.41	229.19	1.464
				SUM:	698.82	3.702
	$CF = \Sigma(FF^*RT_{NDT}) \div \Sigma(FF^2) = (698.82) \div (3.702) = 188.8°F$					

Notes:

(a) F = Calculated fluence from Beaver Valley Unit 1 capsule Y dosimetry analysis results, (x 10^{19} n/cm^2, E > 1.0 Mev).

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

(c) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ration factor of 1.06.

(d) Data not credible.

Table 5.2-4a (Page 1 of 1)

Calculation of Chemistry Factors[a]
(Based on St. Lucie and Fort Calhoun Surveillance Capsule Data)

Material	Capsule	Capsule f[b]	FF[c]	ΔRT_{NDT}[d]	FF $*\Delta RT_{NDT}$	FF^2
St. Lucie Surveillance Weld Metal Heat 90136	97°	0.627	0.869	72.3	76.1	0.755
	104°	0.909	0.973	67.4	79.7	0.947
	284°	1.41	1.10	68.0	90.9	1.21
				SUM:	246.7	2.91
	CF = $\Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$ = (246.7) ÷ (2.91) = 84.8°F					
Fort Calhoun Surveillance Weld Metal Heat 305414	W-225	0.553	0.834	238	183.0	0.696
	W-265	0.771	0.927	221	194.1	0.859
	W-275	1.28	1.07	219	226.2	1.14
				SUM:	603.3	2.695
	CF = $\Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$ = (603.3) ÷ (2.695) = 223.9°F					

Notes:

(a) Use of St. Lucie and Fort Calhoun Surveillance Capsule Data approved by NRC letter dated February 20, 2002, "BEAVER VALLEY POWER STATION, UNIT 1 – ISSUANCE OF AMENDMENT RE: AMENDED PRESSURE-TEMPERATURE LIMITS (TAC NO. MB2301)."

(b) f = Calculated fluence (x 10^{19} n/cm^2, E > 1.0 Mev) from Reference 2.

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

(d) ΔRT_{NDT} values are the measured 30 ft-lb. shift values taken from Reference 2.

Table 5.2-4b (Page 1 of 1)
St. Lucie and Fort Calhoun Surveillance Weld Data[(a)(b)]

Material	Capsule	Cu	Ni	Irradiated Temperature °F	Fluence 10^{19} n/cm^2	ΔRT_{NDT}
St. Lucie	97°	0.2291	0.0699	546.7	0.627	72.3
Weld Metal	104°	0.2291	0.0699	546.7	0.909	67.4
Heat 90136	284°	0.2291	0.0699	546.7	1.41	68.0
Fort Calhoun	W-225	0.35	0.60	527	0.553	238
Weld Metal	W-265	0.35	0.60	534	0.771	221
Heat 305414	W-275	0.35	0.60	538	1.28	219

Notes:

(a) Use of St. Lucie and Fort Calhoun Surveillance Capsule Data approved by NRC letter dated February 20, 2002, "BEAVER VALLEY POWER STATION, UNIT 1 – ISSUANCE OF AMENDMENT RE: AMENDED PRESSURE-TEMPERATURE LIMITS (TAC NO. MB2301)."

(b) Data contained in this table was obtained from Reference 3.

Table 5.2-5 (Page 1 of 1)

Reactor Vessel Beltline Material Properties

Material Description	Cu(%)	Ni(%)	Chemistry Factor	Initial RT_{NDT}(°F) [(a)]
Intermediate Shell Plate B6607-1	0.14	0.62	100.5	43
Intermediate Shell Plate B6607-2	0.14	0.62	100.5	73
Lower Shell Plate B6903-1	0.21	0.54	147.2	27
Lower Shell Plate B7203-2	0.14	0.57	98.7	20
Intermediate to Lower Shell Weld Seam (Heat 90136) 11-714	0.27	0.07	124.3	-56
Intermediate Longitudinal Shell Weld Seams (Heat 305424) 19-714 A&B	0.28	0.63	191.7	-56
Lower Longitudinal Weld Seams (Heat 305414) 20-714 A&B	0.34	0.61	210.5	-56
Surveillance Weld (Heat 305424)	0.26	0.61	181.6	---

Note:

(a) The initial RT_{NDT} values for the plates and are based on measured data while the weld values are generic.

Table 5.2-6 (Page 1 of 1)

Summary of Adjusted Reference Temperature (ARTs) for 30 EFPY

MATERIAL DESCRIPTION	30 EFPY	
	1/4T ART(°F)(a)	3/4T ART(°F)(a)
Intermediate Shell Plate B6607-1	201.4	175.8
Intermediate Shell Plate B6607-2	231.4	205.8
Lower Shell Plate B7203-2	176.2	151
Lower Shell Plate B6903-1	243.2	205.7
- Using S/C Data(b)	245.7	207.6
Intermediate Shell Longitudinal Weld 19-714A/B	161.9	115.4
- Using S/C Data(b)	159.6	113.8
Intermediate to Lower Shell Circ. Weld 11-714	163.4	131.7
- Using S/C Data (c)	93.0	71.4
Lower Shell Longitudinal Weld 20-714A/B	176.8	125.8
- Using S/C Data(d)	187.5	133.2

Notes:

(a) ART = I + ΔRT_{NDT} + M.

(b) Based on Beaver Valley Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_{Δ}.)

(c) Based on credible St. Lucie Unit 1 surveillance data.

(d) Based on Fort Calhoun Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_{Δ}.)

Table 5.2-7 (Page 1 of 1)

Calculation of Adjusted Reference Temperatures (ARTs) for 30 EFPY

Parameter	VALUES	
Operating Time	30 EFPY	
Material	Plate B6903-1	Plate B6903-1
Location	Lower Shell Plate 1/4T ART(°F)	Lower Shell Plate 3/4T ART(°F)
Chemistry Factor, CF (°F)	149.2	149.2
Fluence (f), n/cm^2 (E>1.0 Mev)[(a)]	2.4194×10^{19}	9.404×10^{18}
Fluence Factor, FF	1.238	.9828
ΔRT_{NDT} = CF x FF(°F)[(c)]	184.7 [(c)]	146.6
Initial RT_{NDT}, I(°F)[(a)]	27	27
Margin, M(°F)	34 [(c)]	34
ART = I+(CF*FF)+M, °F[(b)] per RG 1.99, Revision 2	245.7	207.6

Notes:

(a) Initial RT_{NDT} values are measured values for plate material.

(b) This value was rounded per ASTM E29, using the "Rounding Method."

(c) Based on Beaver Valley Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_{Δ}.)

Table 5.2-8 (Page 1 of 1)

Reactor Vessel Toughness Data (Unirradiated)

COMPONENT	HEAT NO.	CODE NO.	MATERIAL TYPE	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	UPPER SHELF ENERGY (FT-LB)	
									MWD	NMWD
Closure Head Dome	C6213-1B	B6610	A533B CL. 1	.15	---	.010	-40	0*	121	---
Closure Head Seg.	A5518-2	B6611	A533B CL. 1	.14	---	.015	-20	-20*	131	---
Closure Head Flange	ZV3758	---	A508 CL. 2	.08	---	.007	60*	60*	>100	---
Vessel Flange	ZV3661	---	A508 CL. 2	.12	---	.010	60*	60*	166	---
Inlet Nozzle	9-5443	---	A508 CL. 2	.10	---	.008	60*	60*	82.5	---
Inlet Nozzle	9-5460	---	A508 CL. 2	.10	---	.010	60*	60*	94	---
Inlet Nozzle	9-5712	---	A508 CL. 2	.08	---	.007	60*	60*	97	---
Outlet Nozzle	9-5415	---	A508 CL. 2	---	---	.008	60*	60*	97	---
Outlet Nozzle	9-5415	---	A508 CL. 2	---	---	.007	60*	60*	112.5	---
Outlet Nozzle	9-5444	---	A508 CL. 2	.09	---	.007	60*	60*	103	---
Upper Shell	123V339	---	A508 CL. 2	---	---	.010	40	40*	155	---
Inter Shell	C4381-2	B6607-2	A533B CL. 1	.14	.62	.015	-10	73	123	82.5
Inter Shell	C4381-1	B6607-1	A533B CL. 1	.14	.62	.015	-10	43	128.5	90
Lower Shell	C6317-1	B6903-1	A533B CL. 1	.20	.54	.010	-50	27	134	80
Lower Shell	C6293-2	B7203-2	A533B CL. 1	.14	.57	.015	-20	20	129.5	83.5
Trans Ring	123V223	---	A508 CL. 2	---	---	---	30	30*	143	---
Bottom Hd Seg	C4423-3	B6618	A533B CL. 1	.13	---	.008	-30	-29*	124	---
Bottom Hd Dome	C4482-1	B6619	A533B CL. 1	.13	---	.015	-50	-33*	125.5	---
Inter to Lower Shell Weld	90136	---	---	.27	.07	---	---	-56	---	> 100
Inter Shell Long. Weld	305424	---	---	.28	.63	---	---	-56	---	> 100
Lower Shell Long. Weld	305414	---	---	.34	.61	---	---	-56	---	> 100
Weld HAZ				---	---	---	-40	-40	---	136.5

