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September 29, 2003
L-03-151

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

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Original Issuance of the Pressure and Temperature Limits Report

FirstEnergy Nuclear Operating Company (FENOC) hereby submits the initial issuance of the Pressure and Temperature Limits Report (PTLR) for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2. The PTLR for each BVPS unit is provided as an Enclosure to this letter.

The initial issuance, Revision 0, of the PTLR for each unit is being provided in accordance with Technical Specification 6.9.6, "PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)." Technical Specification 6.9.6.c states that the PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto. Creation of the PTLR was NRC approved by License Amendment Nos. 256 (Unit No. 1) and 138 (Unit No. 2) dated July 15, 2003. The BVPS Unit Nos. 1 and 2 PTLRs were issued as part of Revision Nos. 35 (Unit No. 1) and 33 (Unit No. 2) of the respective unit's Licensing Requirements Manual, coincident with the implementation of the amendments on September 9, 2003.

No new commitments are contained in this submittal. If there are any questions concerning this matter, please contact Mr. Larry R. Freeland, Manager, Regulatory Affairs/Performance Improvement at 724-682-5284.

Sincerely,

L.W. Pearce

L. William Pearce

Enclosures:
1. BVPS Unit 1 Revision 0 of the Pressure and Temperature Limits Report
2. BVPS Unit 2 Revision 0 of the Pressure and Temperature Limits Report

A001

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

SECTION 4.2 PRESSURE AND TEMPERATURE LIMITS REPORT

BVPS-1 Technical Specification to PTLR Cross-Reference			
Technical Specification	PTLR		
	Section	Figure	Table
3.1.2.4	N/A	N/A	4.2-3
3.4.1.3	N/A	N/A	4.2-3
3.4.9.1	4.2.1.1	4.2-1 4.2-2	N/A
3.4.9.3	4.2.1.2 4.2.1.3	N/A	4.2-3
3.5.3	N/A	N/A	4.2-3
3.5.4.1.2	N/A	N/A	4.2-3
3.10.3	N/A	4.2-1 4.2-2	N/A

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

This Pressure and Temperature Limits Report (PTLR) for Unit 1 has been prepared in accordance with the requirements of Technical Specification 6.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) addressed in, or make reference to, this report are listed below:

TS 3.1.2.4 Reactivity Control Systems – Charging Pumps – Operating,

TS 3.4.1.3 Reactor Coolant System – Shutdown,

TS 3.4.9.1 Reactor Coolant System - Pressure/Temperature Limits,

TS 3.4.9.3 Overpressure Protection Systems,

TS 3.5.3 ECCS Subsystems – $T_{avg} < 350°F$,

TS 3.5.4.1.2 Boron Injection Tank < 350°F, and

TS 3.10.3 Special Test Exceptions - Pressure/Temperature Limitations -Reactor Criticality.

4.2.1 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".

4.2.1.1 RCS Pressure and Temperature (P/T) Limits (TS 3.4.9.1)

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

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 4.2-1 and Table 4.2-1. The RCS P/T limits for cooldown are shown in Figure 4.2-2 and Table 4.2-2. These limits are defined in Reference 2. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 4.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 4.2-1 and 4.2-2 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 4.2-3 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.

4.2.1.2 Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

The power operated relief valves (PORVs) shall each have maximum lift setting and enable temperature in accordance with Table 4.2-3. 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 4.2.1. The PORV lift setting shown in Table 4.2-3 accounts for appropriate instrument error.

4.2.1.3 OPPS Enable Temperature (TS 3.4.9.3)

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 343°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 308°F.

As the arming temperature is higher and, therefore, more conservative than the calculated enable temperature, the OPPS enable temperature, as shown in Table 4.2-3, 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 4.2-3, 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 TS 3.4.9.3. 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.

4.2.1.4 Reactor Vessel Boltup Temperature (TS 3.4.9.1)

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.

4.2.2 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 4.2-1 and 4.2-2, and Tables 4.2-1 and 4.2-2. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching 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.

4.2.3 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 4.2-4, taken from Reference 5, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2-4a, taken from Reference 2, shows the Calculation of Chemistry Factors based on St. Lucie and Fort Calhoun Surveillance Capsule Data.

Table 4.2-4b, taken from Reference 3, shows the St. Lucie and Fort Calhoun Surveillance Weld Data.

Table 4.2-5, taken from Reference 2, provides the reactor vessel beltline material property table.

Table 4.2-6, taken from Reference 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 22 EFPY.

Table 4.2-7, taken from Reference 2, shows the calculation of ARTs for 22 EFPY.

Table 4.2-8 shows the Reactor Vessel Toughness Data (Unirradiated).

Table 4.2-9, taken from Reference 5, provides RT_{PTS} values for 28 EFPY.

Table 4.2-10, taken from Reference 5, provides RT_{PTS} values for 45 EFPY.

4.2.4 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-8475, "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. Westinghouse Report, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule", Revision 1, April 2001.

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

PRESSURE AND TEMPERATURE LIMITS REPORT

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE
LIMITING ART VALUES AT 22 EFPY: 1/4T, 233°F
3/4T, 196°F

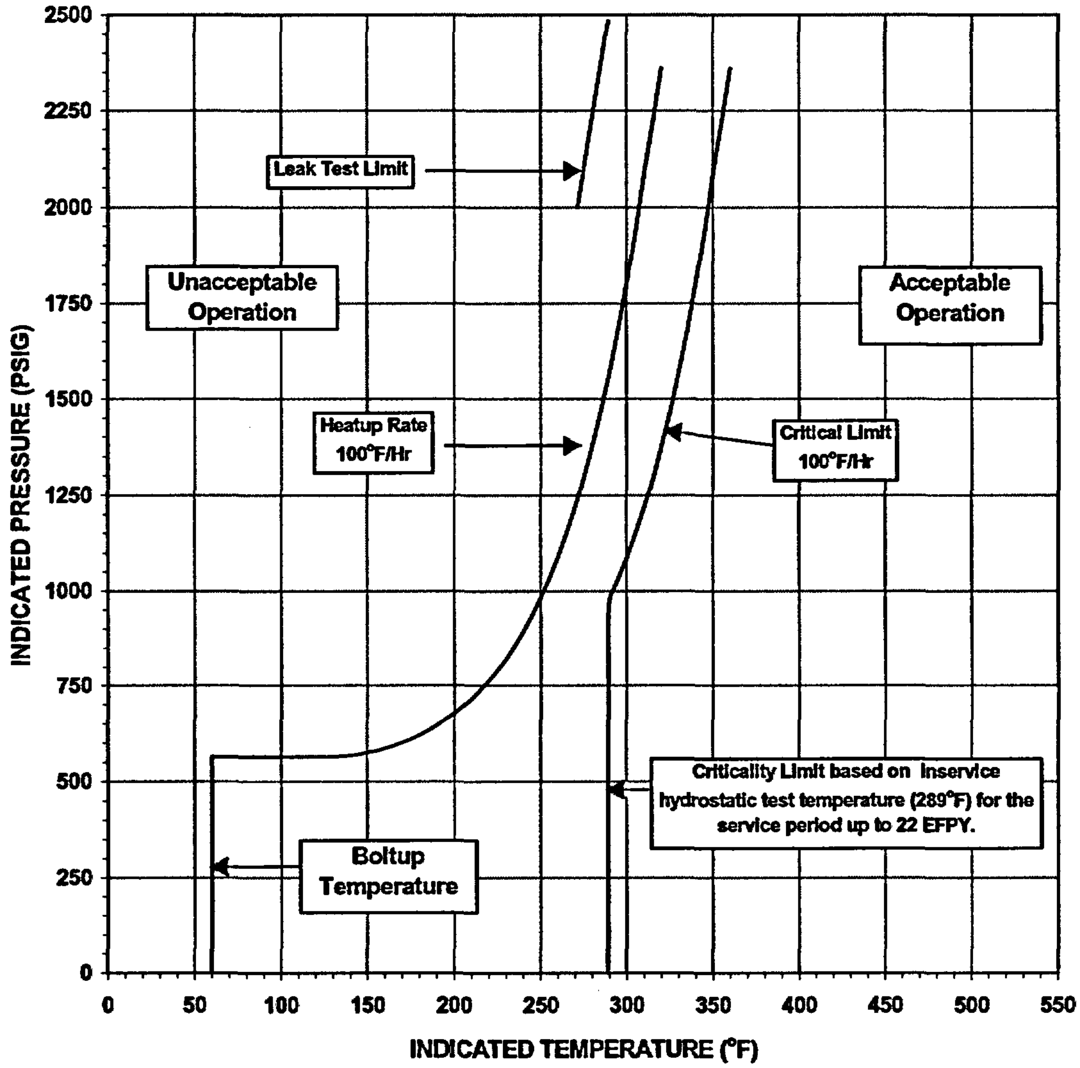

Figure 4.2-1
Reactor Coolant System Heatup
Limitations Applicable for the First 22 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 22 EFPY: 1/4T, 233°F
3/4T, 196°F

