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October 14, 2008
L-08-327

Beaver Valley Power Station
Technical Specification 5.6.4

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

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Unit No. 2 Pressure and Temperature Limits Report, Revision 3

FirstEnergy Nuclear Operating Company (FENOC) hereby submits a revision of the Pressure and Temperature Limits Report (PTLR) for Beaver Valley Power Station (BVPS) Unit No. 2 as required by Section 5.6.4 of the BVPS Technical Specifications. Revision 3 of the BVPS Unit No. 2 PTLR is enclosed.

The revision to the Unit No. 2 PTLR, effective October 9, 2008, was made to include the results of the Capsule X analysis documented in WCAP-16527-NP and WCAP-165127-NP, Supplement 1.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – FENOC Fleet Licensing, at 330-761-6071.

Sincerely,

Peter P. Sena III

Enclosure

Beaver Valley Power Station Unit No. 2, Pressure and Temperature Limits Report, Revision 3

A001
NRR

cc: Mr. S. J. Collins, NRC Region I Administrator
Mr. D. L. Werkheiser, NRC Senior Resident Inspector
Ms. N. S. Morgan, NRR Project Manager
Mr. D. J. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Beaver Valley Power Station Unit No. 2

Pressure and Temperature Limits Report

Revision 3

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 Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)

The power operated relief valves (PORVs) shall each have a nominal maximum lift setting that varies with RCS temperature and which does not exceed the limits in Figure 5.2-8 (Reference 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 OPPS Enable Temperature (LCO 3.4.12)

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

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.

5.2 Pressure and Temperature Limits Report

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 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 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.

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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.

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-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.