*Estimated Per NRC Standard Review Plan Branch Technical Position MTEB 5-2
MWD – Major Working Direction
NMWD – Normal to Major Working Direction
Note: For evaluation of Inservice Reactor Vessel Irradiation damage assessments, the best estimate chemistry values reported in the latest response to Generic Letter 92-01 or equivalent document are applicable.

Table 5.2-9 (Page 1 of 1)

RT_{PTS} Calculation for Beltline Region Materials at EOL (28 EFPY)

Material	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[a] (°F)	Margin (°F)	$RT_{NDT(U)}$[b] (°F)	RT_{PTS}[c] (°F)
Intermediate Shell Plate B6607-1	3.54	1.329	100.5	133.6	34	43	211
Intermediate Shell Plate B6607-2	3.54	1.329	100.5	133.6	34	73	241
Lower Shell Plate B7203-2	3.54	1.329	98.7	131.2	34	20	185
Lower Shell Plate B6903-1	3.54	1.329	147.2	195.6	34	27	257
→ Using S/C Data[e]	3.54	1.329	149.2	198.3	34	27	259
Inter. Shell Long. Weld 19-714A/B	0.708	0.903	191.7	173.1	65.5	-56	183
→ Using S/C Data[e]	0.708	0.903	188.8	170.5	65.5	-56	180
Lower Shell Long. Weld 20-714A/B	0.708	0.903	210.5	190.1	65.5	-56	200
→ Using S/C Data[f]	0.708	0.903	223.9	202.2	65.5	-56	212
Circumferential Weld 11-714	3.53	1.329	124.3	165.2	65.5	-56	175
→ Using S/C Data[d]	3.53	1.329	84.8	112.3	44	-56	101

Notes:

(a) ΔRT_{PTS} = CF * FF.

(b) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.

(c) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS}$ + Margin (°F).

(d) Based on credible St. Lucie Unit 1 surveillance data.

(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_Δ.

(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_Δ.

Table 5.2-10 (Page 1 of 1)

RT_{PTS} Calculation for Beltline Region Materials at Life Extension (54 EFPY)

Material	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	CF (ºF)	ΔRT_{PTS}[c] (ºF)	Margin (ºF)	$RT_{NDT(U)}$[a] (ºF)	RT_{PTS}[b] (ºF)
Intermediate Shell Plate B6607-1	6.06	1.44	100.5	144.6	34	43	221.6
Intermediate Shell Plate B6607-2	6.06	1.44	100.5	144.6	34	73	251.6
Lower Shell Plate B7203-2	6.09	1.44	98.7	142.1	34	20	196.1
Lower Shell Plate B6903-1	6.09	1.44	147.2	211.9	34	27	272.9
→ Using S/C Data[e]	6.09	1.44	149.2	214.7	34	27	275.7
Inter. Shell Long. Weld 19-714A/B	1.17	1.04	191.7	200.1	65.5	-56	209.6
→ Using S/C Data[e]	1.17	1.04	188.8	197.1	65.5	-56	206.6
Lower Shell Long. Weld 20-714A/B	1.17	1.04	210.5	219.7	65.5	-56	229.2
→ Using S/C Data[f]	1.17	1.04	223.9	233.7	65.5	-56	243.2
Circumferential Weld 11-714	6.07	1.44	124.3	178.8	65.5	-56	188.3
→ Using S/C Data[d]	6.07	1.44	84.8	122.0	44	-56	110.0

Notes:

(a) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (ºF).

(c) $\Delta RT_{PTS} = CF * FF$.

(d) Based on credible St. Lucie Unit 1 surveillance data.

(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_{Δ}.

(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_{Δ}.

Enclosure B
L-12-077

Beaver Valley Power Station, Unit No. 2
Pressure and Temperature Limits Report, Revision 4
(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 2 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 2. 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.

Pressure-temperature limit curves shown in Figure 5.2-7 were 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.

Figures 5.2-1 thru 5.2-6 and Tables 5.2-1 and 5.2-2 are based upon analysis of Capsule W per Reference 2. The tables and curves generated as a result of the Capsule X analysis (Reference 12) and presented in Reference 14 are conservative with respect to those for the Capsule W analysis. As a result, while Tables 5.2-5, 5.2-8, and 5.2-9 are updated with Capsule X fluence data and ART calculations, the pressure-temperature limits provided in Tables 5.2-1 and 5.2-2 and Figures 5.2-1 thru 5.2-6 continue to reflect Capsule W values through 22 EFPY and are bounding.

5.2.1.2 <u>Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)</u>

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 11). 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

5.2 Pressure and Temperature Limits Report

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 <u>OPPS Enable Temperature (LCO 3.4.12)</u>

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.

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 2, and are based upon analysis of Capsule W. 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 22 EFPY. As a result, Tables 5.2-3 and 5.2-4 and Figure 5.2-8 continue to reflect Capsule W values and remain valid for Capsule X up to 22 EFPY.

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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.

5.2.2 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 3 and 4) 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 Pressure and Temperature Limits Report

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 14, 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 2-2 of Reference 14, provides the reactor vessel extended beltline material property table.

Table 5.2-8, taken from Tables 4-5 and 4-6 of Reference 14, provides a summary of the Adjusted Reference Temperature (ARTs) for 22 EFPY.

Table 5.2-9, taken from Tables 4-5 and 4-6 of Reference 14, shows the calculation of ARTs for 22 EFPY.

Table 5.2-10, taken from Table 6 of Reference 5, provides RT_{PTS} values for 32 EFPY.

Table 5.2-11, taken from Table 7 of Reference 13, provides RT_{PTS} values for the Beltline Region Materials at 54 EFPY.

Table 5.2-12, taken from Table 8 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 have been updated to reflect Capsule X analysis and fluence data. This data has not, however, been incorporated into the pressure-temperature limits provided in Figures 5.2-1 thru 5.2-6 and Tables 5.2-1 and 5.2-2, which continue to reflect Capsule W analyses. See Section 5.2.1.1 for additional information.

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5.2.4 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.

2. WCAP-15677, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," J. H. Ledger, August 2001.

3. WCAP-15675, Revision 0, "Analysis of Capsule W from First Energy Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," J. H. Ledger, S. L. Anderson, J. Conermann, August 2001.

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 0, "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.

11. Westinghouse Letter FENOC-04-31, dated April 14, 2004, "LTOPS Setpoint Evaluation for Beaver Valley Unit 2 Capsule W for 22 EFPY – Calculation Note."

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 0, "Analysis of Capsule X from FirstEnergy Nuclear Operating Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," B. N. Burgos, July 2007.