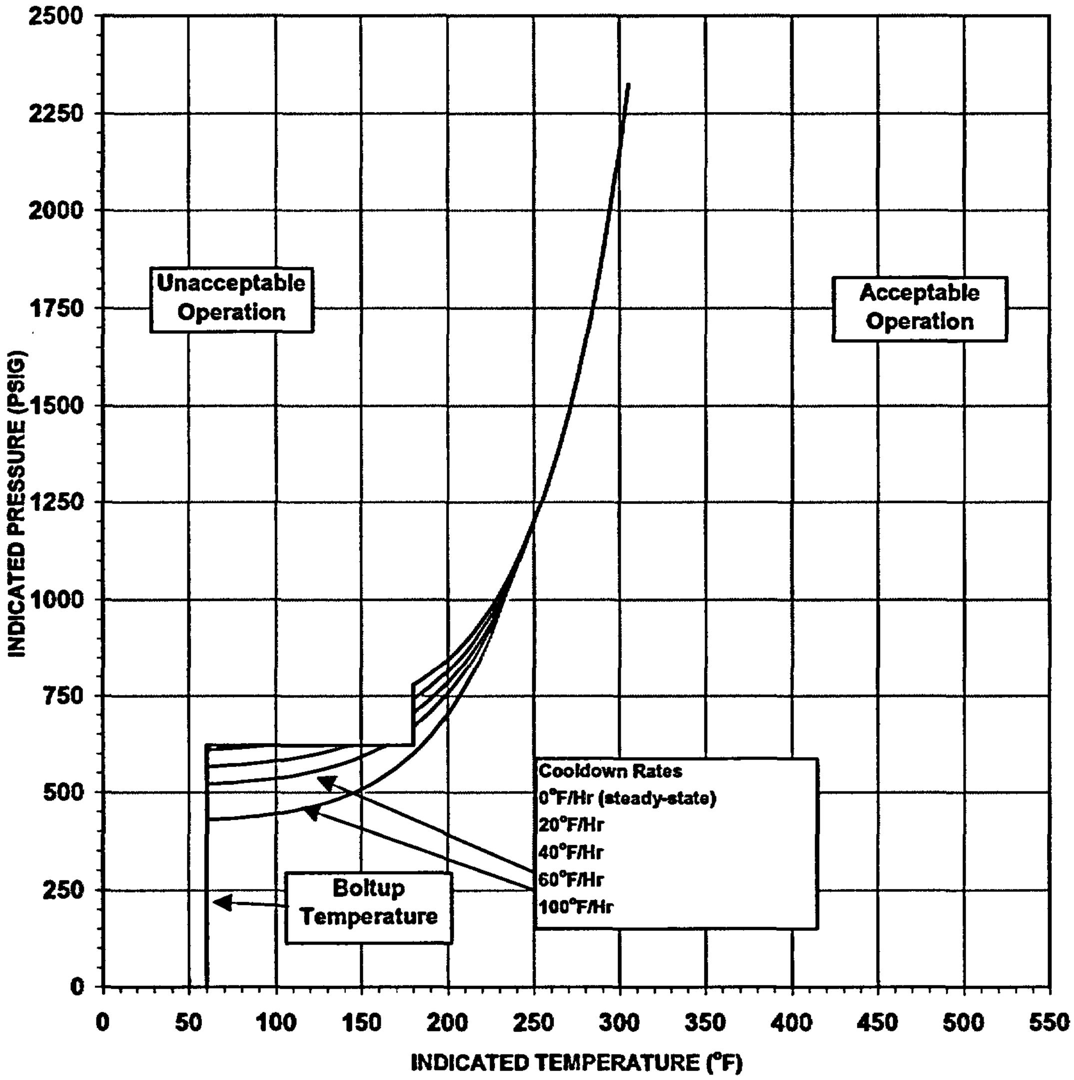

Figure 4.2-2
Reactor Coolant System Cooldown
Limitations Applicable for the First 22 EFPY (TS 3.4.9.1)

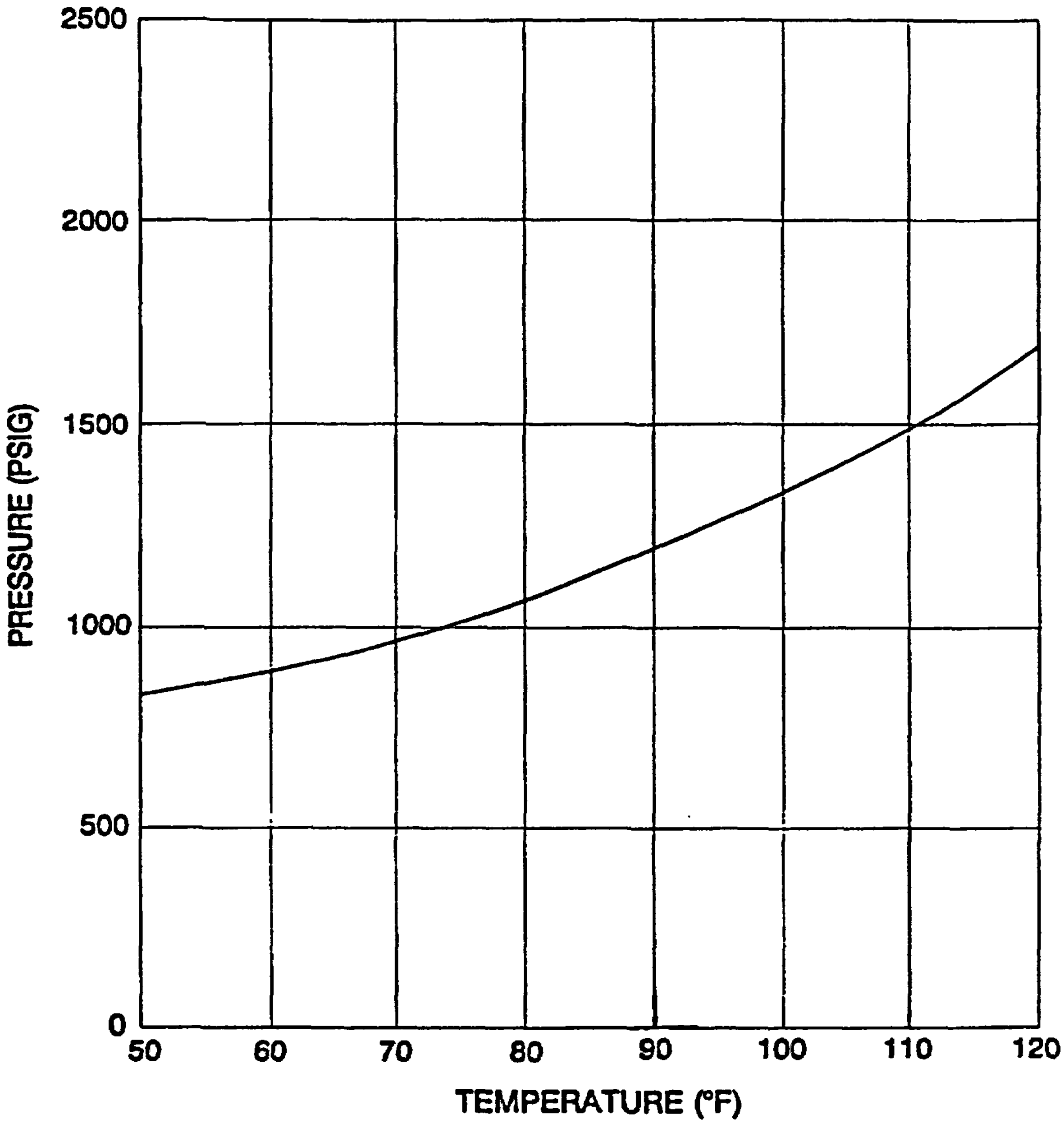

Figure 4.2-3
Isolated Loop Pressure – Temperature Limit Curve (TS 3.4.9.1)

Table 4.2-1
Heatup Curve Data Points for 22 EFPY (TS 3.4.9.1)

100°F/HR HEATUP				100°F/HR CRITICALITY				LEAK TEST LIMIT	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	200	677	289	0	289	716	271	2000
60	564	205	696	289	564	289	739	289	2485
65	564	210	716	289	565	289	764		
70	564	215	739	289	565	289	792		
75	564	220	764	289	566	289	822		
80	564	225	792	289	566	289	856		
85	564	230	822	289	568	289	894		
90	564	235	856	289	569	289	936		
95	564	240	894	289	571	290	982		
100	564	245	936	289	572	295	1033		
105	564	250	982	289	575	300	1089		
110	564	255	1033	289	577	305	1151		
115	564	260	1089	289	580	310	1219		
120	564	265	1151	289	583	315	1294		
125	564	270	1219	289	586	320	1378		
130	565	275	1294	289	591	325	1470		
135	566	280	1378	289	593	330	1571		
140	568	285	1470	289	600	335	1682		
145	571	290	1571	289	601	340	1806		
150	575	295	1682	289	611	345	1941		
155	580	300	1806	289	612	350	2091		
160	586	305	1941	289	621	355	2222		
165	593	310	2091	289	621	360	2361		
170	601	315	2222	289	621				
175	611	320	2361	289	621				
180	621			289	621				
180	621			289	633				
180	621			289	646				
185	633			289	661				
190	646			289	677				
195	661			289	696				