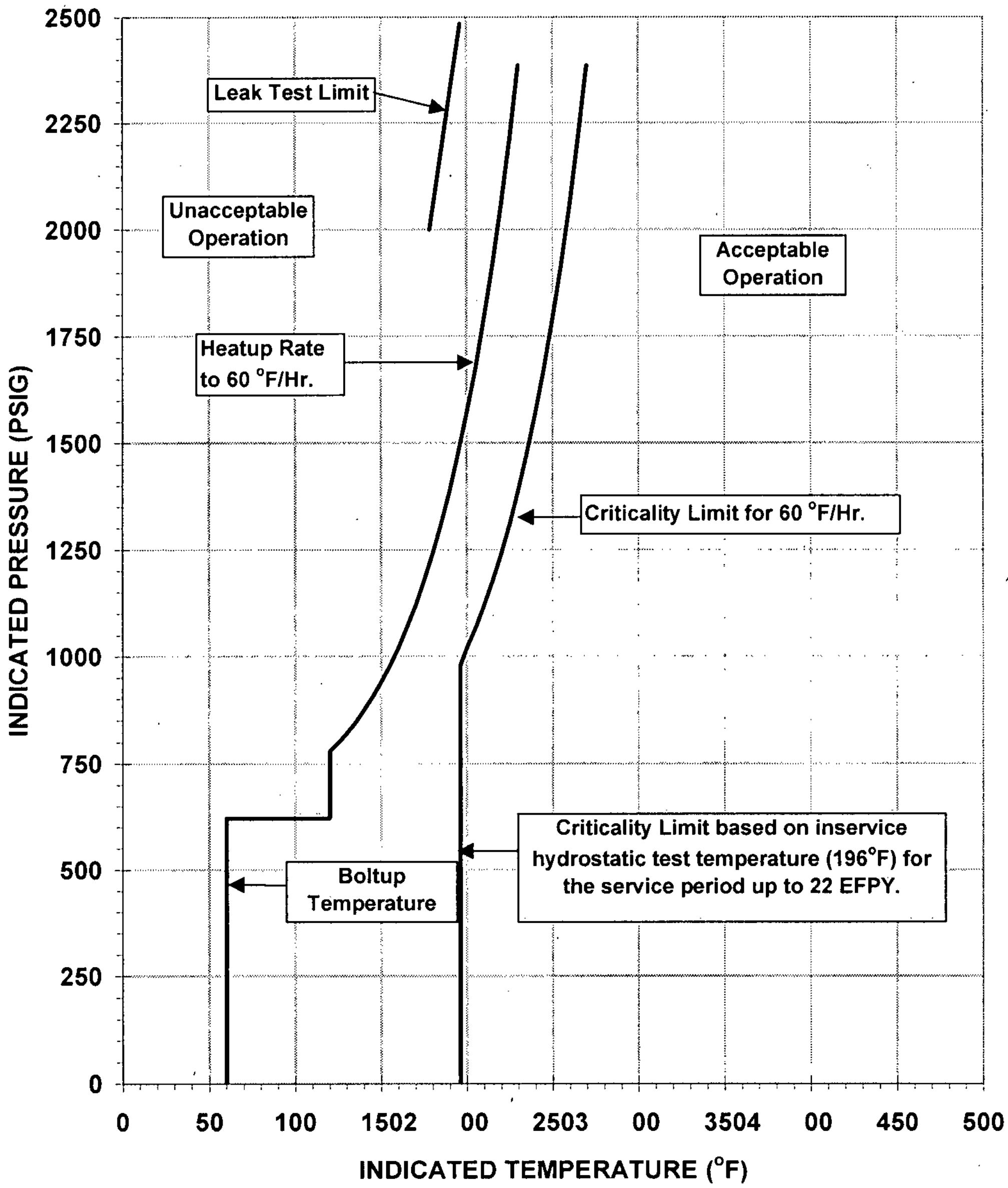

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

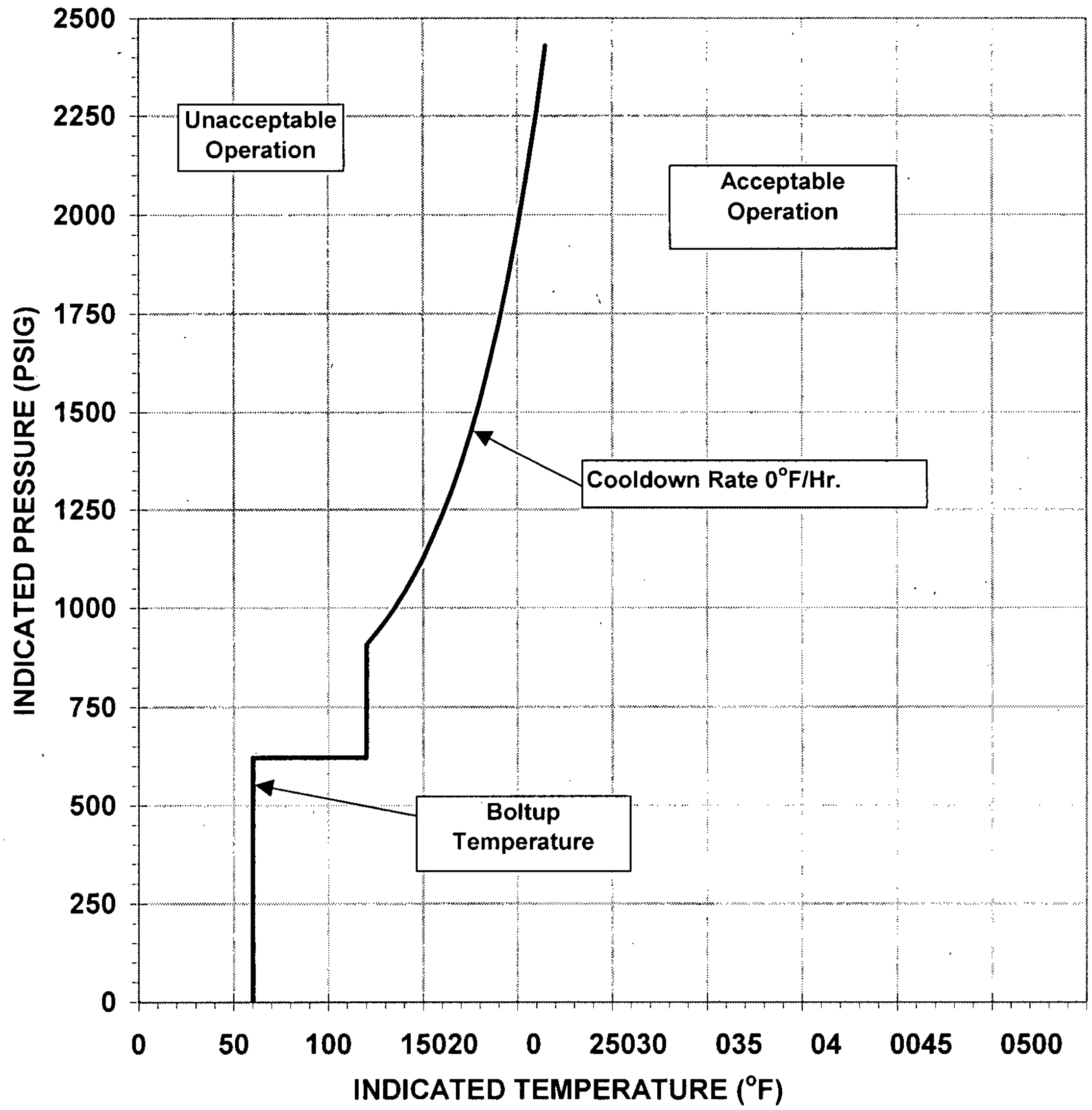

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 0°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