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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.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 22 EFPY: 1/4T, 140°F
3/4T, 129°F
CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

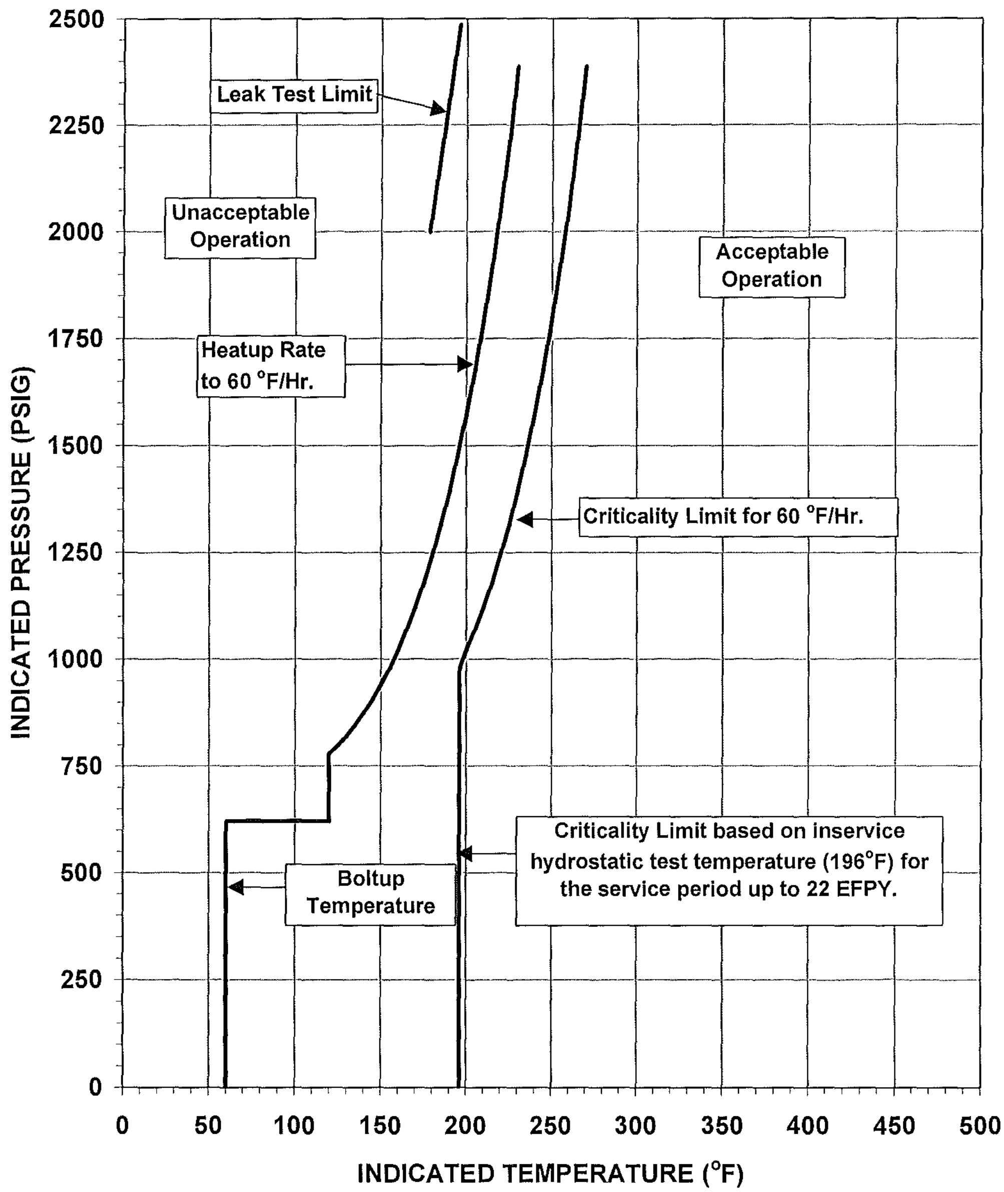

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

NOTE: Values based upon analysis of Capsule W for 22 EFPY and are bounding for Capsule X at 22 EFPY (see Section 5.2.1.1). As a result, ART values shown do not coincide with Tables 5.2-7 and 5.2-8.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 22 EFPY: 1/4T, 140°F
3/4T, 129°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 0°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

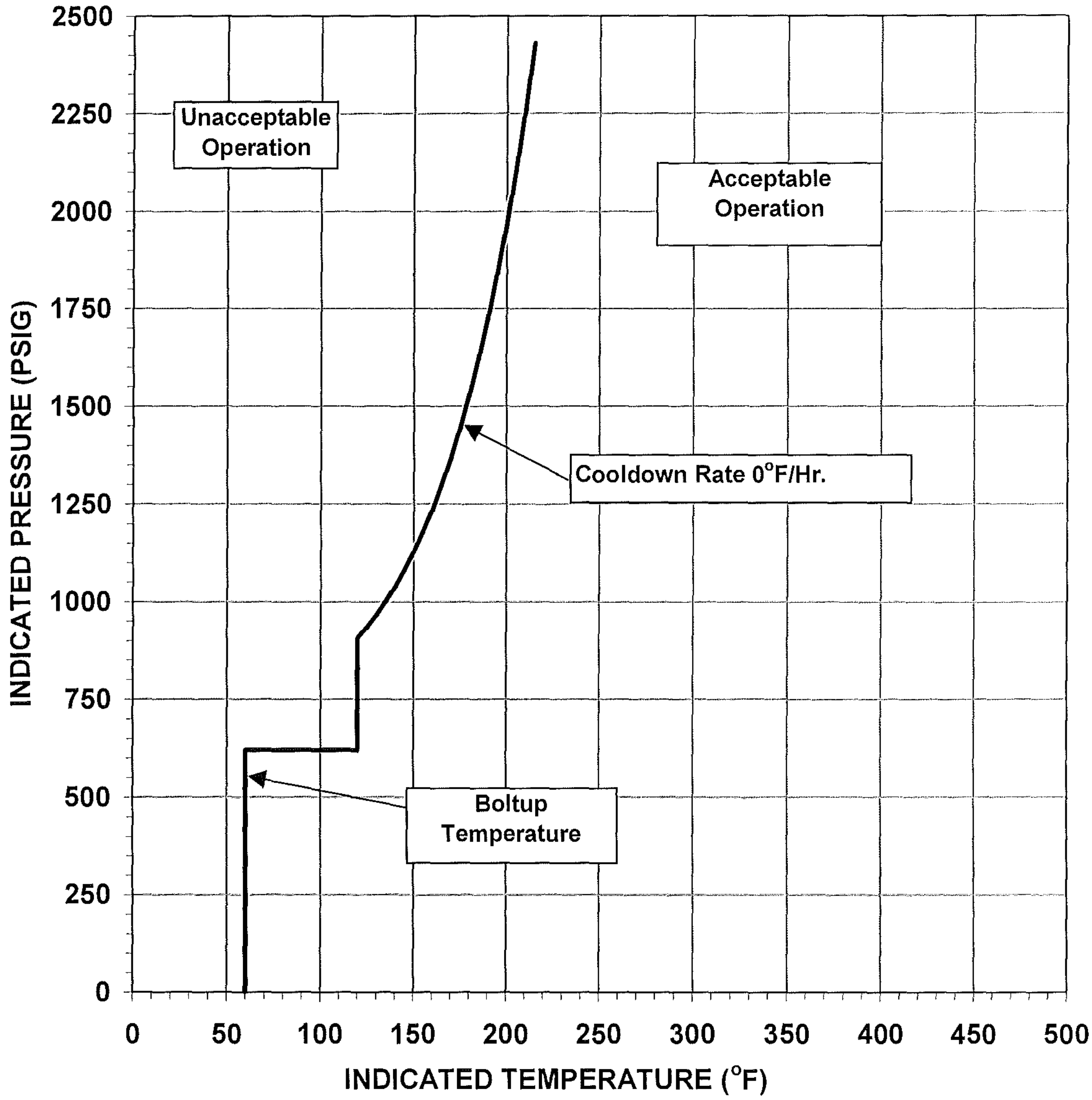

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 0°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

NOTE: Values based upon analysis of Capsule W for 22 EFPY and are bounding for Capsule X at 22 EFPY (see Section 5.2.1.1). As a result, ART values shown do not coincide with Tables 5.2-7 and 5.2-8.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 22 EFPY: 1/4T, 140°F
3/4T, 129°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