Table 4.2-2 (Page 1 of 2)
Cooldown Curve Data Points for 22 EFPY (TS 3.4.9.1)

STEADY STATE		20°F/HR.		40°F/HR.		60°F/HR.		100°F/HR.	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	609	60	566	60	521	60	430
65	621	65	611	65	567	65	523	65	431
70	621	70	612	70	568	70	524	70	432
75	621	75	614	75	570	75	525	75	433
80	621	80	615	80	572	80	527	80	435
85	621	85	617	85	574	85	529	85	437
90	621	90	619	90	576	90	531	90	439
95	621	95	621	95	578	95	534	95	442
100	621	100	621	100	581	100	536	100	445
105	621	105	621	105	584	105	540	105	448
110	621	110	621	110	587	110	543	110	452
115	621	115	621	115	591	115	547	115	457
120	621	120	621	120	596	120	552	120	462
125	621	125	621	125	600	125	557	125	468
130	621	130	621	130	606	130	562	130	474
135	621	135	621	135	612	135	569	135	481
140	621	140	621	140	618	140	576	140	490
145	621	145	621	145	621	145	584	145	499
150	621	150	621	150	621	150	592	150	509
155	621	155	621	155	621	155	602	155	520
160	621	160	621	160	621	160	613	160	533
165	621	165	621	165	621	165	621	165	547
170	621	170	621	170	621	170	621	170	563
175	621	175	621	175	621	175	621	175	581
180	621	180	621	180	621	180	621	180	600
180	621	180	621	180	621	180	621	185	622
180	778	180	742	180	706	180	670	190	647
185	792	185	757	185	723	185	689	195	674
190	808	190	775	190	742	190	709	200	704
195	826	195	794	195	762	195	732	205	737

Table 4.2-2 (Page 2 of 2)
Cooldown Curve Data Points for 22 EFPY (TS 3.4.9.1)

STEADY STATE		20°F/HR.		40°F/HR.		60°F/HR.		100°F/HR.	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
200	846	200	815	200	785	200	757	210	774
205	868	205	839	205	811	205	785	215	815
210	892	210	865	210	839	210	815	220	861
215	918	215	894	215	871	215	850	225	911
220	947	220	925	220	905	220	888	230	967
225	980	225	961	225	944	225	930	235	1030
230	1016	230	1000	230	986	230	976	240	1099
235	1055	235	1043	235	1033	235	1028	245	1147
240	1099	240	1090	240	1086	240	1085	250	1201
245	1147	245	1143	245	1143	245	1147	255	1260
250	1201	250	1201	250	1201	250	1201	260	1325
255	1260	255	1260	255	1260	255	1260	265	1397
260	1325	260	1325	260	1325	260	1325	270	1477
265	1397	265	1397	265	1397	265	1397	275	1565
270	1477	270	1477	270	1477	270	1477	280	1662
275	1565	275	1565	275	1565	275	1565	285	1770
280	1662	280	1662	280	1662	280	1662	290	1888
285	1770	285	1770	285	1770	285	1770	295	2020
290	1888	290	1888	290	1888	290	1888	300	2165
295	2020	295	2020	295	2020	295	2020	305	2325
300	2165	300	2165	300	2165	300	2165		
305	2325	305	2325	305	2325	305	2325		

PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-3

Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

FUNCTION	SETPOINT
OPPS Enable Temperature	343°F
PORV Setpoint	≤ 403 psig

Table 4.2-4

Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f [a]	FF [b]	ΔRT_{NDT} [c]	FF $*\Delta RT_{NDT}$	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 4.2-4a

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

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 4.2-4b

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 4.2-5

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 4.2-6

Summary of Adjusted Reference Temperature (ARTs) for 22 EFPY

MATERIAL DESCRIPTION	22 EFPY	
	1/4T ART(°F)[a]	3/4T ART(°F)[a]
Intermediate Shell Plate B6607-1	193	166
Intermediate Shell Plate B6607-2	223	196
Lower Shell Plate B7203-2	168	141
Lower Shell Plate B6903-1	230	191
- Using S/C Data[b]	233	193
Intermediate Shell Longitudinal Weld 19-714A/B	145	102
- Using S/C Data[b]	143	100
Intermediate to Lower Shell Circ. Weld 11-714	152	119
- Using S/C Data [c]	86	63
Lower Shell Longitudinal Weld 20-714A/B	159	111
- Using S/C Data[d]	168	117

Notes:

(a) $ART = I + \Delta RT_{NDT} + M$.

(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 4.2-7

Calculation of Adjusted Reference Temperatures (ARTs) for 22 EFPY

PARAMETER	VALUES	
Operating Time	22 EFPY	
Material	Plate B6903-1	Plate B6607-2
Location	Lower Shell Plate 1/4T ART(°F)	Intermediate Shell Plate 3/4T ART(°F)
Chemistry Factor, CF (°F)	149.2	100.5
Fluence (f), n/cm^2 (E>1.0 Mev)[a]	1.70×10^{19}	6.62×10^{18}
Fluence Factor, FF	1.15	.884
ΔRT_{NDT} = CF x FF(°F)[c]	171.6 [c]	88.84
Initial RT_{NDT}, I(°F)[a]	27	73
Margin, M(°F)	34 [c]	34
ART = I+(CF*FF)+M, °F[b] per RG 1.99, Revision 2	233	196

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 4.2-8
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 4.2-9

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

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

(c) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°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 4.2-10

RT_{PTS} Calculation for Beltline Region Materials at Life extension (45 EFPY)

Material	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Plate B6607-1	5.85	1.43	100.5	143.7	34	43	221
Intermediate Shell Plate B6607-2	5.85	1.43	100.5	143.7	34	73	251
Lower Shell Plate B7203-2	5.85	1.43	98.7	141.1	34	20	195
Lower Shell Plate B6903-1	5.85	1.43	147.2	210.5	34	27	272
→ Using S/C Data[e]	5.85	1.43	149.2	213.4	34	27	274
Inter. Shell Long. Weld 19-714A/B	1.13	1.03	191.7	197.5	65.5	-56	207
→ Using S/C Data[e]	1.13	1.03	188.8	194.5	65.5	-56	204
Lower Shell Long. Weld 20-714A/B	1.13	1.03	210.5	216.8	65.5	-56	226
→ Using S/C Data[f]	1.13	1.03	223.9	230.6	65.5	-56	240
Circumferential Weld 11-714	5.82	1.43	124.3	177.7	65.5	-56	187
→ Using S/C Data[d]	5.82	1.43	84.8	121.3	44	-56	109

Notes:
(a) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.
(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F).
(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.

SECTION 4.2 PRESSURE AND TEMPERATURE LIMITS REPORT

BVPS-2 Technical Specification to PTLR Cross-Reference			
Technical Specification	PTLR		
	Section	Figure	Table
3.1.2.2	N/A	N/A	4.2-3
3.1.2.4	N/A	N/A	4.2-3
3.4.1.3	N/A	N/A	4.2-3
3.4.9.1	4.2.1.1	4.2-1 4.2-2 4.2-3 4.2-4 4.2-5 4.2-6	N/A
3.4.9.3	4.2.1.2 4.2.1.3	4.2-8	4.2-3
3.5.2	N/A	N/A	4.2-3
3.5.3	N/A	N/A	4.2-3

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

This Pressure and Temperature Limits Report (PTLR) for Unit 2 has been prepared in accordance with the requirements of Technical Specification 6.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) addressed in, or make reference to, this report are listed below:

TS 3.1.2.2 Reactivity Control Systems – Flow Paths – Operating,

TS 3.1.2.4 Reactivity Control Systems – Charging Pumps – Operating,

TS 3.4.1.3 Reactor Coolant System – Shutdown,

TS 3.4.9.1 Reactor Coolant System - Pressure/Temperature Limits,

TS 3.4.9.3 Overpressure Protection Systems,

TS 3.5.2 ECCS Subsystems – $T_{avg} \geq 350°F$, and

TS 3.5.3 ECCS Subsystems – $T_{avg} < 350°F$.

4.2.1 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".

4.2.1.1 RCS Pressure and Temperature (P/T) Limits (TS 3.4.9.1)

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

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 4.2-1 and Table 4.2-1. The RCS P/T limits for cooldown are shown in Figure 4.2-2 through 4.2-6 and Table 4.2-2. These limits are defined in Reference 2. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 4.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 4.2-1 through and 4.2-6 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 heatup and cooldown curves are shown for 14 effective full power years (EFPY), although the capsule data provided in Tables 4.2-5 through 4.2-10 state 15 EFPY. The EFPY for the heatup and cooldown curves were generated for 15 EFPY. However, the heatup and cooldown curves were revised by license amendment 243 (1.4 % power uprate). The change from 15 to 14 EFPY was done to impose a conservative administrative limit due to the increased neutron fluence associated with the 1.4% increase in reactor power. The capsule data tables however, continue to reflect 15 EFPYs to retain actual capsule analysis results.

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 4.2-7 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.

