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June 25, 2007
L-07-087

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

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Core Operating Limits Report, COLR 18-3
Pressure and Temperature Limits Report, Revision 3

Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Core Operating Limits Report, COLR 13-2
Pressure and Temperature Limits Report, Revision 2

FirstEnergy Nuclear Operating Company (FENOC) hereby submits revisions of the Core Operating Limits Report (COLR) and the Pressure and Temperature Limits Report (PTLR) for Beaver Valley Power Station (BVPS) Unit No. 1 and Unit No. 2 as required by Section 5.6.3 and 5.6.4 of the BVPS Technical Specifications. The BVPS Unit No. 1 COLR 18-3 is included as Enclosure 1 and Revision 3 of the BVPS Unit No. 1 PTLR is included as Enclosure 2. The BVPS Unit No. 2 COLR 13-2 is included as Enclosure 3 and Revision 2 of the BVPS Unit No. 2 PTLR is included as Enclosure 4.

The revisions to the COLR and the PTLR for each unit, effective June 23, 2007, are associated with the implementation of the BVPS Improved Technical Specification Conversion License Amendments, 278 (Unit 1) and 161 (Unit 2).

No regulatory commitments are contained in this submittal. If you have questions or require additional information, please contact Mr. Thomas A. Lentz, Manager - FENOC Fleet Licensing, at (330) 761-6071.

Sincerely,

James H. Lash

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NRR

Enclosures:

1. Beaver Valley Power Station Unit No. 1, Core Operating Limits Report, COLR 18-3
2. Beaver Valley Power Station Unit No. 1, Pressure and Temperature Limits Report, Revision 3
3. Beaver Valley Power Station Unit No. 2, Core Operating Limits Report, COLR 13-2
4. Beaver Valley Power Station Unit No. 2, Pressure and Temperature Limits Report, Revision 2

c: Ms. N. S. Morgan, NRR Project Manager
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Mr. S. J. Collins, NRC Region I Administrator
Mr. D. J. Allard, Director BRP/DEP
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Enclosure 1

Beaver Valley Power Station

Unit No. 1

Core Operating Limits Report

COLR 18-3

5.0 ADMINISTRATIVE CONTROLS

5.1 Core Operating Limits Report

This Core Operating Limits Report provides the cycle specific parameter limits developed in accordance with the NRC approved methodologies specified in Technical Specification Administrative Control 5.6.3.

5.1.1 SL 2.1.1 Reactor Core Safety Limits

See Figure 5.1-1.

5.1.2 SHUTDOWN MARGIN (SDM)

a. In MODES 1, 2, 3, and 4, SHUTDOWN MARGIN shall be ≥ 1.77% Δk/k.[(1)]

b. Prior to manually blocking the Low Pressurizer Pressure Safety Injection Signal, the Reactor Coolant System shall be borated to ≥ the MODE 5 boron concentration and shall remain ≥ this boron concentration at all times when this signal is blocked.

c. In MODE 5, SHUTDOWN MARGIN shall be ≥ 1.0% Δk/k.

5.1.3 LCO 3.1.3 Moderator Temperature Coefficient (MTC)

a. Upper Limit - MTC shall be maintained within the acceptable operation limit specified in Technical Specification Figure 3.1.3-1.

b. Lower Limit - MTC shall be maintained less negative than - 4.4×10^{-4} Δk/k/°F at RATED THERMAL POWER.

c. 300 ppm Surveillance Limit: (- 37 pcm/°F)

d. 60 ppm Surveillance Limit: (- 43 pcm/°F)

5.1.4 LCO 3.1.5 Shutdown Bank Insertion Limits

The Shutdown Banks shall be withdrawn to at least 225 steps.[(2)]

5.1.5 LCO 3.1.6 Control Bank Insertion Limits

a. Control Banks A and B shall be withdrawn to at least 225 steps.[(2)]

b. Control Banks C and D shall be limited in physical insertion as shown in Figure 5.1-2.[(2)]

c. Sequence Limits - The sequence of withdrawal shall be A, B, C and D bank, in that order.

d. Overlap Limits[(2)] - Overlap shall be such that step 129 on banks A, B, and C corresponds to step 1 on the following bank. When C bank is fully withdrawn, these limits are verified by confirming D bank is withdrawn at least to a position equal to the all-rods-out position minus 128 steps.