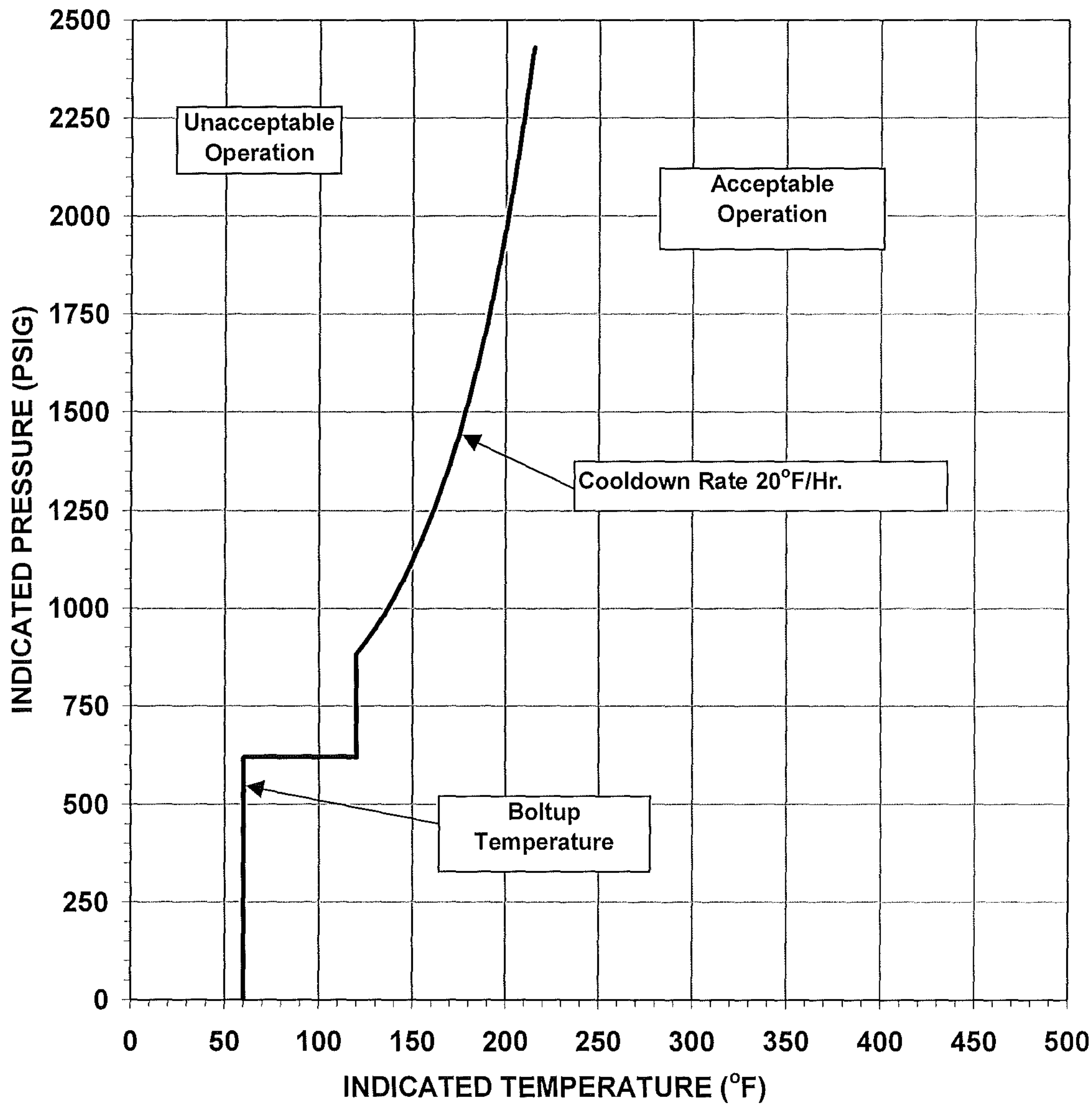

Figure 5.2-3 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 20°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

NOTE: Values based upon analysis of Capsule W for 22 EFPY and are bounding for Capsule X at 22 EFPY (see Section 5.2.1.1). As a result, ART values shown do not coincide with Tables 5.2-7 and 5.2-8.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 22 EFPY: 1/4T, 140°F
3/4T, 129°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

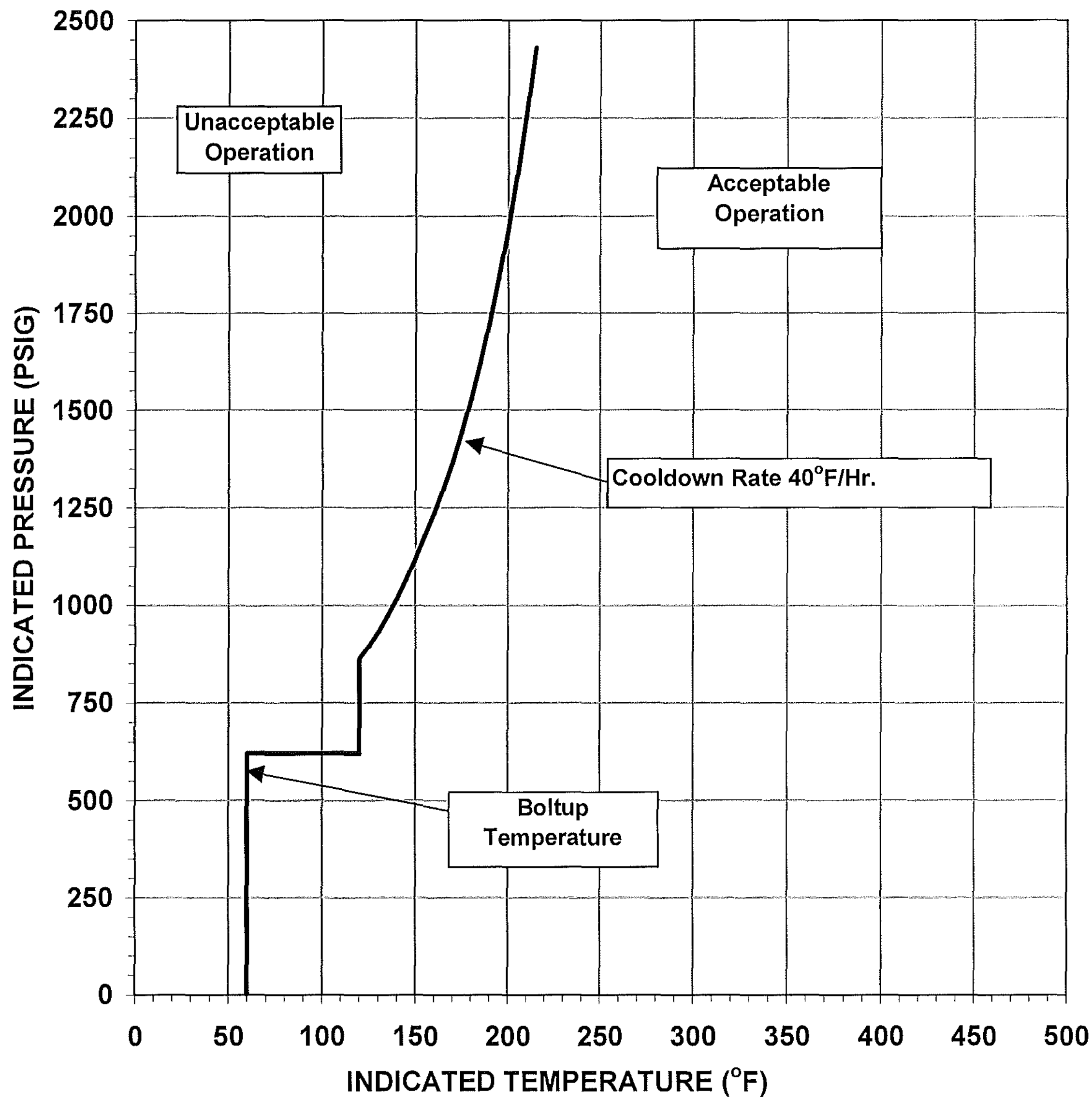

Figure 5.2-4 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 40°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

NOTE: Values based upon analysis of Capsule W for 22 EFPY and are bounding for Capsule X at 22 EFPY (see Section 5.2.1.1). As a result, ART values shown do not coincide with Tables 5.2-7 and 5.2-8.