4.2.1.2 Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

The power operated relief valves (PORVs) shall each have nominal maximum lift setting is in accordance with Figure 4.2-8. The OPPS enable temperature is in accordance with Table 4.2-3. The PORV lift setting provided is for the case with reactor coolant pump (RCP) restrictions. These restrictions are shown in Table 4.2-4, which is taken from Reference 10. Due to the setpoint limitations as a result of the reactor vessel flange requirements, there is no operational benefit achieved by restricting the number of RCPs running to less than two below an indicated RCS temperature of 190°F. Therefore, the PORV setpoints shown in Table 4.2-4 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 4.2.1. The PORV lift setting shown in Figure 4.2-8 accounts for appropriate instrument error.

4.2.1.3 OPPS Enable Temperature (TS 3.4.9.3)

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 236.5°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 350°F.

As the calculated enable temperature is higher and, therefore, more conservative than the arming temperature, the OPPS enable temperature, as shown in Table 4.2-3, 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 4.2-3, 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 TS 3.4.9.3. This is accomplished by placing two keylock 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.

4.2.1.4 Reactor Vessel Boltup Temperature (TS 3.4.9.1)

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.

4.2.2 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 4.2-1 through 4.2-6, and Tables 4.2-1 and 4.2-2. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching 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 12 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.

4.2.3 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 4.2-5, taken from Reference 2, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2-6, taken from Reference 2, provides the reactor vessel beltline material property table.

Table 4.2-7, taken from Reference 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 15 EFPY.

Table 4.2-8, taken from Reference 2, shows the calculation of ARTs for 15 EFPY.

Table 4.2-9 shows the Reactor Vessel Toughness Data (Unirradiated).

Table 4.2-10, taken from Reference 6, provides RT_{PTS} values for 32 EFPY.

4.2.4 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-15139, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal Operation at 15 EFPY Using Code Case N-626," T. J. Laubham, January 1999.
3. WCAP-14484, Revision 0, "Analysis of Capsule V from the Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," P. A. Grendys, S. L. Anderson, J. F. Williams, February 1996.
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-14485, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," P. A. Grendys, March 1986.
6. WCAP-14784, Revision 2, "Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 2," T. J. Laubham, February 1996.
7. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.
8. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)
9. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
10. Westinghouse Report NPD-OPES(99)-055, "Low Temperature Overpressure Protection System Setpoint Review for Beaver Valley Unit 2 15 EFPY Heatup and Cooldown Curves," March 1999.
11. Westinghouse Letter FENOC-01-261, COMS Arming Temperature", E. A. Dzenis, September 10, 2001.
12. FirstEnergy Nuclear Operating Company letter L-01-157, "Supplement to License Amendment Requests Nos. 295 and 167," dated December 21, 2001.

LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

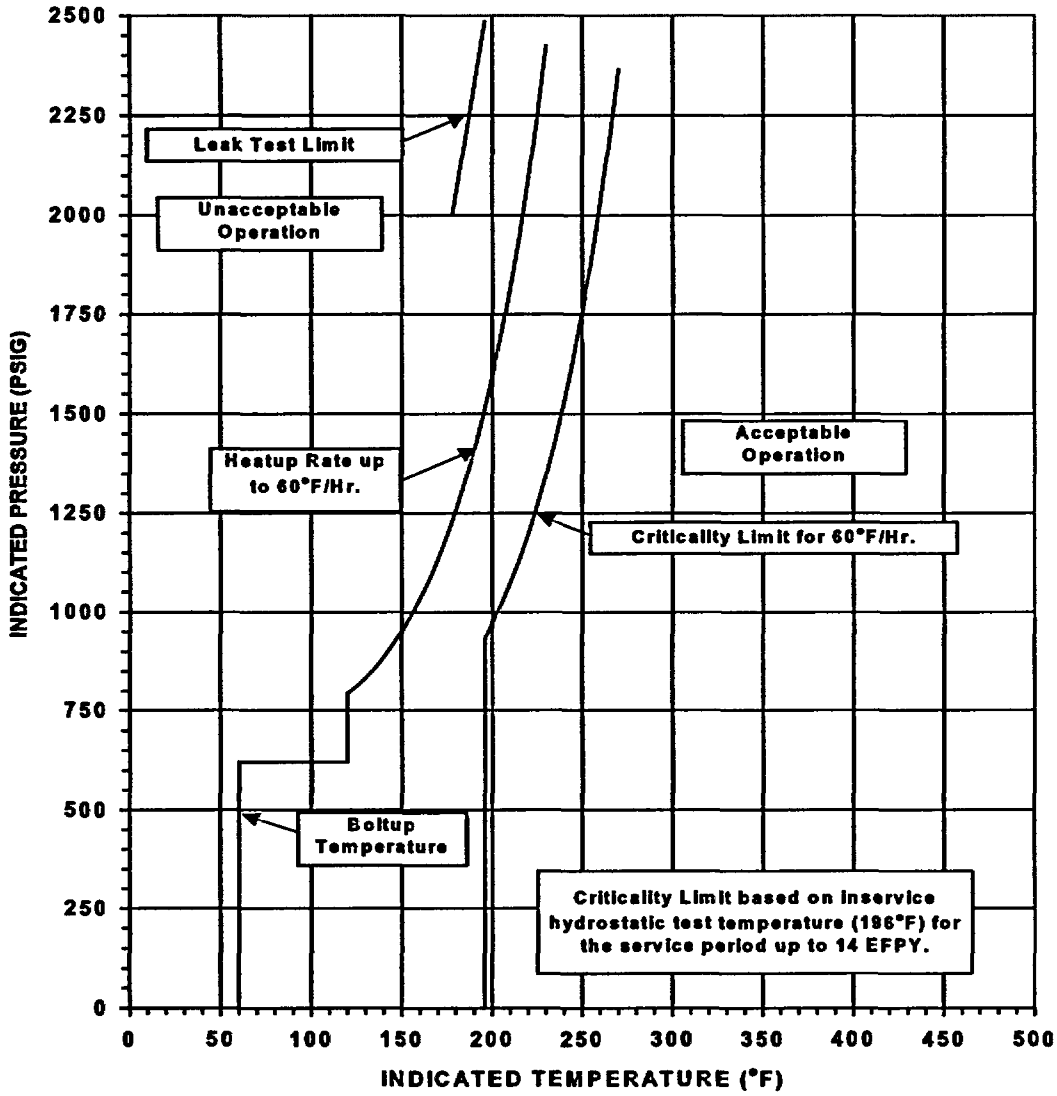

Figure 4.2-1
Reactor Coolant System Heatup
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

PRESSURE AND TEMPERATURE LIMITS REPORT

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 0°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

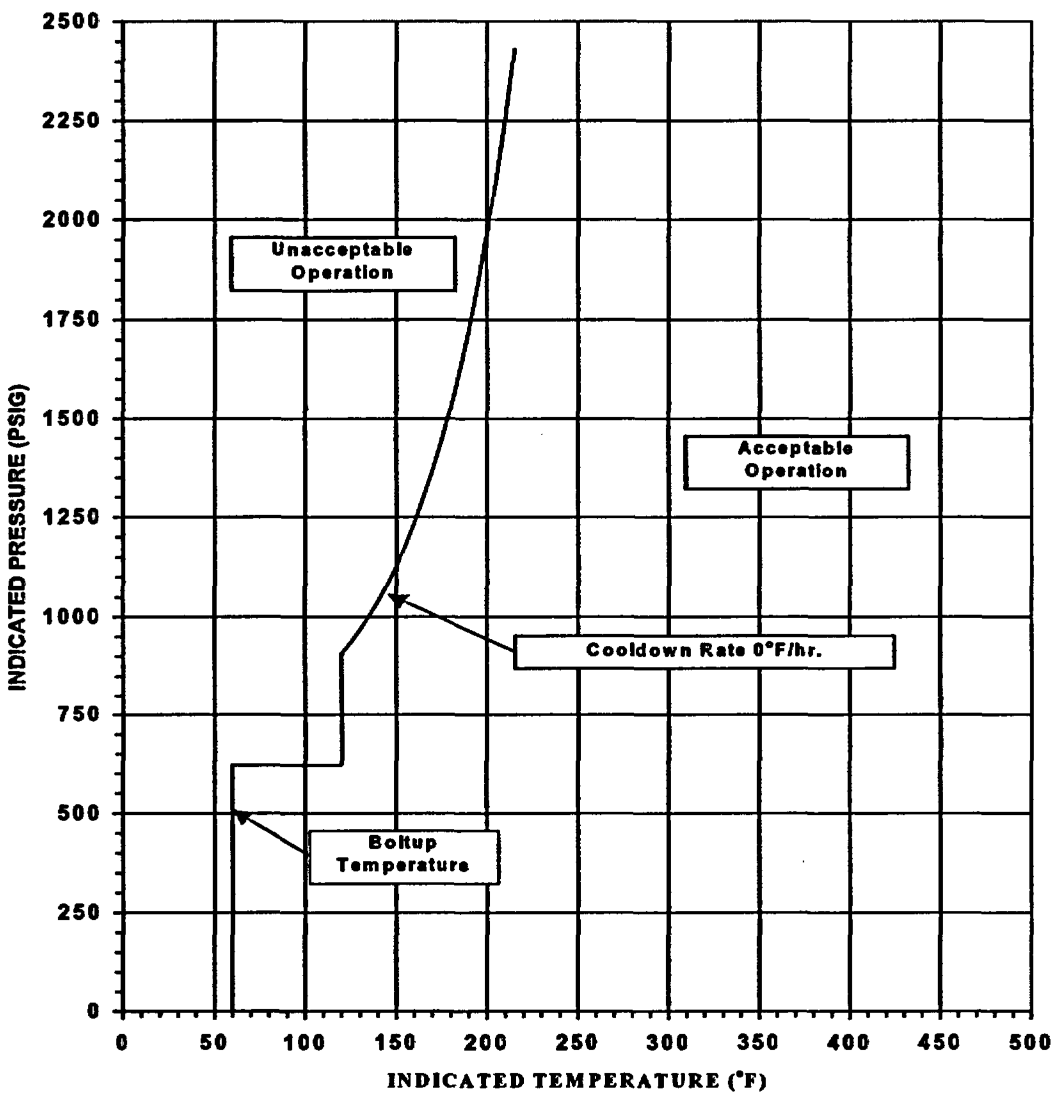

Figure 4.2-2
Reactor Coolant System Cooldown (up to 0°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