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(1) The MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ SDM requirements are included to address SDM requirements (e.g., MODE 1 Required Actions to verify SDM) that are not within the applicability of LCO 3.1.1, SHUTDOWN MARGIN (SDM).

(2) As indicated by the group demand counter

5.1 Core Operating Limits Report

5.1.6 LCO 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

The Heat Flux Hot Channel Factor - $F_Q(Z)$ limit is defined by:

$$F_Q(Z) \leq \left[\frac{CFQ}{P}\right] * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{CFQ}{0.5}\right] * K(Z) \qquad \text{for } P \leq 0.5$$

Where: CFQ = 2.40 $\quad P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

K(Z) = the function obtained from Figure 5.1-3.

$$F_Q^C(Z) = F_Q^M(Z) * 1.0815$$

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

The W(Z) values are provided in Table 5.1-1.

The $F_Q(Z)$ penalty function, applied when the analytic $F_Q(Z)$ function increases from one monthly measurement to the next, is provided in Table 5.1-2.

5.1.7 LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

$$F_{\Delta H}^N \leq CF_{\Delta H} * (1 + PF_{\Delta H} (1-P))$$

Where: $CF_{\Delta H} = 1.62$

$PF_{\Delta H} = 0.3$

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

5.1 Core Operating Limits Report

5.1.8 LCO 3.2.3 Axial Flux Difference (AFD)

The AFD acceptable operation limits are provided in Figure 5.1-4.

5.1.9 LCO 3.3.1 Reactor Trip System Instrumentation - Overtemperature and Overpower ΔT Parameter Values from Table Notations 1 and 2

a. Overtemperature ΔT Setpoint Parameter Values:

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K1 \leq 1.242$
Overtemperature ΔT reactor trip setpoint Tavg coefficient	$K2 \geq 0.0183/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K3 \geq 0.001/psia$
Tavg at RATED THERMAL POWER	$T' \leq 577.9°F$[1]
Nominal pressurizer pressure	$P' \geq 2250$ psia
Measured reactor vessel average temperature lead/lag time constants	$\tau_1 \geq 30$ secs $\tau_2 \leq 4$ secs
Measured reactor vessel ΔT lag time constant	$\tau_4 \leq 6$ secs
Measured reactor vessel average temperature lag time constant	$\tau_5 \leq 2$ secs

f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37% and +15%, $f(\Delta I) = 0$ (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(1) T' represents the cycle-specific Full Power Tavg value used in core design.

5.1 Core Operating Limits Report

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37%, the ΔT trip setpoint shall be automatically reduced by 2.52% of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +15%, the ΔT trip setpoint shall be automatically reduced by 1.47% of its value at RATED THERMAL POWER.

b. Overpower ΔT Setpoint Parameter Values:

Parameter	Value
Overpower ΔT reactor trip setpoint	$K4 \leq 1.085$
Overpower ΔT reactor trip setpoint Tavg rate/lag coefficient	$K5 \geq 0.02/°F$ for increasing average temperature
	$K5 = 0/°F$ for decreasing average temperature
Overpower ΔT reactor trip setpoint Tavg heatup coefficient	$K6 \geq 0.0021/°F$ for $T > T''$ $K6 = 0/°F$ for $T \leq T''$
Tavg at RATED THERMAL POWER	$T'' \leq 577.9°F$[(2)]
Measured reactor vessel average temperature rate/lag time constant	$\tau_3 \geq 10$ secs
Measured reactor vessel ΔT lag time constant	$\tau_4 \leq 6$ secs
Measured reactor vessel average temperature lag time constant	$\tau_5 \leq 2$ secs

(2) T" represents the cycle-specific Full Power Tavg value used in core design.

5.1 Core Operating Limits Report

5.1.10 LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Parameter	Indicated Value
Reactor Coolant System Tavg	Tavg ≤ 581.5°F(1)
Pressurizer Pressure	Pressure ≥ 2218 psia(2)
Reactor Coolant System Total Flow Rate	Flow ≥ 267,300 gpm(3)

(1) The Reactor Coolant System (RCS) indicated T_{avg} value is determined by adding the appropriate allowances for rod control operation and verification via control board indication (3.6°F) to the cycle specific full power T_{avg} used in the core design.