<u>MATERIAL PROPERTY BASIS</u>

LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

LIMITING ART VALUES AT 22 EFPY: 1/4T, 140°F
3/4T, 129°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

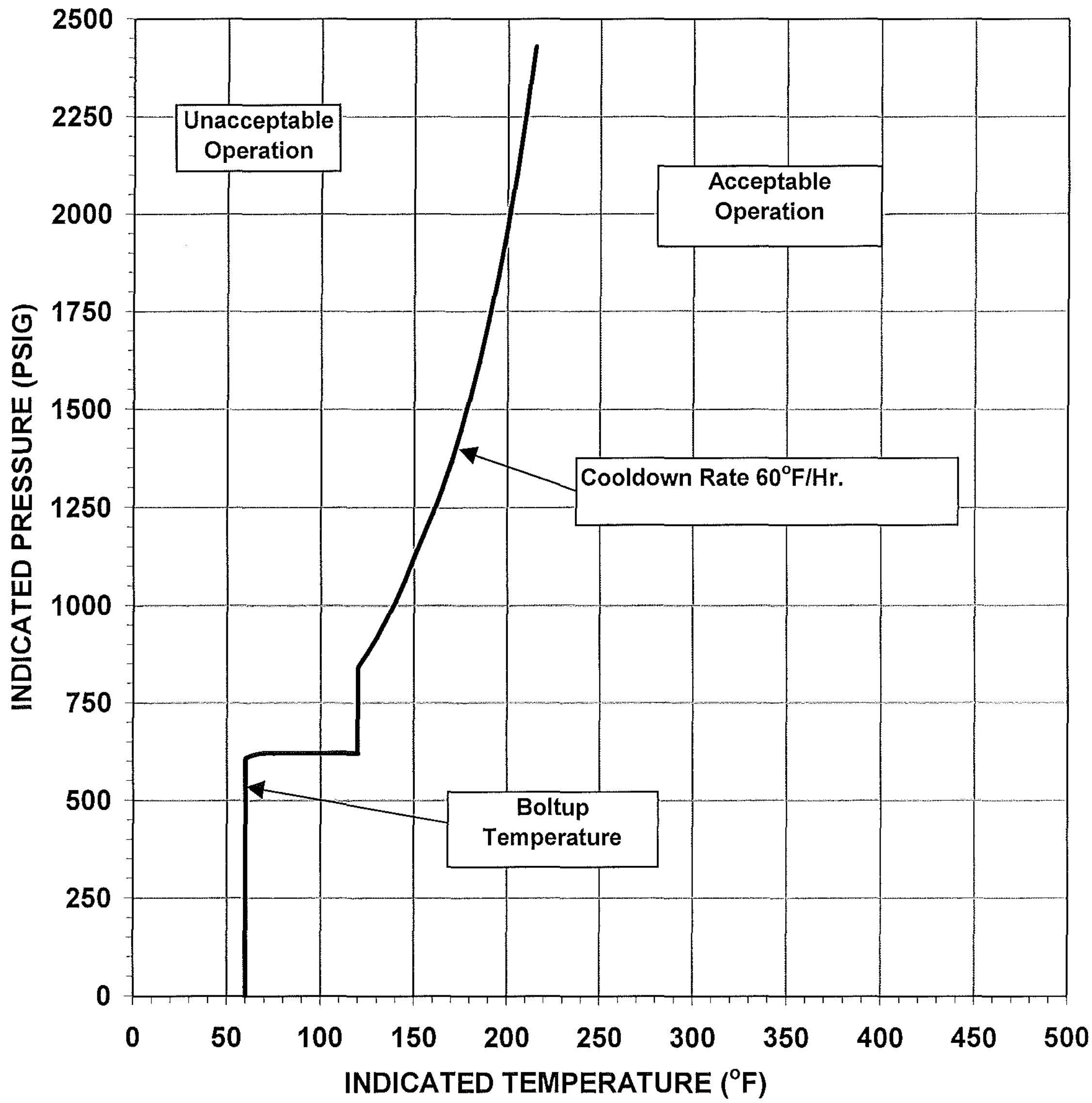

Figure 5.2-5 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 60°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

NOTE: Values based upon analysis of Capsule W for 22 EFPY and are bounding for Capsule X at 22 EFPY (see Section 5.2.1.1). As a result, ART values shown do not coincide with Tables 5.2-7 and 5.2-8.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1
LIMITING ART VALUES AT 22 EFPY: 1/4T, 140°F
3/4T, 129°F
CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

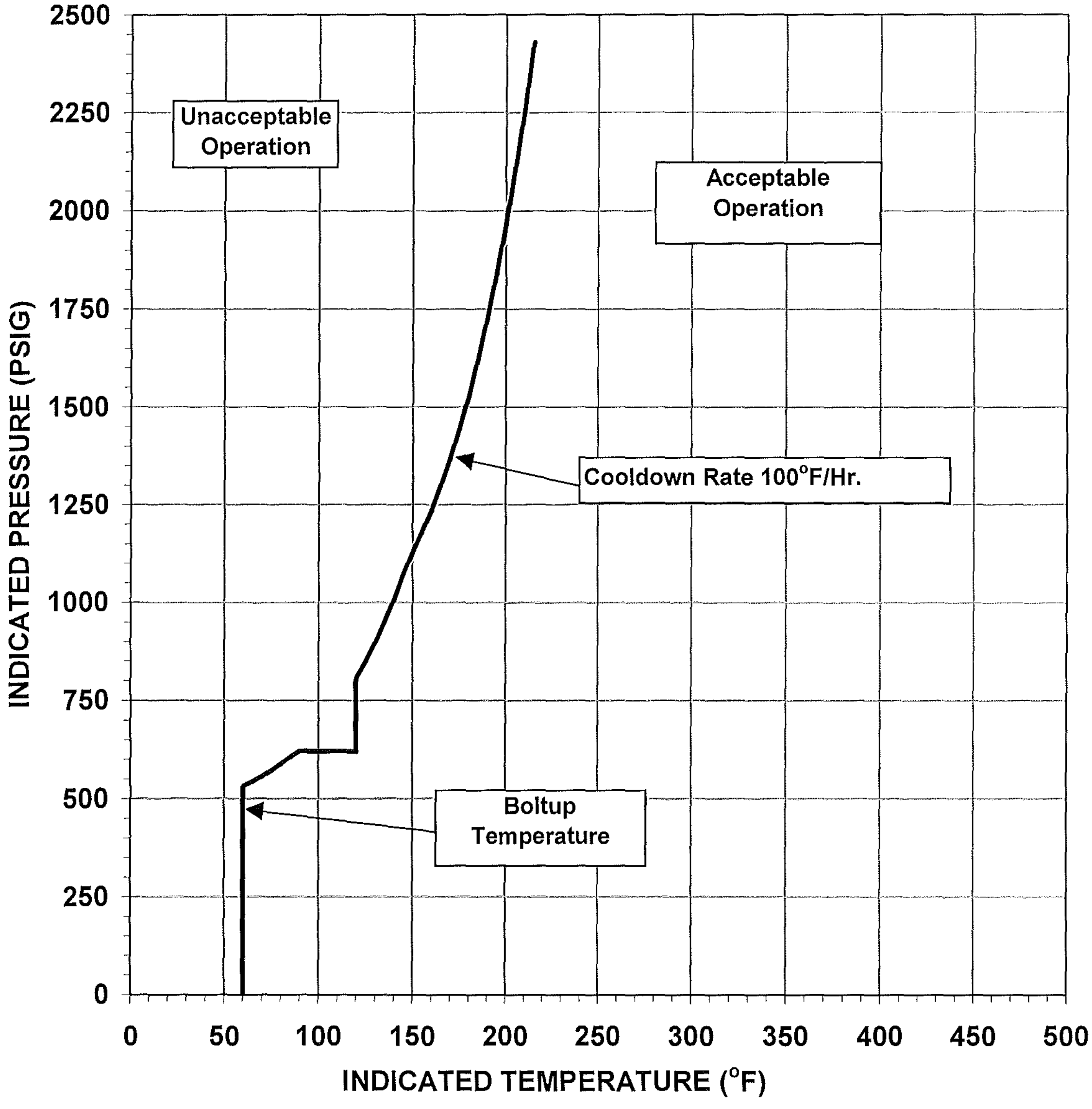

Figure 5.2-6 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 100°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

NOTE: Values based upon analysis of Capsule W for 22 EFPY and are bounding for Capsule X at 22 EFPY (see Section 5.2.1.1). As a result, ART values shown do not coincide with Tables 5.2-7 and 5.2-8.