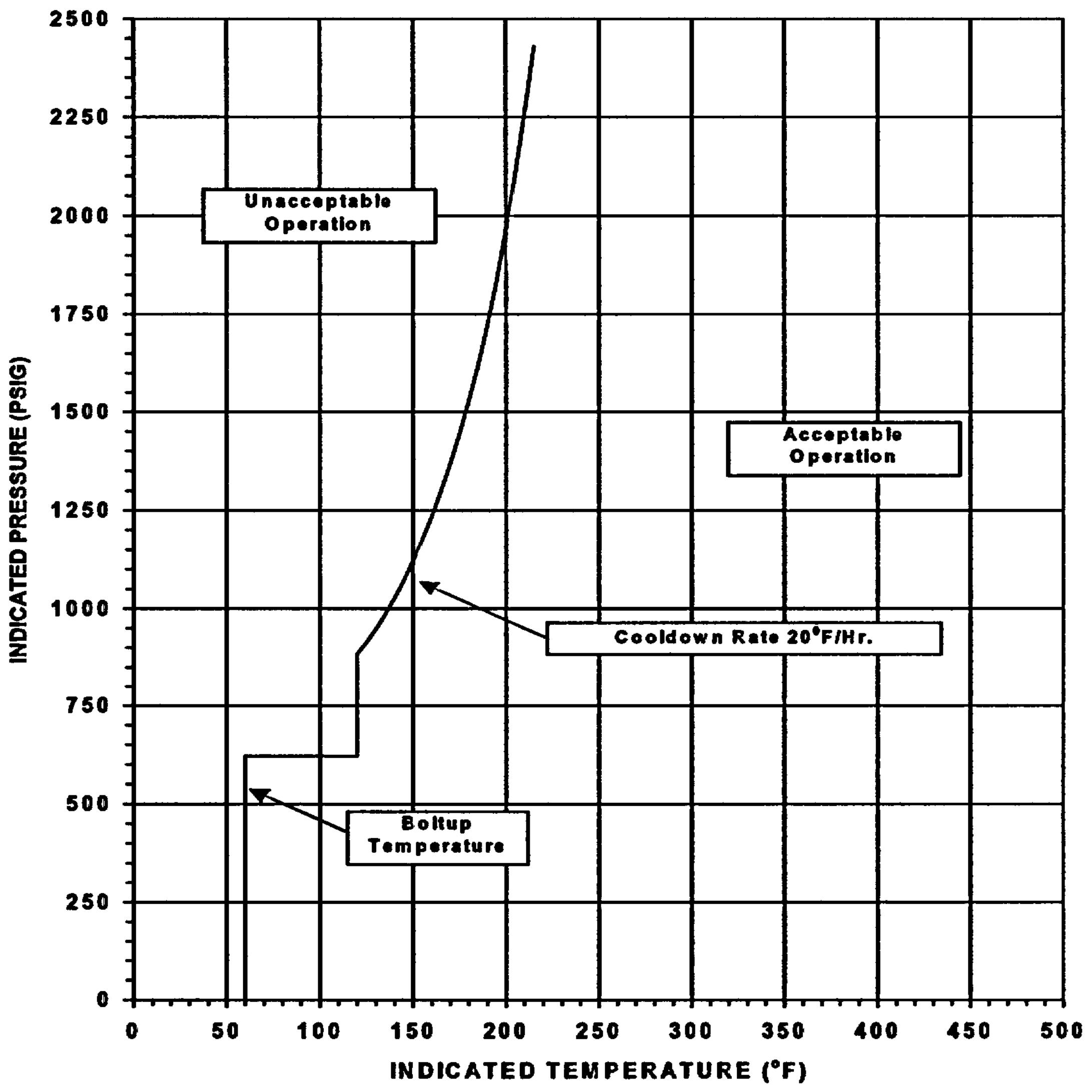

Figure 4.2-3
Reactor Coolant System Cooldown (up to 20°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

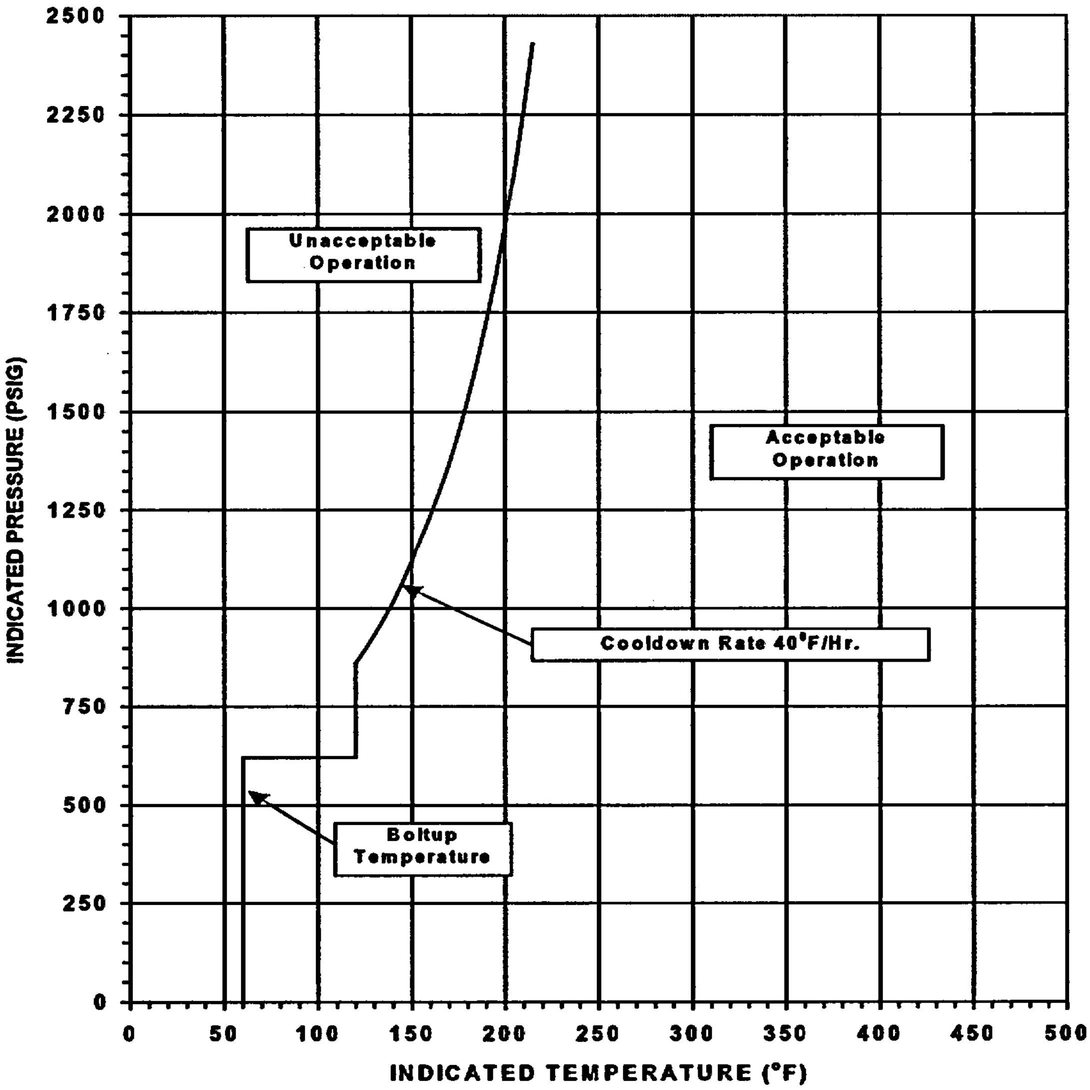

Figure 4.2-4
Reactor Coolant System Cooldown (up to 40°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