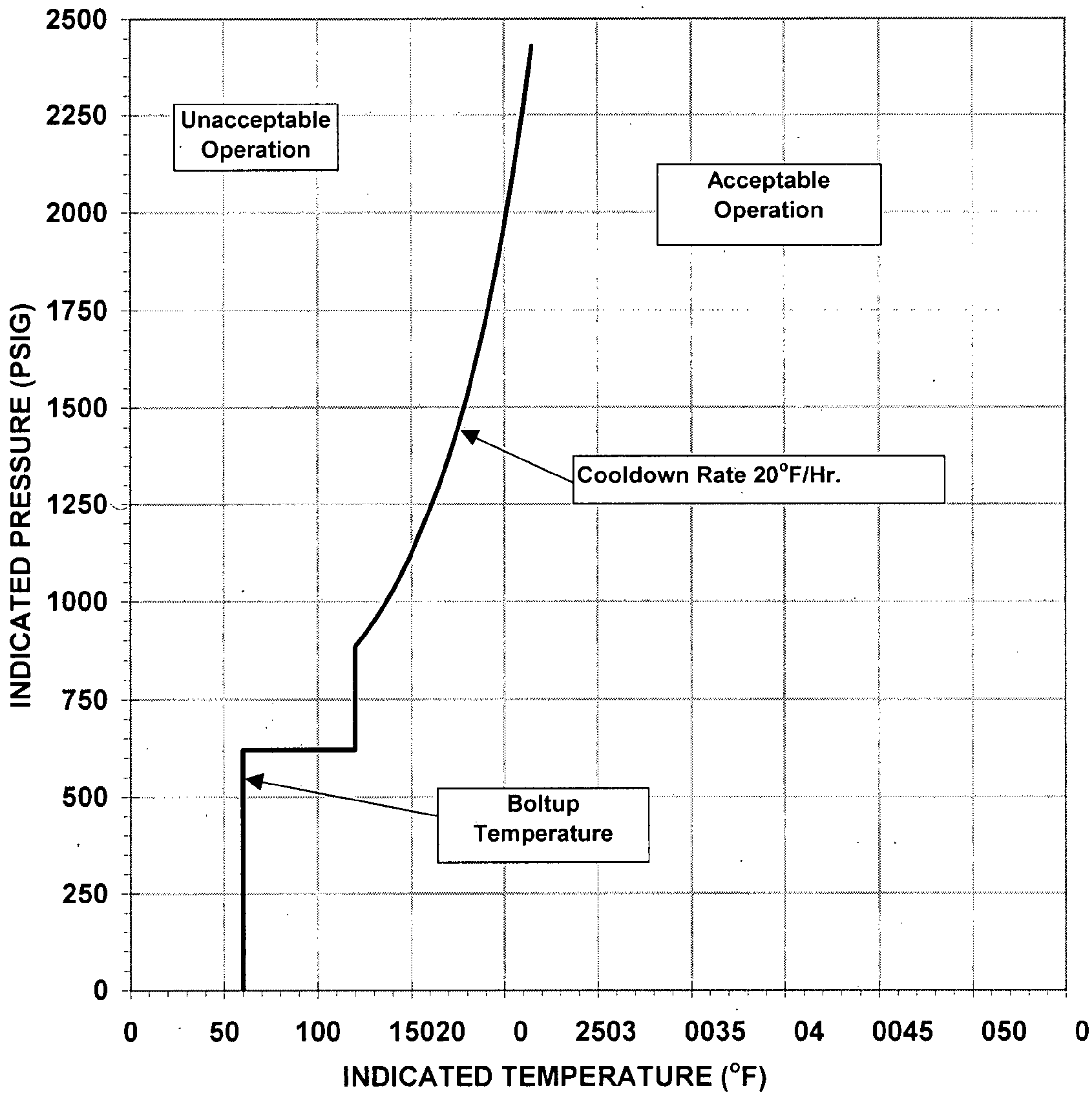

Figure 5.2-3 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 20°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

<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 