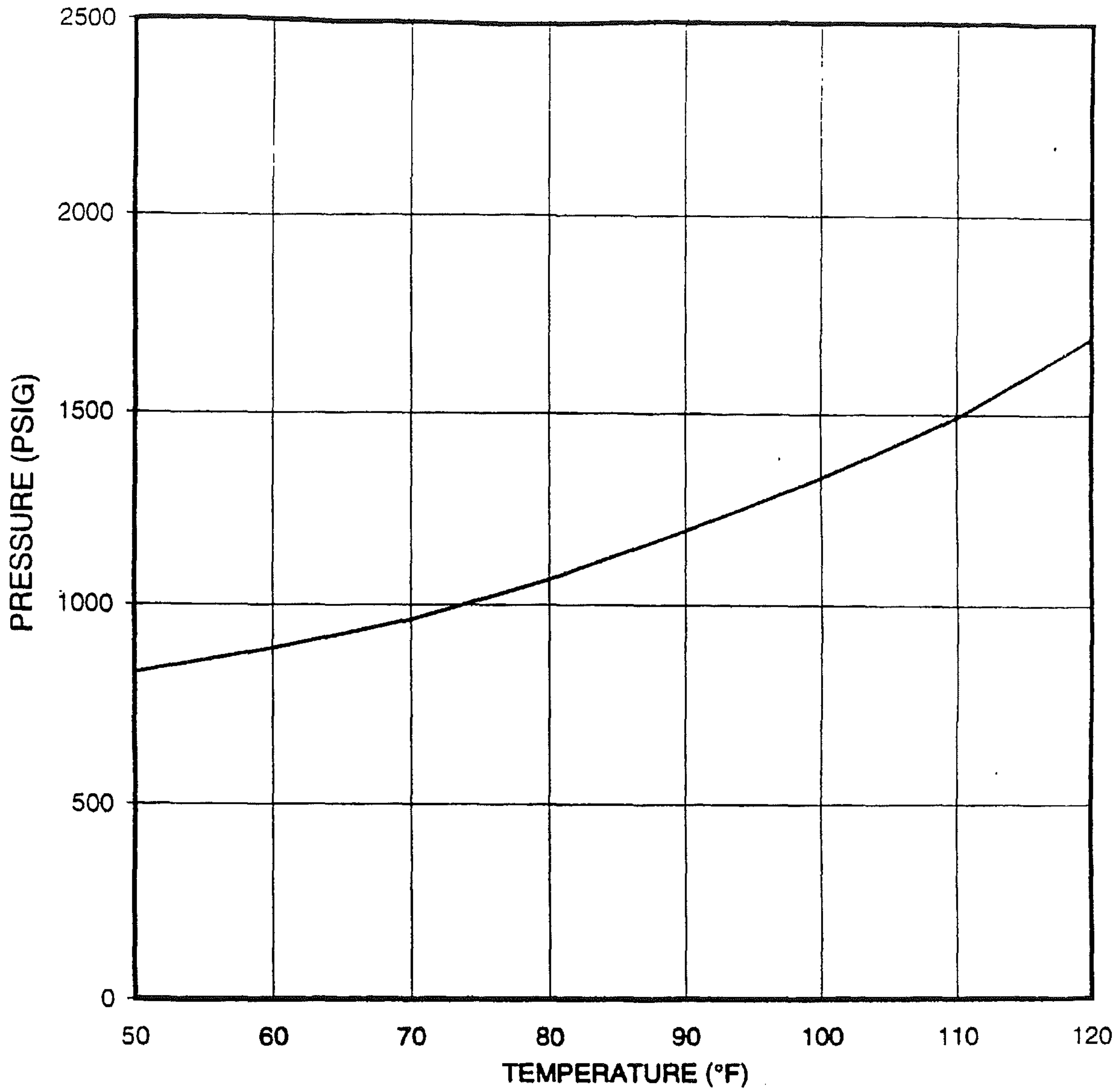

Figure 5.2-7 (Page 1 of 1)
Isolated Loop Pressure – Temperature Limit Curve (LCO 3.4.3)

See Table 5.2-4 for RCP restrictions.

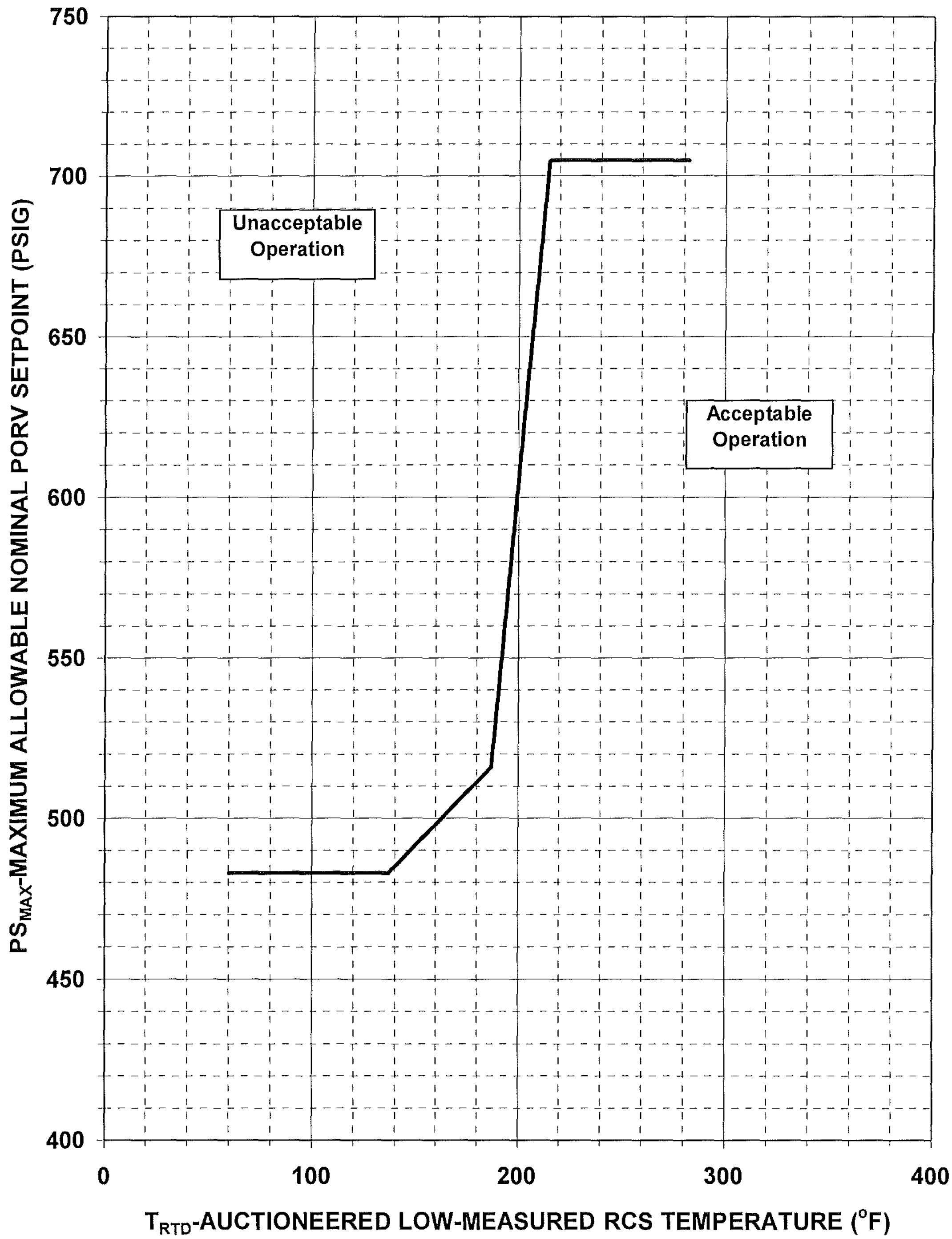

Figure 5.2-8 (Page 1 of 1)
Maximum Allowable Nominal PORV Setpoint for the
Overpressure Protection System (LCO 3.4.12)

NOTE: Data shown reflects analysis for Capsule W thru 22 EFPY and is the same or bounding for Capsule X thru 22 EFPY (See Section 5.2.1.3).