(2) The pressurizer pressure value includes allowances for pressurizer pressure control operation and verification via control board indication.

(3) The RCS total flow rate includes allowances for normalization of the cold leg elbow taps with a beginning of cycle precision RCS flow calorimetric measurement and verification on a periodic basis via control board indication.

5.1 Core Operating Limits Report

5.1.11 LCO 3.9.1 Boron Concentration (MODE 6)

The boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained ≥ 2400 ppm. This value includes a 50 ppm conservative allowance for uncertainties.

5.1 Core Operating Limits Report

5.1.12 References

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

2. WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions," September 1986.

3. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

4. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.

5. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

6. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

7. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

8. Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

9. Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

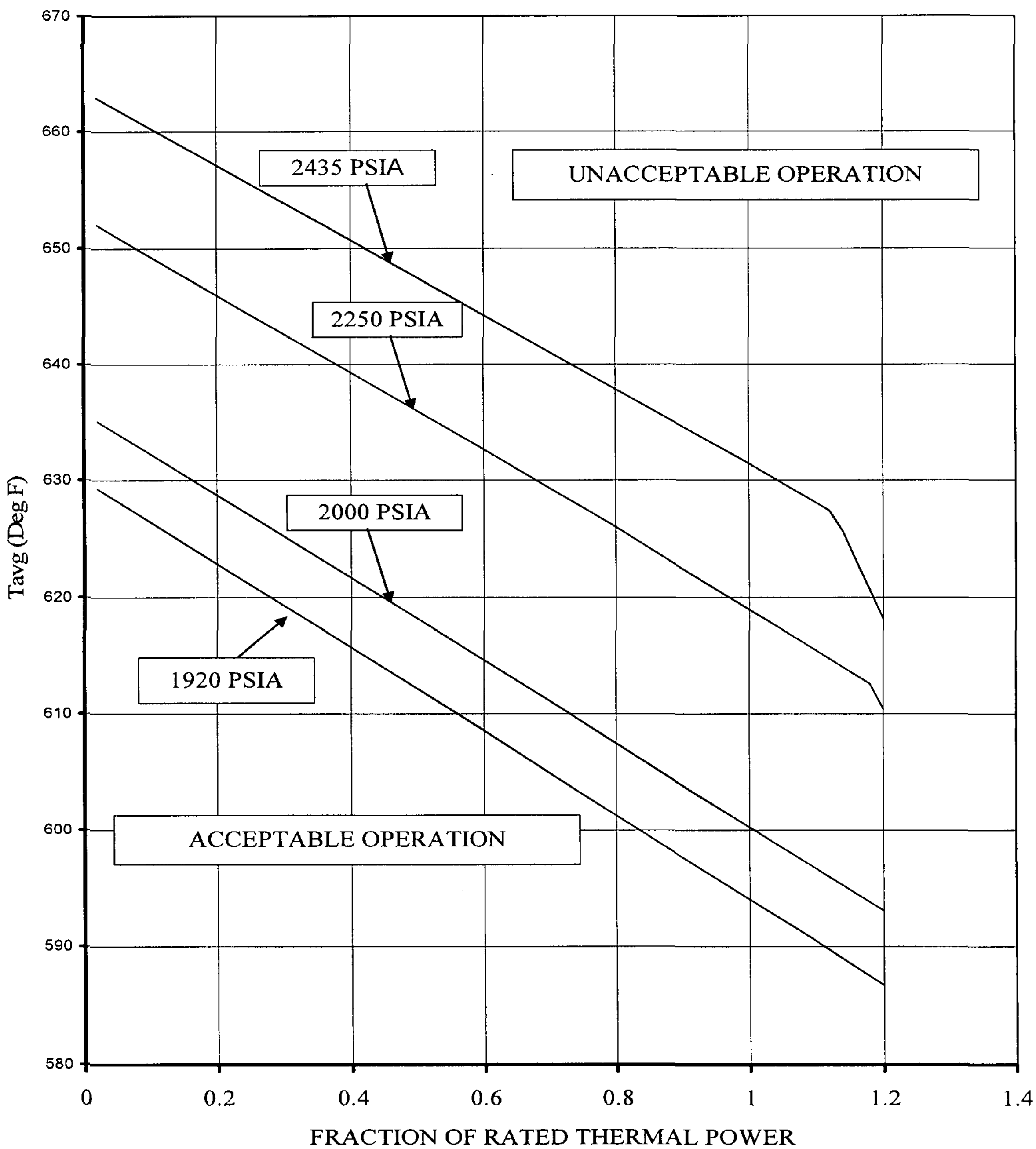

Figure 5.1-1 (Page 1 of 1)