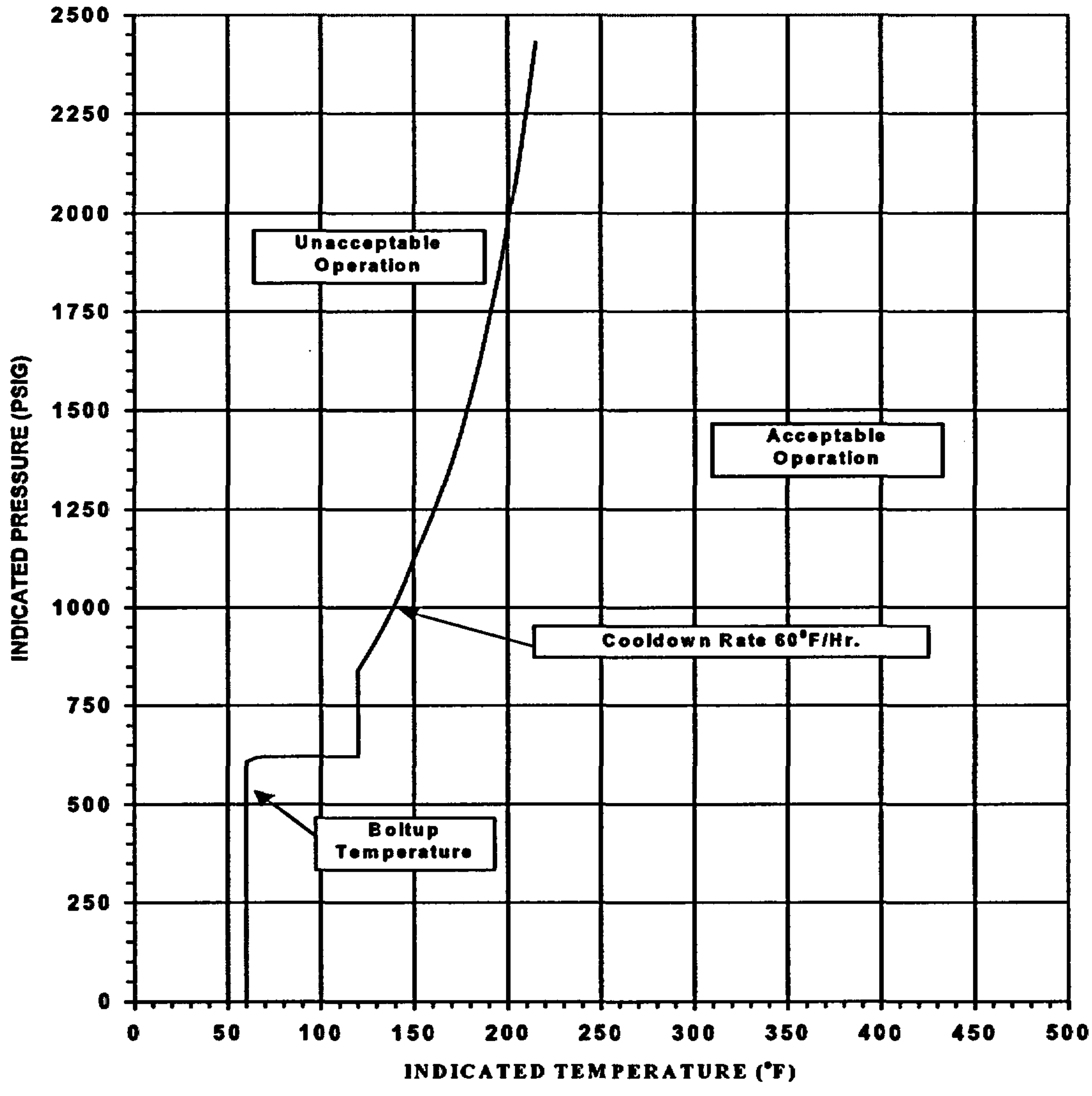

Figure 4.2-5
Reactor Coolant System Cooldown (up to 60°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

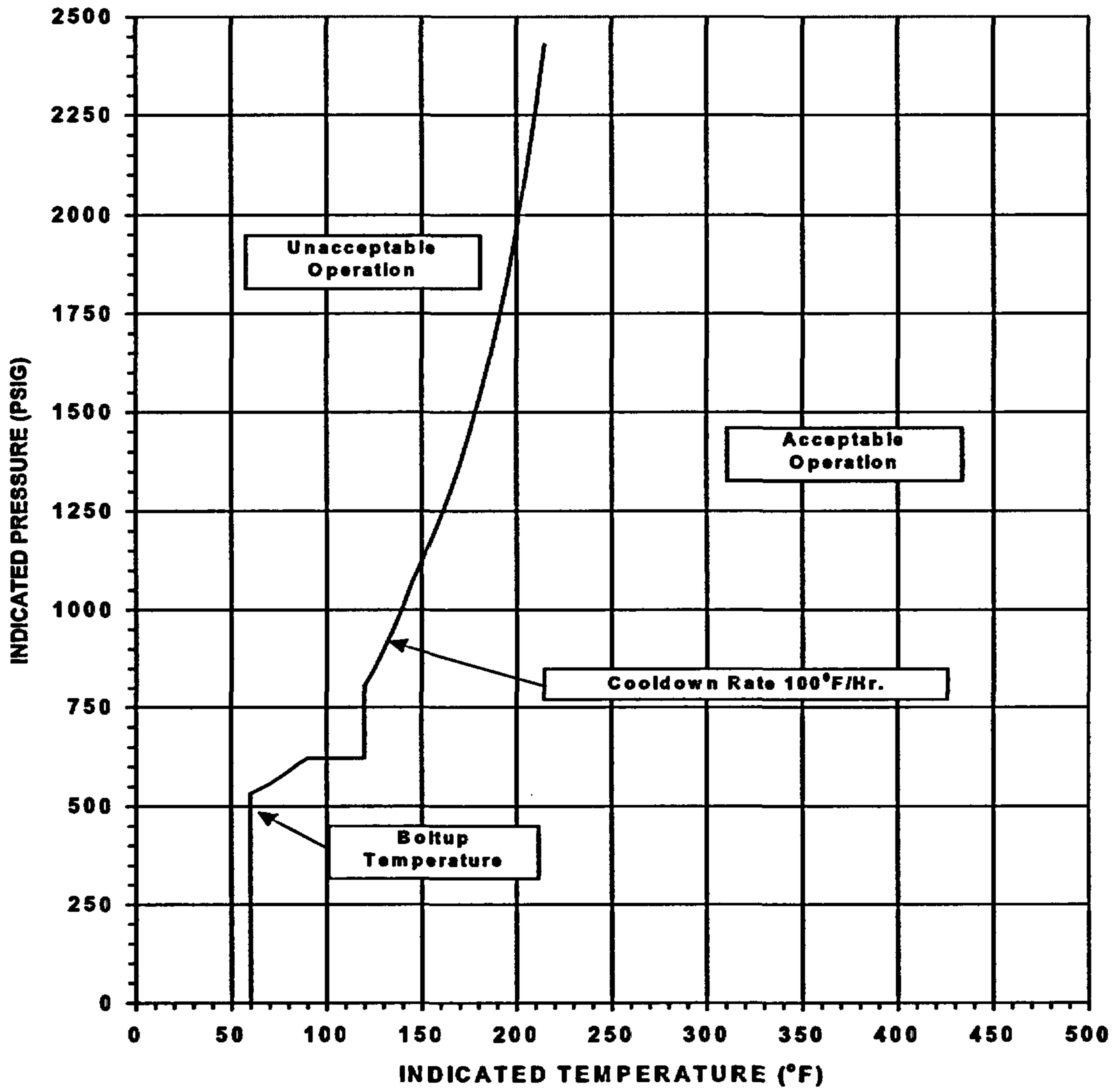

Figure 4.2-6
Reactor Coolant System Cooldown (up to 100°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