40°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

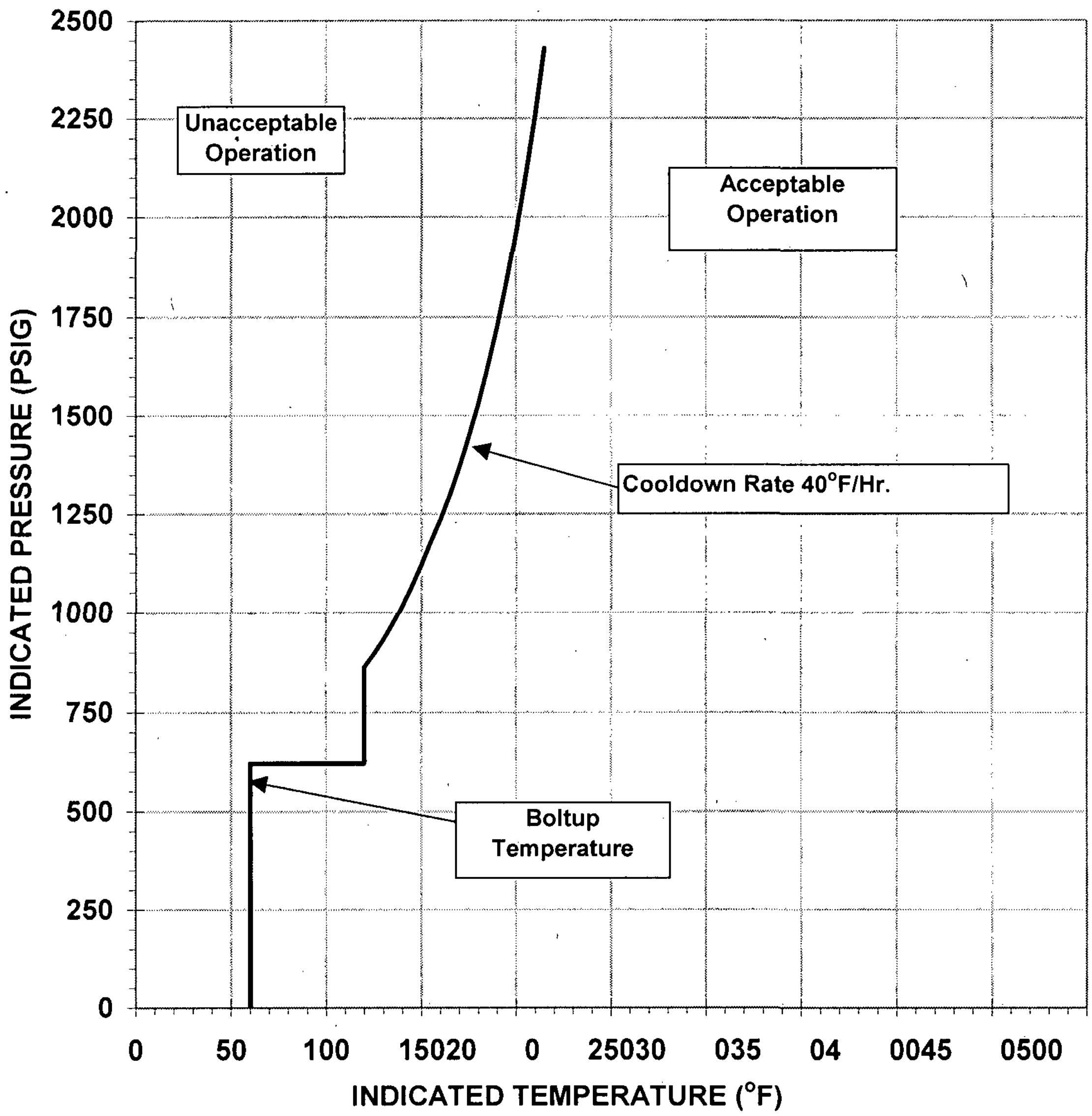

Figure 5.2-4 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 40°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