Table 5.2-1 (Page 1 of 1)
Heatup Curve Data Points for 22 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	196	0	178	2000
60	621	196	621	196	2485
65	621	196	621		
70	621	196	621		
75	621	196	621		
80	621	196	621		
85	621	196	621		
90	621	196	621		
95	621	196	621		
100	621	196	621		
105	621	196	621		
110	621	196	621		
115	621	196	621		
120	621	196	779		
120	621	196	799		
120	779	196	821		
125	799	196	846		
130	821	196	874		
135	846	196	905		
140	874	196	940		
145	905	196	978		
150	940	200	1021		
155	978	205	1068		
160	1021	210	1120		
165	1068	215	1178		
170	1120	220	1242		
175	1178	225	1312		
180	1242	230	1390		
185	1312	235	1476		
190	1390	240	1571		
195	1476	245	1675		
200	1571	250	1791		
205	1675	255	1919		
210	1791	260	2060		
215	1919	265	2215		
220	2060	270	2387		
225	2215				
230	2387				

NOTE: Data shown reflects analysis for Capsule W thru 22 EFPY and is the same or bounding for Capsule X thru 22 EFPY (See Section 5.2.1.3).

Table 5.2-2 (Page 1 of 1)
Cooldown Curve Data Points for 22 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	608	532
65	621	621	621	618	544
70	621	621	621	621	557
75	621	621	621	621	572
80	621	621	621	621	588
85	621	621	621	621	606
90	621	621	621	621	621
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	907	884	862	842	807
125	935	914	895	877	849
130	966	948	932	917	897
135	1001	985	972	961	949
140	1039	1026	1017	1010	1007
145	1081	1072	1066	1064	1071
150	1127	1122	1121	1123	1127
155	1179	1178	1179	1179	1179
160	1235	1235	1235	1235	1235
165	1298	1298	1298	1298	1298
170	1367	1367	1367	1367	1367
175	1444	1444	1444	1444	1444
180	1528	1528	1528	1528	1528
185	1622	1622	1622	1622	1622
190	1725	1725	1725	1725	1725
195	1839	1839	1839	1839	1839
200	1966	1966	1966	1966	1966
205	2105	2105	2105	2105	2105
210	2259	2259	2259	2259	2259
215	2430	2430	2430	2430	2430

NOTE: Data shown reflects analysis for Capsule W thru 22 EFPY and is the same or bounding for Capsule X thru 22 EFPY (See Section 5.2.1.3).

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

NOTE: Data shown reflects analysis for Capsule W thru 22 EFPY and is the same or bounding for Capsule X thru 22 EFPY (See Section 5.2.1.3).

Table 5.2-4 (Page 1 of 1)

Reactor Coolant Pump Restrictions

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

NOTE: Data shown reflects analysis for Capsule W thru 22 EFPY and is the same or bounding for Capsule X thru 22 EFPY (See Section 5.2.1.3).

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

Material	Capsule	Capsule f[b]	FF[c]	ΔRT_{NDT}[d]	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell Plate B9004-2 (Longitudinal)	U	0.608	0.861	24.0	20.66	0.741
	V	2.629	1.259	56.0	70.50	1.585
	W	3.625	1.335	71.0	94.79	1.782
	X	5.601	1.424	98.0	139.55	2.028
Intermediate Shell Plate B9004-2 (Transverse)	U	0.608	0.861	17.7	15.24	0.741
	V	2.629	1.259	46.1	58.04	1.585
	W	3.625	1.335	63.4	84.64	1.782
	X	5.601	1.424	104.1	148.24	2.028
				SUM:	631.66	12.272
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = 51.5°F$					
Surveillance Weld Metal 83642	U	0.608	0.861	4.1[e]	3.53	0.741
	V	2.629	1.259	25.7[e]	32.36	1.585
	W	3.625	1.335	6.0[e]	8.01	1.782
	X	5.601	1.424	22.9[e]	32.61	2.028
				SUM:	76.51	6.136
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = 12.5°F$					

Notes:

(a) Regulatory Guide 1.99, Revision 2, Position 2.1.

(b) f = fluence (10^{19} n/cm^2); Fluence values were taken from Capsule W analysis (Reference 12).

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

(d) ΔRT_{NDT} values are the measured 30 ft-lb. shift values for BVPS-2 taken from Reference 12.

(e) The surveillance weld metal ΔRT_{NDT} values have been conservatively adjusted by a ratio factor of 1.0; the calculated ratio was 0.905, which would result in a lower calculated CF.

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
	B9011-2	2V2437-02-001	0.13	0.88	60 (Gen)
	B9011-3	2V2445-02-003	0.13	0.84	70
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.

Table 5.2-8 (Page 1 of 1)

Summary of Adjusted Reference Temperature (ARTs) for 22 EFPY[a]

MATERIAL DESCRIPTION	Method Used To Calculate the CF[b]	22 EFPY ART	
		1/4T ART (°F)	3/4T ART (°F)
Intermediate Shell Plate B9004-1	Position 1.1	139	128
Intermediate Shell Plate B9004-2	Position 1.1	115	103
	Position 2.1	114	101
Lower Shell Plate B9005-1	Position 1.1	119	105
Lower Shell Plate B9005-2	Position 1.1	116	104
Vessel Beltline Welds[c]	Position 1.1	47	29
	Position 2.1	-2	-9

Notes:

(a) Table updated to reflect Capsule X analysis per Reference 14; 1/4T and 3/4T ART values for B9004-1 will differ from as described on Figures 5.2-1 thru 5.2-6. See Section 5.2.1.1 for additional information.

(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 22 EFPY[(a)]

PARAMETER	VALUES	
Operating Time	22 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)]	1.515	0.589
Fluence Factor, FF	1.115	0.852
ΔRT_{NDT} = CF x FF(°F)	45.16	34.50
Intitial RT_{NDT}, I(°F)	60	60
Margin, M(°F)	34	34
ART, per Regulatory Guide 1.99, Revision 2	139	128

Notes:

(a) Table updated to reflect Capsule X analysis per Reference 14; 1/4T and 3/4T ART values for B9004-1 will differ from as described on Figures 5.2-1 thru 5.2-6. See Section 5.2.1.1 for additional information.

(b) Fluence (f), is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 2.43 at 22 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 EOL (32 EFPY)

Material	Method	f [(a)] Fluence	FF[(b)]	CF (°F)	ΔRT_{PTS} (°F)	Margin (°F)	$RT_{NDT(U)}$ [(c)] (°F)	RT_{PTS} (°F)
Intermediate Shell Plate B9004-1	RG 1.99, R2, P1.1	3.847	1.348	40.5	54.6	34	60	149
Intermediate Shell Plate B9004-2	RG 1.99, R2, P1.1	3.847	1.348	37.0	49.9	34	40	124
	RG 1.99, R2, P2.1	3.847	1.348	41.9	56.5	17	40	114
Lower Shell Plate B9005-1	RG 1.99, R2, P1.1	3.847	1.348	51.0	68.7	34	28	131
Lower Shell Plate B9005-2	RG 1.99, R2, P1.1	3.847	1.348	44.0	59.3	34	33	126
Vessel Beltline Welds	RG 1.99, R2, P1.1	3.847	1.348	34.4	46.4	46.4	-30	63
	RG 1.99, R2, P2.1	3.847	1.348	10.6	14.3	14.3	-30	-1

<u>Notes:</u>

(a) f = peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV) at 32 EFPY (45° fluence for longitudinal welds)

(b) $FF = f^{(0.28 - 0.10 \log f)}$

(c) $RT_{NDT(U)}$ values are measured values.