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION
(Technical Specification Safety Limit 2.1.1)

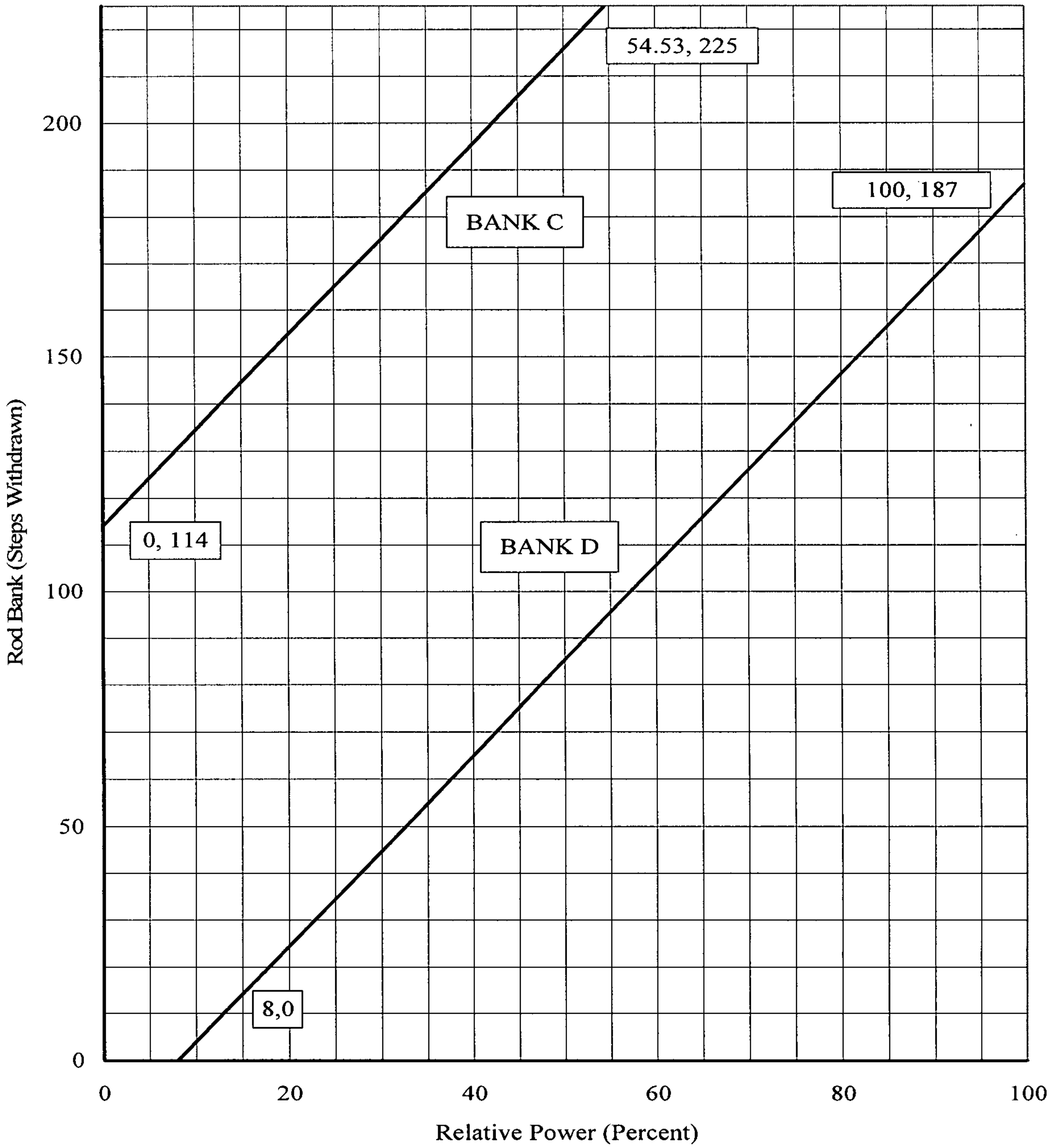

Figure 5.1-2 (Page 1 of 1)