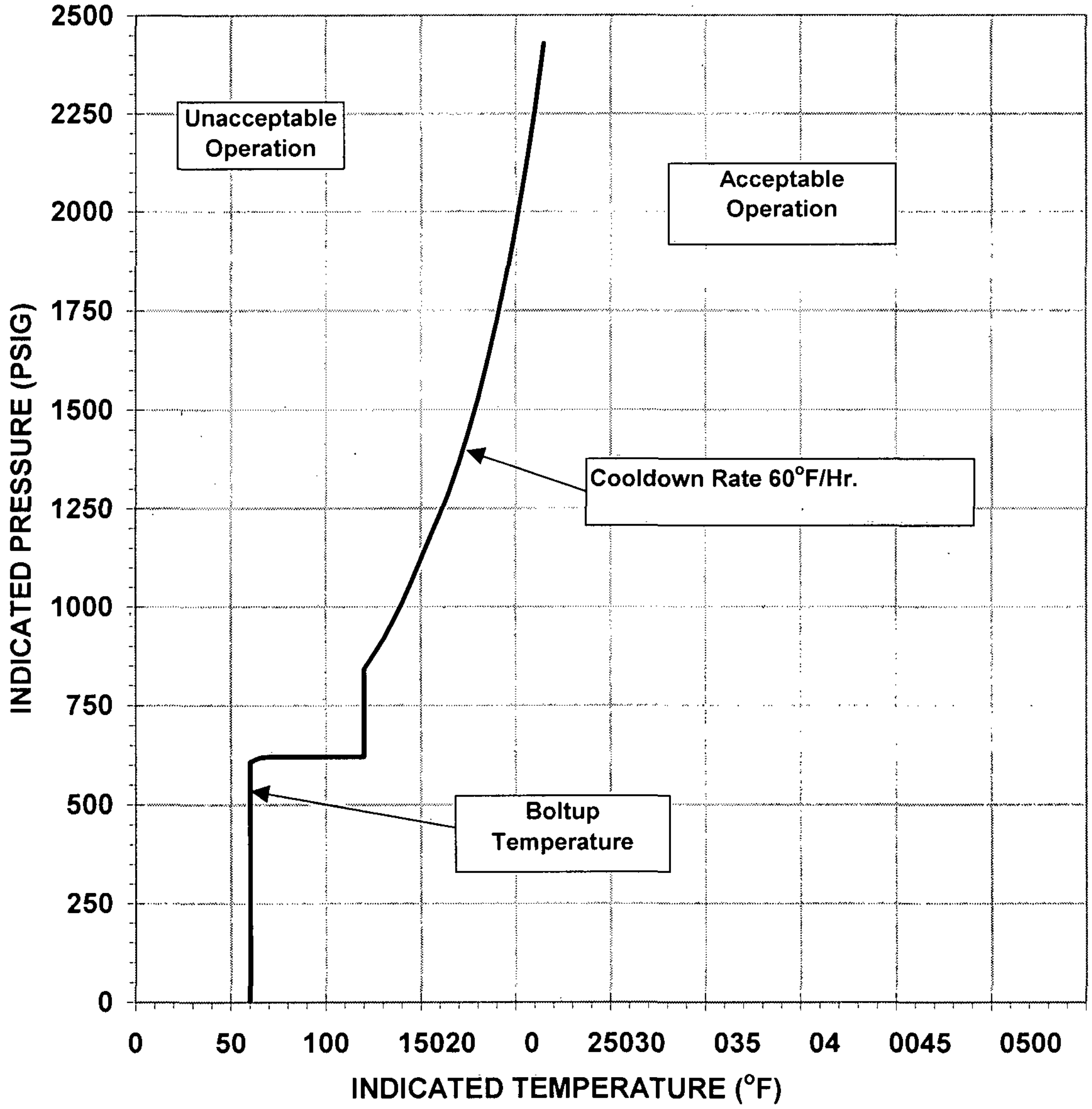

Figure 5.2-5 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 60°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