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

Table 5.2-11 (Page 1 of 1)

RT_{PTS} Calculation for Beltline Region Materials at Life Extension (54 EFPY)

Material	RG Pos.	Fluence ($x10^{19}$ n/cm^2 E>1.0 MeV)	FF[(a)]	CF (°F)	ΔRT_{PTS}[(b)] (°F)	Margin[(c)] (°F)	RT_{NDT}[(d)] (°F)	RT_{PTS}[(e)] (°F)
Intermediate Shell Plate B9004-1	1.1	6.22	1.4429	40.5	58.4	34.0	60.0	152.4
Intermediate Shell Plate B9004-2	1.1	6.22	1.4429	37.0	53.4	34.0	40.0	127.4
	2.1	6.22	1.4429	51.5	74.3	17.0	40.0	131.3
Lower Shell Plate B9005-1	1.1	6.29	1.4449	51.0	73.7	34.0	28.0	135.7
Lower Shell Plate B9005-2	1.1	6.29	1.4449	44.0	63.6	34.0	33.0	130.6
Lower Shell Longitudinal Welds 101-142 A&B (Heat 83642)	1.1	1.78	1.1584	34.4	39.8	39.8	-30.0	49.7
	2.1	1.78	1.1584	12.5	14.5	14.5	-30.0	-1.0
Intermediate Shell Longitudinal Weld 101-124 A&B (Heat 83642)	1.1	1.76	1.1554	34.4	39.7	39.7	-30.0	49.5
	2.1	1.76	1.1554	12.5	14.4	14.4	-30.0	-1.1
Intermediate to Lower Shell Girth Weld 101-171 (Heat 83642)	1.1	6.24	1.4435	34.4	49.7	49.7	-30.0	69.3
	2.1	6.24	1.4435	12.5	18.0	18.0	-30.0	6.1

Notes:

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

(b) ΔRT_{PTS} = CF * FF.

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

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

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

Table 5.2-12 (Page 1 of 2)

RT_{PTS} Calculation for Extended Beltline Region Materials at Life Extension (54 EFPY)

Material	RG Pos.	Fluence ($x10^{19}$ n/cm^2 E>1.0 MeV)	FF[(a)]	CF (°F)	ΔRT_{PTS}[(b)] (°F)	Margin[(c)] (°F)	RT_{NDT}[(d)] (°F)	RT_{PTS}[(e)] (°F)
Upper Shell Plate B9003-1	1.1	0.4920	0.8022	91.00	73.0	34.0	50.0	157.0
Upper Shell Plate B9003-2	1.1	0.4920	0.8022	83.00	66.6	34.0	60.0	160.6
Upper Shell Plate B9003-3	1.1	0.4920	0.8022	91.00	73.0	34.0	50.0	157.0
Upper Shell Long Weld 51912-3490	1.1	0.4920	0.8022	73.71	59.1	56.0	-50.0	65.1
Upper Shell Long Weld 51912-3536	1.1	0.4920	0.8022	73.71	59.1	56.0	-70.0	45.1
Upper Shell Long Weld EAIB	1.1	0.4920	0.8022	27.00	21.7	40.3	10.0	72.0
Upper Shell Long Weld IAGA	1.1	0.4920	0.8022	41.00	32.9	32.9	-30.0	35.8
Upper Shell Long Weld BOHB	1.1	0.4920	0.8022	68.00	54.5	64.3	10	128.8
Upper Shell Long Weld BAOED	1.1	0.4920	0.8022	27.00	21.7	21.7	-50.0	-6.7
Upper to Inter Girth Weld 4P5174	1.1	0.5950	0.8546	122.00	104.3	56.0	-50.0	110.3
Upper to Inter Girth Weld 51922	1.1	0.5950	0.8546	68.00	58.1	65.5	-56.0	67.6
Upper to Inter Girth Weld AAGC	1.1	0.5950	0.8546	41.00	35.0	35.0	-70.0	0.1
Upper to Inter Girth Weld KOIB	1.1	0.5950	0.8546	41.00	35.0	35.0	-60.0	10.1
Inlet Nozzle B9011-1	1.1	0.0490	0.2895	77.00	22.3	22.3	60.0	104.6
Inlet Nozzle B9011-2	1.1	0.0490	0.2895	96.00	27.8	43.9	60.0	131.7
Inlet Nozzle B9011-3	1.1	0.0490	0.2895	96.00	27.8	27.8	70.0	125.6
Inlet Nozzle Welds 4P5174	1.1	0.0490	0.2895	122.00	35.3	35.3	-50.0	20.6
Inlet Nozzle Welds LOHB	1.1	0.0490	0.2895	41.00	11.9	11.9	-60.0	-36.3
Inlet Nozzle Welds HABJC	1.1	0.0490	0.2895	27.00	7.8	7.8	-70.0	-54.4
Inlet Nozzle Welds BABBD	1.1	0.0490	0.2895	27.00	7.8	7.8	-70.0	-54.4
Inlet Nozzle Welds FABGC	1.1	0.0490	0.2895	41.00	11.9	11.9	-80.0	-56.3
Inlet Nozzle Welds EOBC	1.1	0.0490	0.2895	27.00	7.8	7.8	-60.0	-44.4
Inlet Nozzle Welds FAAFC	1.1	0.0490	0.2895	95.00	27.5	27.5	-60.0	-5.0
Inlet Nozzle Welds CCJC	1.1	0.0490	0.2895	27.00	7.8	7.8	-60.0	-44.4
Inlet Nozzle Welds FAGB	1.1	0.0490	0.2895	27.00	7.8	7.8	-30.0	-14.4

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RT_{PTS} Calculation for Extended Beltline Region Materials at Life Extension (54 EFPY)

Material	RG Pos.	Fluence ($x10^{19}$ n/cm^2 E>1.0 MeV)	FF[(a)]	CF (°F)	ΔRT_{PTS}[(b)] (°F)	Margin[(c)] (°F)	RT_{NDT}[(d)] (°F)	RT_{PTS}[(e)] (°F)
Inlet Nozzle Welds BAOED	1.1	0.0490	0.2895	27.00	7.8	7.8	-50.0	-34.4
Outlet Nozzle B9012-1	1.1	0.0234	0.1894	94.00	17.8	17.8	-10.0	25.6
Outlet Nozzle B9012-2	1.1	0.0234	0.1894	94.50	17.9	17.9	-10.0	25.8
Outlet Nozzle B9012-3	1.1	0.0234	0.1894	93.50	17.7	17.7	-10.0	25.4
Outlet Nozzle Weld BABBD	1.1	0.0234	0.1894	27.00	5.1	5.1	-70.0	-59.8
Outlet Nozzle Weld FAAFC	1.1	0.0234	0.1894	95.00	18.0	18.0	-60.0	-24.0
Outlet Nozzle Weld HAAEC	1.1	0.0234	0.1894	41.00	7.8	7.8	-80.0	-64.5
Outlet Nozzle Weld HABJC	1.1	0.0234	0.1894	27.00	5.1	5.1	-70.0	-59.8
Outlet Nozzle Weld HAGB	1.1	0.0234	0.1894	27.00	5.1	5.1	-40.0	-29.8
Outlet Nozzle Weld GACJC	1.1	0.0234	0.1894	41.00	7.8	7.8	-80.0	-64.5
Outlet Nozzle Weld JAHB	1.1	0.0234	0.1894	41.00	7.8	7.8	-40.0	-24.5

Notes:

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

(b) ΔRT_{PTS} = CF * FF.

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

(d) Initial RT_{NDT} value for the upper shell forging is a measured value. All other values are generic.

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