CONTROL ROD INSERTION LIMITS

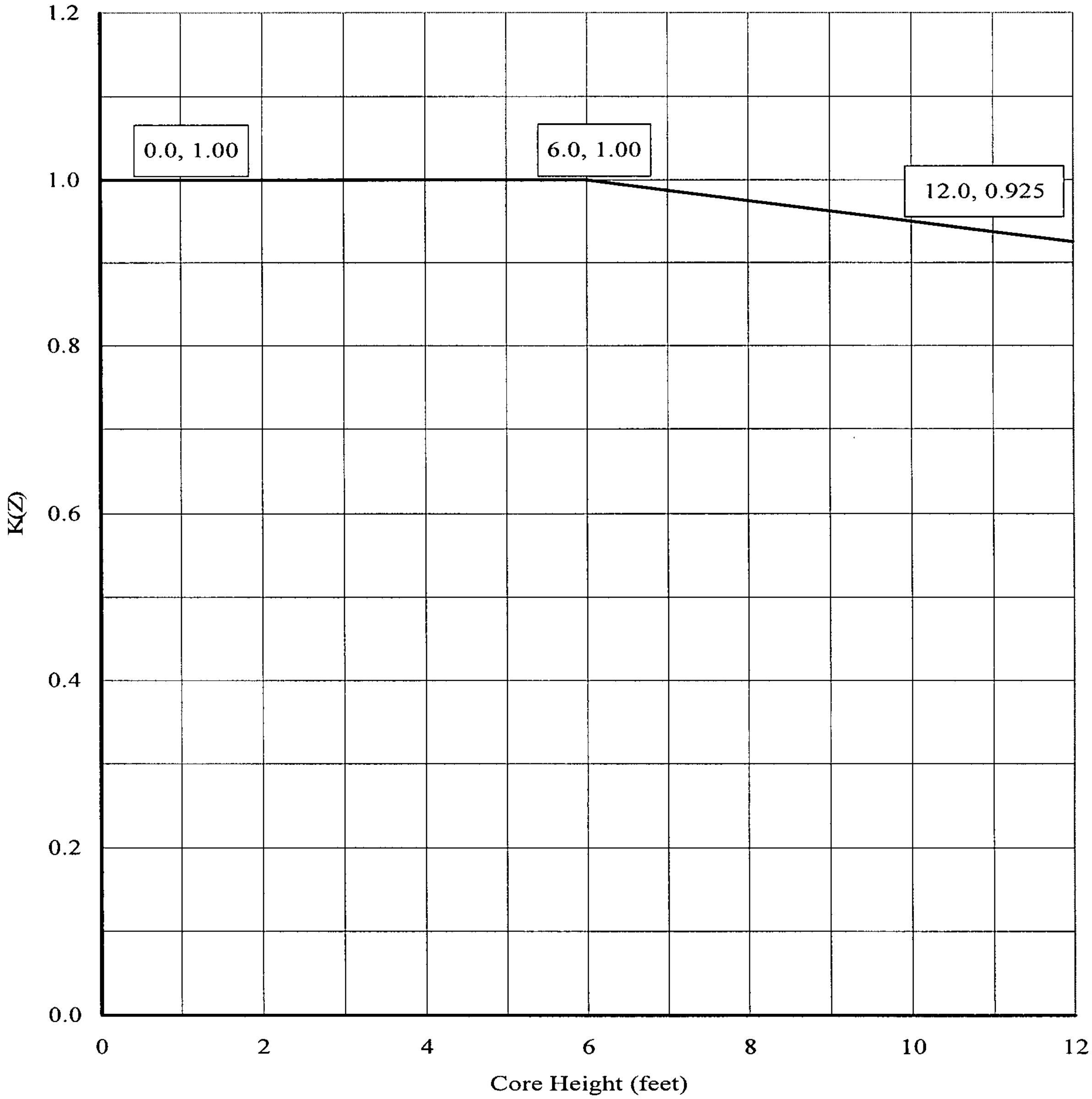

Figure 5.1-3 (Page 1 of 1)

F_QT NORMALIZED OPERATING ENVELOPE, K(Z)