<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 100°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

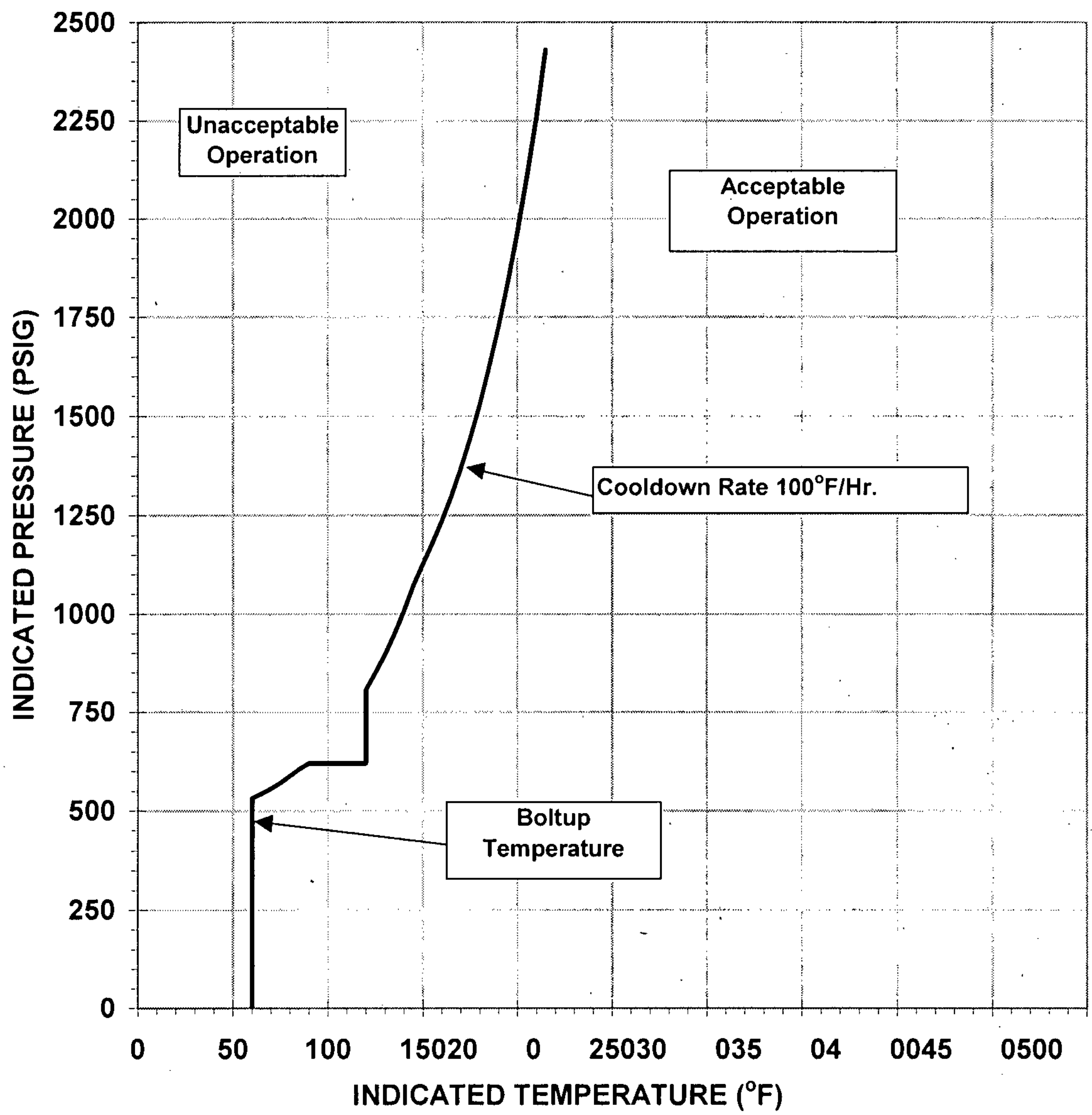

Figure 5.2-6 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 100°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