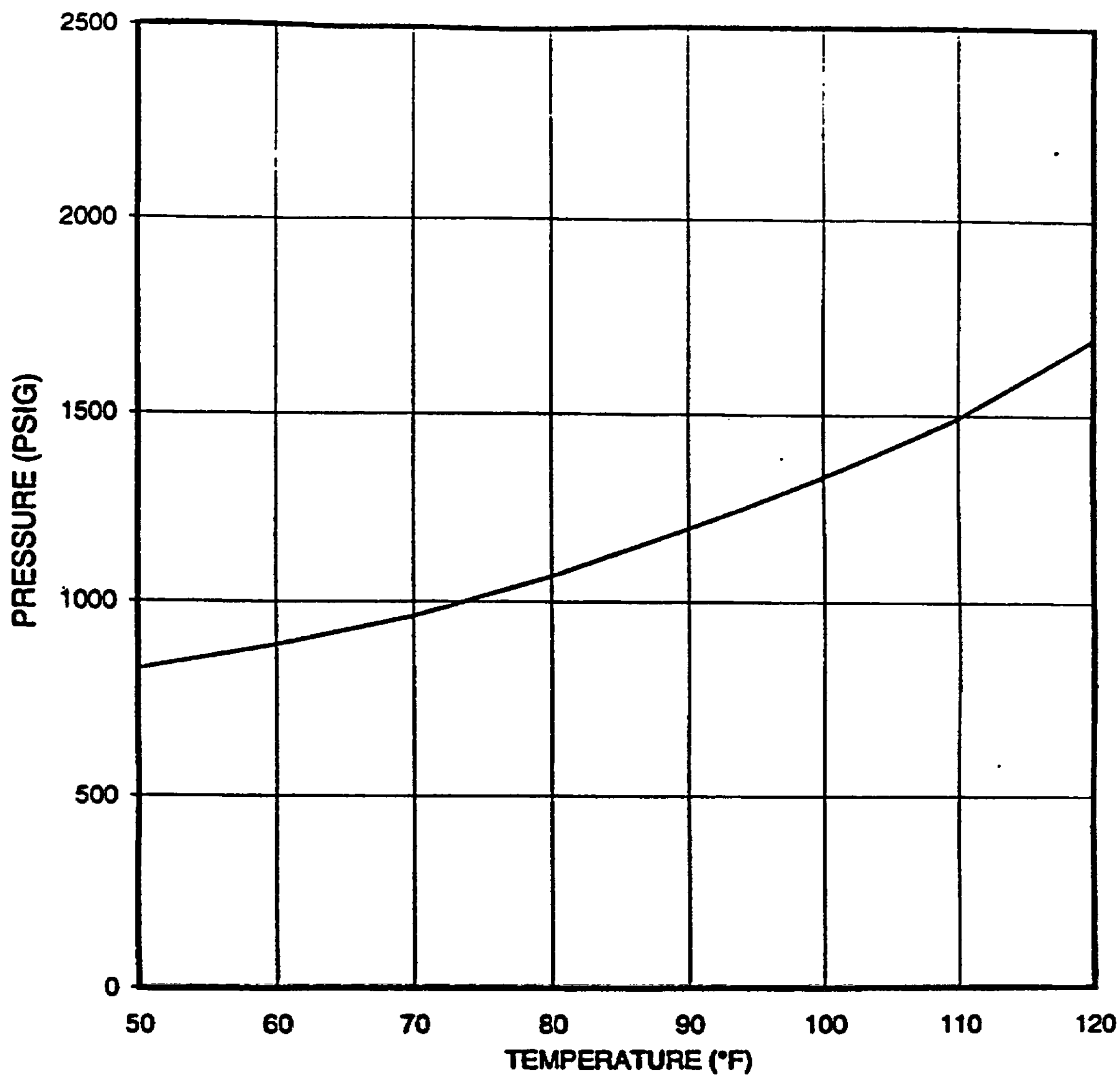

Figure 4.2-7
Isolated Loop Pressure – Temperature Limit Curve (TS 3.4.9.1)

See Table 4.2-4 for RCP restrictions.

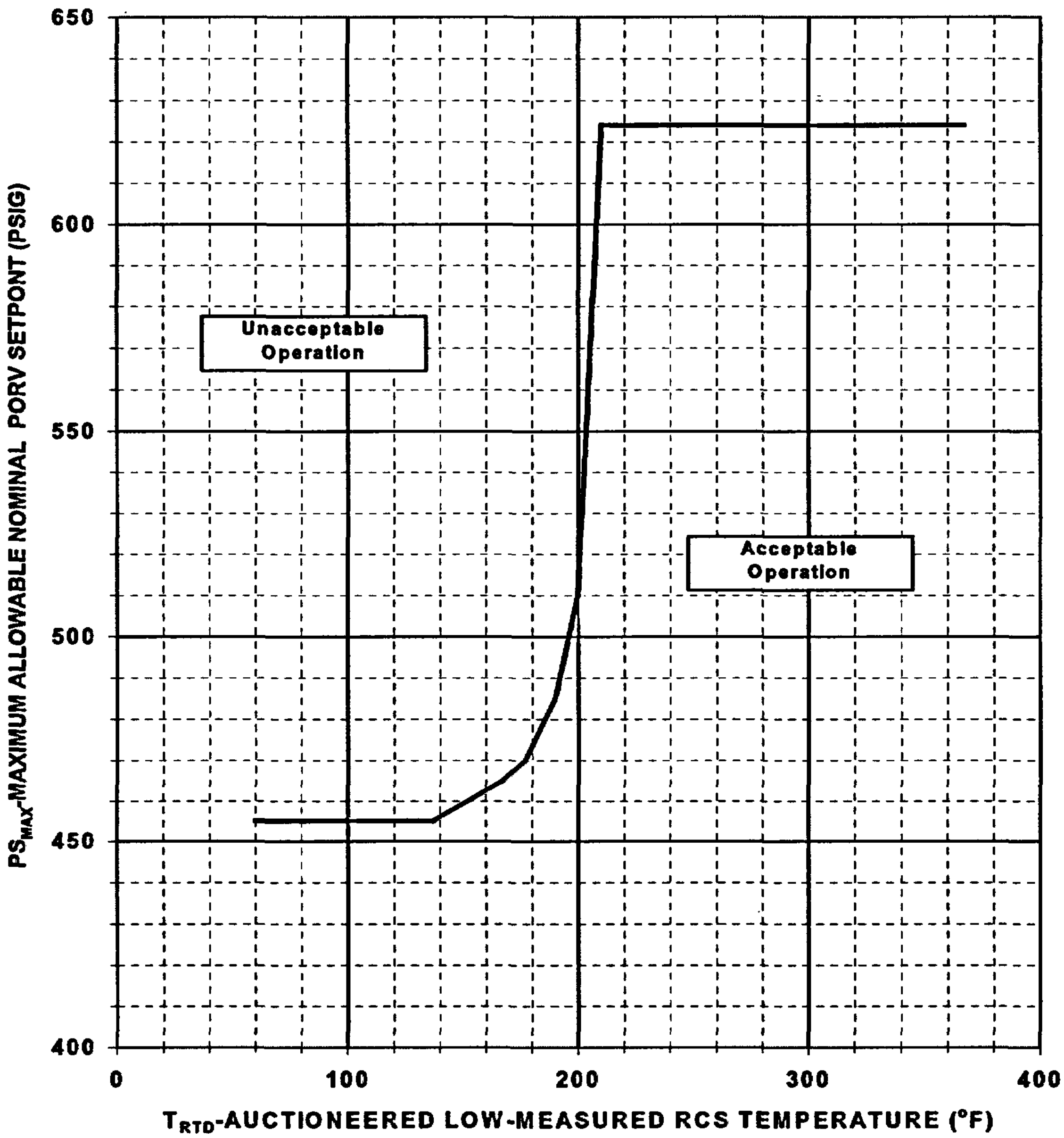

Figure 4.2-8
Maximum Allowable Nominal PORV Setpoint for the Overpressure Protection System (TS 3.4.9.3)

Table 4.2-1
Heatup Curve Data Points for 14 EFPY (TS 3.4.9.1)

60°F/HR HEATUP		60°F/HR CRITICALITY		LEAK TEST LIMIT	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60.00	621.00	196.00	0.00	178.00	2000.00
65.00	621.00	196.00	668.58	196.00	2485.00
85.00	621.00	196.00	693.10		
90.00	621.00	196.00	687.74		
95.00	621.00	196.00	686.96		
100.00	621.00	196.00	689.93		
105.00	621.00	196.00	696.41		
110.00	621.00	196.00	705.98		
115.00	621.00	196.00	718.57		
120.00	621.00	196.00	734.02		
120.00	794.02	196.00	752.40		
125.00	812.40	196.00	773.69		
130.00	833.69	196.00	798.06		
135.00	858.06	196.00	825.62		
140.00	885.62	196.00	856.64		
145.00	916.64	196.00	891.32		
150.00	951.32	196.00	930.00		
155.00	990.00	200.00	973.00		
160.00	1033.00	205.00	1020.74		
165.00	1080.74	210.00	1073.65		
170.00	1133.65	215.00	1132.24		
175.00	1192.24	220.00	1197.06		
180.00	1257.06	225.00	1268.73		
185.00	1328.73	230.00	1347.95		
190.00	1407.95	235.00	1435.48		
195.00	1495.48	240.00	1532.16		
200.00	1592.16	245.00	1638.92		
205.00	1698.92	250.00	1756.81		
210.00	1816.81	255.00	1886.95		
215.00	1946.95	260.00	2030.62		
220.00	2090.62	265.00	2189.19		
225.00	2249.19	270.00	2364.21		
230.00	2424.21				

Table 4.2-2
Cooldown Curve Data Points for 14 EFPY (TS 3.4.9.1)