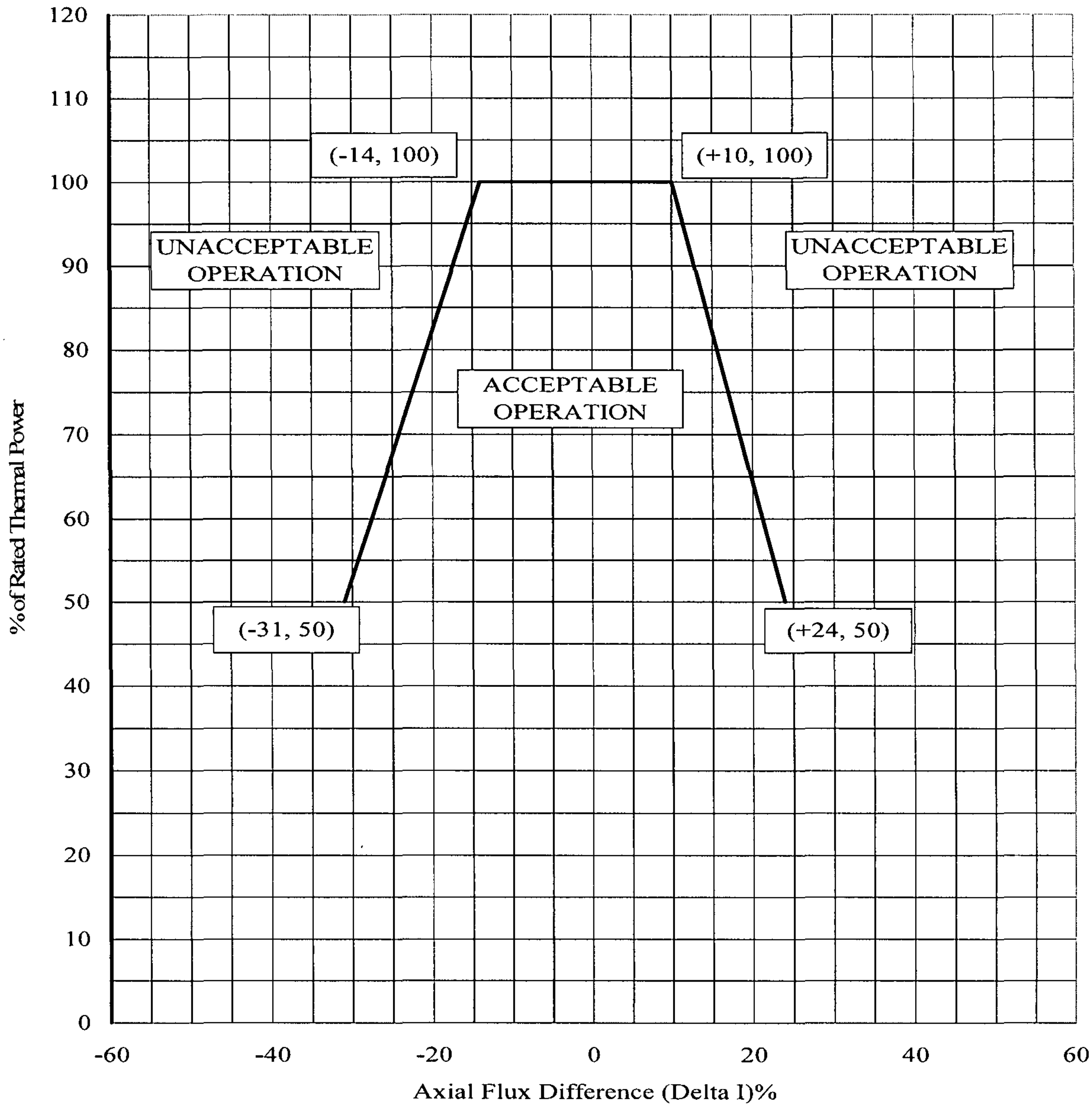

Figure 5.1-4 (Page 1 of 1)