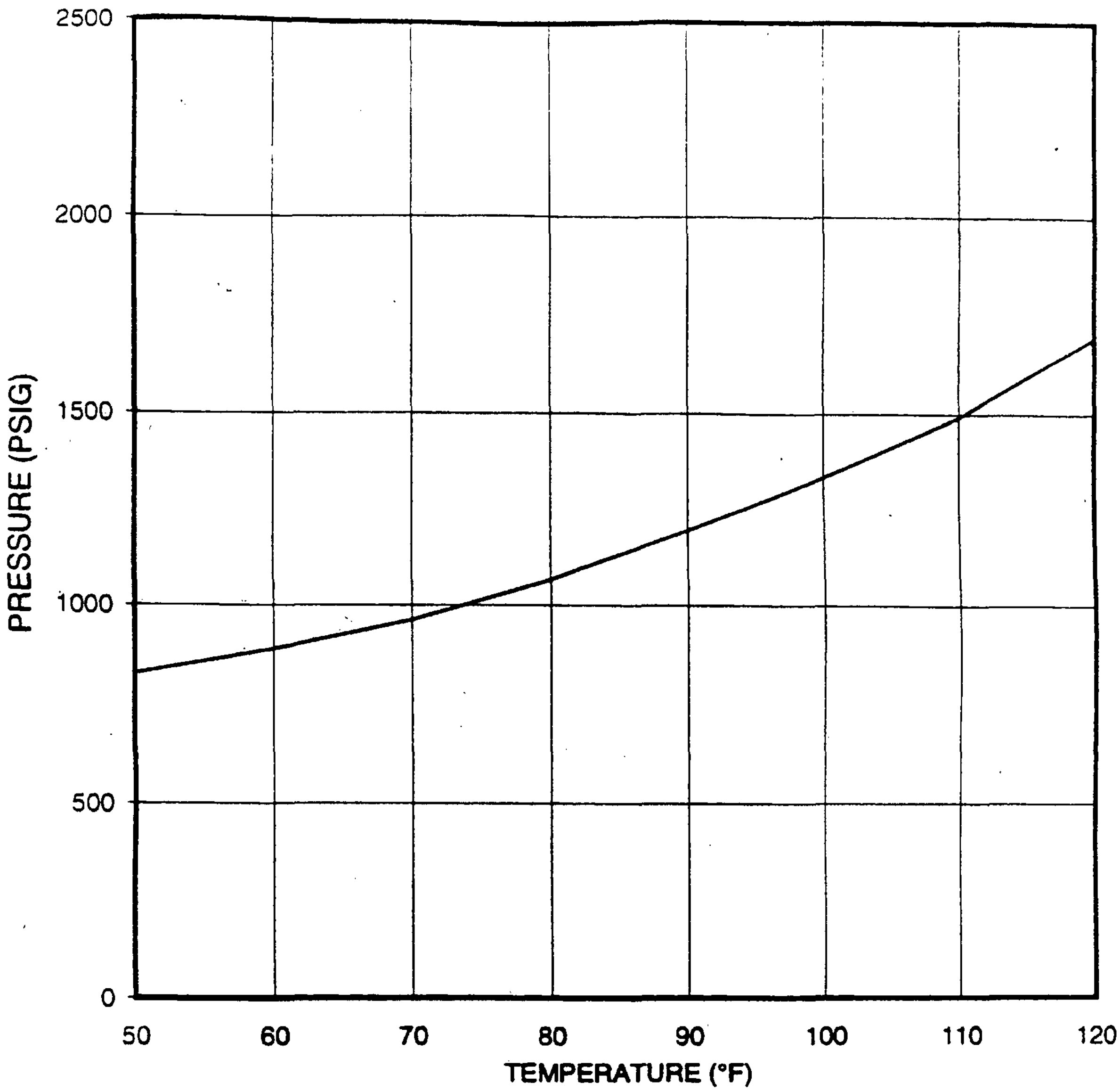

Figure 5.2-7 (Page 1 of 1)
Isolated Loop Pressure – Temperature Limit Curve (LCO 3.4.3)

See Table 5.2-4 for RCP restrictions.