	0^{O}F/HR.	20^{O}F/HR.	40^{O}F/HR.	60^{O}F/HR.	100^{O}F/HR.
Temp. (^{o}F)	Press. (psig)	Press. (psig)	Press. (psig)	Press. (psig)	Press. (psig)
60.00	621.00	621.00	621.00	607.90	531.77
65.00	621.00	621.00	621.00	618.15	543.64
70.00	621.00	621.00	621.00	621.00	556.89
75.00	621.00	621.00	621.00	621.00	571.67
80.00	621.00	621.00	621.00	621.00	588.13
85.00	621.00	621.00	621.00	621.00	606.46
90.00	621.00	621.00	621.00	621.00	621.00
95.00	621.00	621.00	621.00	621.00	621.00
100.00	621.00	622.00	621.00	621.00	621.00
105.00	621.00	621.00	621.00	621.00	621.00
110.00	621.00	621.00	621.00	621.00	621.00
115.00	621.00	621.00	621.00	621.00	621.00
120.00	621.00	621.00	621.00	621.00	621.00
120.00	907.20	883.94	862.00	841.62	806.75
125.00	935.36	914.39	894.99	877.44	849.30
130.00	966.47	948.05	931.50	917.13	896.50
135.00	1000.87	985.31	971.92	961.10	948.86
140.00	1038.88	1026.49	1016.66	1009.79	1006.92
145.00	1080.88	1072.06	1066.18	1063.72	1071.29
150.00	1127.31	1122.43	1120.97	1123.42	1123.42
155.00	1178.62	1178.62	1178.62	1178.62	1178.62
160.00	1235.32	1235.32	1235.32	1235.32	1235.32
165.00	1297.99	1297.99	1297.99	1297.99	1297.99
170.00	1367.25	1367.25	1367.25	1367.25	1367.25
175.00	1443.79	1443.79	1443.79	1443.79	1443.79
180.00	1528.38	1528.38	1528.38	1528.38	1528.38
185.00	1621.87	1621.87	1621.87	1621.87	1621.87
190.00	1725.19	1725.19	1725.19	1725.19	1725.19
195.00	1839.38	1839.38	1839.38	1839.38	1839.38
200.00	1965.58	1965.58	1965.58	1965.58	1965.58
205.00	2105.05	2105.05	2105.05	2105.05	2105.05
210.00	2259.18	2259.18	2259.18	2259.18	2259.18
215.00	2429.53	2429.53	2429.53	2429.53	2429.53

Table 4.2-3

Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

FUNCTION	SETPOINT
OPPS Enable Temperature	350°F
PORV Setpoint	Figure 4.2-8

Table 4.2-4

Reactor Coolant Pump Restrictions

T_{RCS}	Running RCPs
< 190°F	0 – 2
≥ 190°F	3

Table 4.2-5
Calculation of Chemistry Factors Using Surveillance Capsule Data[a][e]

Material	Capsule	Capsule f [b]	FF[c]	ΔRT_{NDT}[d]	FF $*\Delta RT_{NDT}$	FF^2
Intermediate Shell Plate B9004-2 (Longitudinal)	U	0.601	0.857	24.26	20.8	0.735
	V	2.64	1.26	55.93	70.5	1.59
Intermediate Shell Plate B9004-2 (Transverse)	U	0.601	0.857	17.56	15.1	0.735
	V	2.64	1.26	46.27	58.3	1.59
				SUM:	164.7	4.66
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = 35.3$					
Weld Metal	U	0.601	0.857	3.64	3.1	0.735
	V	2.64	1.26	25.47	32.1	1.59
				SUM:	35.2	2.32
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = 15.2$					

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 V analysis (Reference 4).
(c) FF = fluence factor = $f^{(0.28 - 0.1 * \log f)}$.
(d) ΔRT_{NDT} values obtained from CVGRAPH Version 4.0.
(e) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-6
Reactor Vessel Beltline Material Properties [c]

Material	Method Used To Calculate CF[a]	Average Cu wt %	Average Ni wt %	Chemistry Factor (°F)	Initial RT_{NDT}[b] (°F)
Closure Head Flange	N/A	N/A	0.74	N/A	-10
Vessel Flange	N/A	N/A	0.73	N/A	0
Intermediate Shell Plate B9004-1	Position 1.1	0.065	0.55	44.0	60
Intermediate Shell Plate B9004-2	Position 1.1	0.06	0.57	37.0	40
	Position 2.1	N/A	N/A	35.3	40
Lower Shell Plate B9005-1	Position 1.1	0.08	0.58	51.0	28
Lower Shell Plate B9005-2	Position 1.1	0.07	0.57	44.0	33
Weld Metal	Position 1.1	0.05	0.07	34.1	-30
(Longitudinal & Circumferential Seams)	Position 2.1	N/A	N/A	15.2	-30

Notes:

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

(b) Initial RT_{NDT} values of the base metal and weld metal materials are measured values.

(c) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-7

Summary of Adjusted Reference Temperature (ARTs) for 15 EFPY [b]

MATERIAL DESCRIPTION	Method Used To Calculate the CF[a]	15 EFPY ART	
		1/4T ART(°F)	3/4T ART(°F)
Intermediate Shell Plate B9004-1	Position 1.1	140	128
Intermediate Shell Plate B9004-2	Position 1.1	112	97
	Position 2.1	94	84
Lower Shell Plate B9005-1	Position 1.1	115	101
Lower Shell Plate B9005-2	Position 1.1	112	101
Longitudinal Welds (located at 45° azimuthal angle)	Position 1.1	19	3
	Position 2.1	-8	-15
Circumferential Weld	Position 1.1	41	23
	Position 2.1	1	-7

Notes:

(a) Regulatory Guide 1.99, Revision 2.

(b) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-8

Calculation of Adjusted Reference Temperatures (ARTs) for 15 EFPY [(b)]

PARAMETER	VALUES	
Operating Time	15 EFPY	
Material - Intermediate Shell Plate	B9004-1	B9004-1
Location	1/4T ART	3/4T ART
Chemistry Factor, CF (°F)	44.0	44.0
Fluence, (f), (10^{19} n/cm^2)[(a)]	1.13	0.439
Fluence Factor, FF	1.03	0.771
ΔRT_{NDT} = CF x FF(°F)	45.5	33.9
Initial RT_{NDT}, I(°F)	60	60
Margin, M(°F)	34	33.9
ART, per Regulatory Guide 1.99, Revision 2	140	128

Notes:

(a) Fluence (f), is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 1.81 at 15 EFPY.

(b) See Section 4.2.1.1 for a discussion of EFPY.

The Beaver Valley Unit 2 reactor vessel wall thickness is 7.875 inches at the beltline region.

Table 4.2-9
Reactor Vessel Toughness Data (Unirradiated)

COMPONENT	CODE NO.	MATERIAL SPEC. NO.	Cu %	Ni %	P %	T_{NDT} °F	50 FT/LB 35 MIL TEMP °F	RT_{NDT} °F	USE FT-LBS.
Closure Head Dome	B9008-1	A533B, CL. 1	.13	.54	.013	-20	50	-10	137
Closure Head Flange	B9002-1	A508, CL. 2	---	.74	.012	-10	<40	-10	136
Vessel Flange	B9001-1	A508, CL. 2	---	.73	.010	0	<10	0	132.5
Inlet Nozzle	B9011-1	A508, CL. 2	---	.88	.006	0	<10	0	104
Inlet Nozzle	B9011-2	A508, CL. 2	---	.88	.010	10	<10	10	115
Inlet Nozzle	B9011-3	A508, CL. 2	---	.84	.009	20	<40	20	122
Outlet Nozzle	B9012-1	A508, CL. 2	---	.71	.007	-10	< 0	-10	137
Outlet Nozzle	B9012-2	A508, CL. 2	---	.74	.006	-10	< 0	-10	121
Outlet Nozzle	B9012-3	A508, CL. 2	---	.68	.008	-10	< 0	-10	112
Nozzle Shell	B9003-1	A533B, CL. 1	.13	.61	.008	-10	110	50	91
Nozzle Shell	B9003-2	A533B, CL. 1	.12	.58	.009	0	120	60	79.5
Nozzle Shell	B9003-3	A533B, CL. 1	.13	.61	.008	-10	110	50	97.5
Inter. Shell	B9004-1	A533B, CL. 1	.07	.53	.010	0	120	60	83
Inter. Shell	B9004-2	A533B, CL. 1	.07	.59	.007	-10	100	40	75.5
Lower Shell	B9005-1	A533B, CL. 1	.08	.59	.009	-50	88	28	82
Lower Shell	B9005-2	A533B, CL. 1	.07	.58	.009	-40	93	33	77.5
Bottom Head Torus	B9010-1	A533B, CL. 1	.15	.49	.007	-30	56	-4	97
Bottom Head Dome	B9009-1	A533B, CL. 1	.14	.53	.007	-30	35	-25	116
Weld (Inter. & Lower Shell Long. Seams & Girth Seam)*			.08	.07	.008	-30	<30	-30	144.5
HAZ (Plate B9004-2)			---	---	----	-80	40	-20	76

* Same heat of wire and lot of flux used in all seams including surveillance weldment.

(1) 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.
(2) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-10

RT_{PTS} Calculation for Beltline Region Materials at EOL (32 EFPY) [d]

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	3.85	1.348	44.0	54.6	34	60	149
Intermediate Shell Plate B9004-2	RG 1.99, R2, P1.1	3.85	1.348	37.0	49.9	34	40	124
	RG 1.99, R2, P2.1	3.85	1.348	35.3	47.6	17	40	105
Lower Shell Plate B9005-1	RG 1.99, R2, P1.1	3.85	1.348	51.0	68.8	34	28	131
Lower Shell Plate B9005-2	RG 1.99, R2, P1.1	3.85	1.348	44.0	59.3	34	33	126
Circumferential Weld	RG 1.99, R2, P1.1	3.85	1.348	34.1	45.9	45.9	-30	62
	RG 1.99, R2, P2.1	3.85	1.348	15.2	20.5	20.5	-30	11
Longitudinal Weld	RG 1.99, R2, P1.1	1.21	1.053	34.1	35.9	35.9	-30	42
	RG 1.99, R2, P2.1	1.21	1.053	15.2	16.0	16.0	-30	2

Notes:

(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) See Section 4.2.1.1 for a discussion of EFPY.