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF

PERCENT OF RATED THERMAL POWER FOR RAOC

Table 5.1-1 (Page 1 of 2)
F_Q Surveillance W(Z) Function versus Burnup

Exclusion Zone	Axial Point	Elevation (feet)	3600 MWD/MTU	10000 MWD/MTU	16000 MWD/MTU
*	1	12.0	1.0000	1.0000	1.0000
*	2	11.8	1.0000	1.0000	1.0000
*	3	11.6	1.0000	1.0000	1.0000
*	4	11.4	1.0000	1.0000	1.0000
*	5	11.2	1.0000	1.0000	1.0000
*	6	11.0	1.0000	1.0000	1.0000
*	7	10.8	1.0000	1.0000	1.0000
	8	10.6	1.3293	1.3167	1.3044
	9	10.4	1.3254	1.2928	1.2820
	10	10.2	1.3213	1.2644	1.2607
	11	10	1.3136	1.2405	1.2430
	12	9.8	1.3036	1.2300	1.2299
	13	9.6	1.2957	1.2204	1.2196
	14	9.4	1.2865	1.2089	1.2110
	15	9.2	1.2745	1.2010	1.2122
	16	9.0	1.2644	1.1999	1.2303
	17	8.8	1.2585	1.2097	1.2457
	18	8.6	1.2573	1.2249	1.2575
	19	8.4	1.2567	1.2386	1.2674
	20	8.2	1.2534	1.2489	1.2744
	21	8.0	1.2487	1.2563	1.2788
	22	7.8	1.2415	1.2608	1.2804
	23	7.6	1.2326	1.2623	1.2793
	24	7.4	1.2216	1.2611	1.2756
	25	7.2	1.2090	1.2572	1.2693
	26	7.0	1.2019	1.2507	1.2603
	27	6.8	1.1957	1.2419	1.2513
	28	6.6	1.1875	1.2308	1.2422
	29	6.4	1.1776	1.2178	1.2311
	30	6.2	1.1663	1.2029	1.2181

Note: Top and Bottom 10% Excluded

Table 5.1-1 (Page 2 of 2)
F_Q Surveillance W(Z) Function versus Burnup

Exclusion Zone	Axial Point	Elevation (feet)	3600 MWD/MTU	10000 MWD/MTU	16000 MWD/MTU
	31	6.0	1.1542	1.1865	1.2027
	32	5.8	1.1416	1.1685	1.1884
	33	5.6	1.1338	1.1521	1.1791
	34	5.4	1.1330	1.1421	1.1761
	35	5.2	1.1307	1.1402	1.1760
	36	5.0	1.1292	1.1398	1.1750
	37	4.8	1.1285	1.1398	1.1725
	38	4.6	1.1269	1.1389	1.1691
	39	4.4	1.1252	1.1375	1.1646
	40	4.2	1.1253	1.1351	1.1591
	41	4.0	1.1256	1.1349	1.1530
	42	3.8	1.1251	1.1364	1.1461
	43	3.6	1.1242	1.1401	1.1406
	44	3.4	1.1227	1.1429	1.1401
	45	3.2	1.1198	1.1453	1.1400
	46	3.0	1.1216	1.1471	1.1449
	47	2.8	1.1297	1.1545	1.1594
	48	2.6	1.1388	1.1732	1.1722
	49	2.4	1.1482	1.1925	1.1884
	50	2.2	1.1609	1.2113	1.2059
	51	2.0	1.1750	1.2297	1.2227
	52	1.8	1.1857	1.2474	1.2391
	53	1.6	1.1922	1.2639	1.2547
	54	1.4	1.1981	1.2790	1.2691
*	55	1.2	1.0000	1.0000	1.0000
*	56	1.0	1.0000	1.0000	1.0000
*	57	0.8	1.0000	1.0000	1.0000
*	58	0.6	1.0000	1.0000	1.0000
*	59	0.4	1.0000	1.0000	1.0000
*	60	0.2	1.0000	1.0000	1.0000
*	61	0.0	1.0000	1.0000	1.0000

Note: Top and Bottom 10% Excluded

Table 5.1-2 (Page 1 of 1)
$F_Q(Z)$ Penalty Factor versus Burnup

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor
All Burnups	1.02

Note: The Penalty Factor, to be applied to $F_Q(Z)$ in accordance with Technical Specification Surveillance Requirement (SR) 3.2.1.2, is the maximum factor by which $F_Q(Z)$ is expected to increase over a 39 Effective Full Power Day (EFPD) interval (surveillance interval of 31 EFPD plus the maximum allowable extension not to exceed 25% of the surveillance interval per Technical Specification SR 3.0.2) starting from the burnup at which the $F_Q(Z)$ was determined.