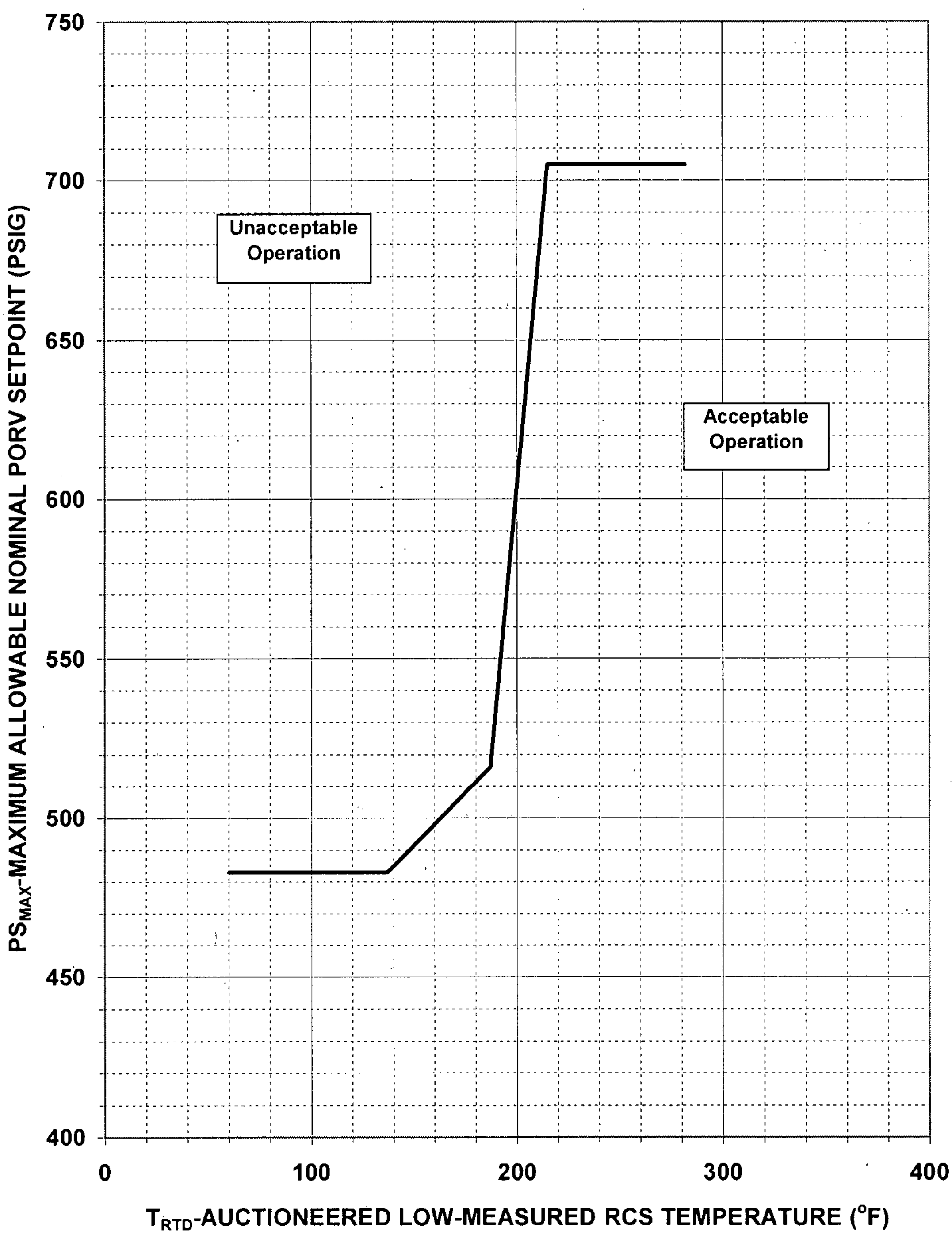

Figure 5.2-8 (Page 1 of 1)
Maximum Allowable Nominal PORV Setpoint for the Overpressure Protection System (LCO 3.4.12)

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*ΔRT_{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

Table 5.2-12 (Page 2 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)
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

<u>Notes:</u>

(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.