Enclosure 2

Beaver Valley Power Station

Unit No. 1

Pressure and Temperature Limits Report

Revision 3

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.

5.2 Pressure and Temperature Limits Report

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 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 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 5.5-3 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 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.

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 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 22 EFPY.

Table 5.2-7, taken from Reference 2, shows the calculation of ARTs for 22 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 5, provides RT_{PTS} values for 45 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-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.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	INTERMEDIATE & LOWER SHELL PLATE
LIMITING ART VALUES AT 21 EFPY:	1/4T, 233°F
	3/4T, 196°F

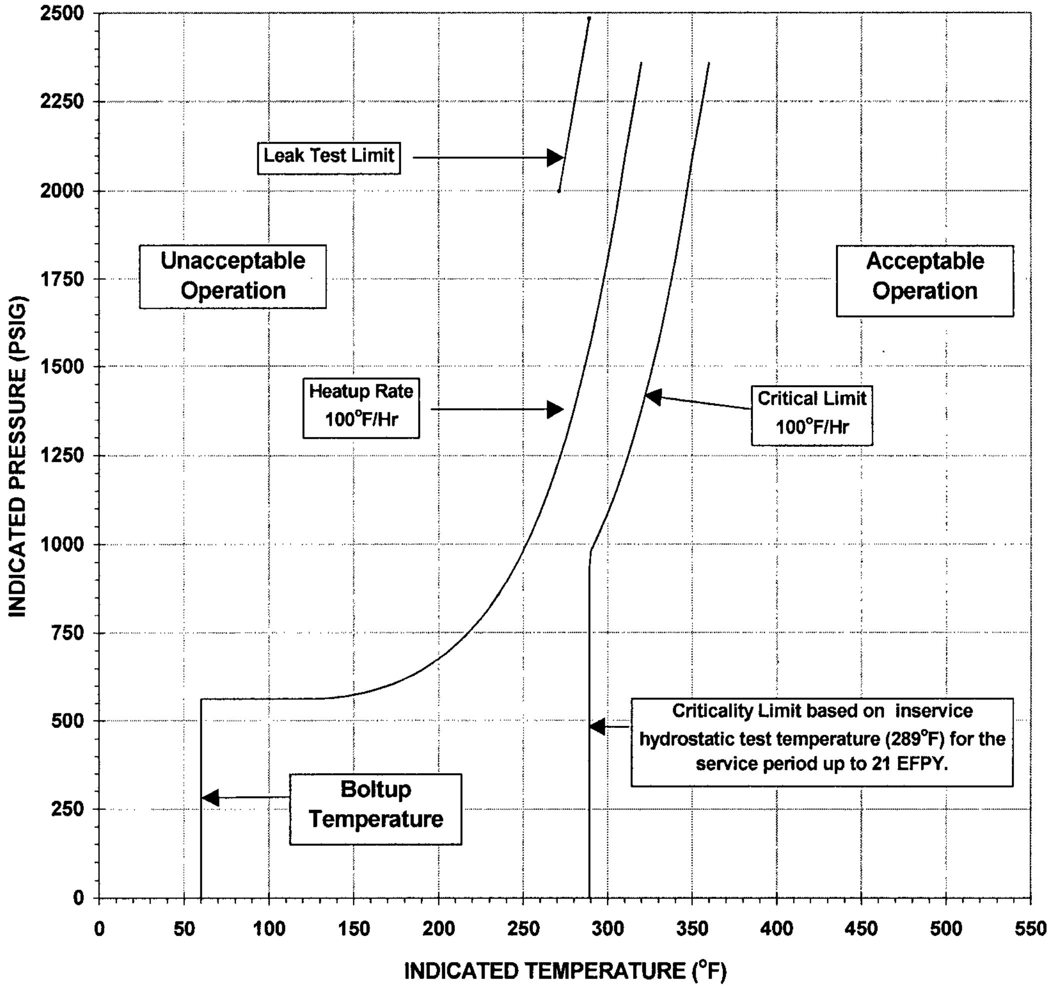

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 21 EFPY (LCO 3.4.3)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	INTERMEDIATE & LOWER SHELL PLATE
LIMITING ART VALUES AT 21 EFPY:	1/4T, 233°F
	3/4T, 196°F

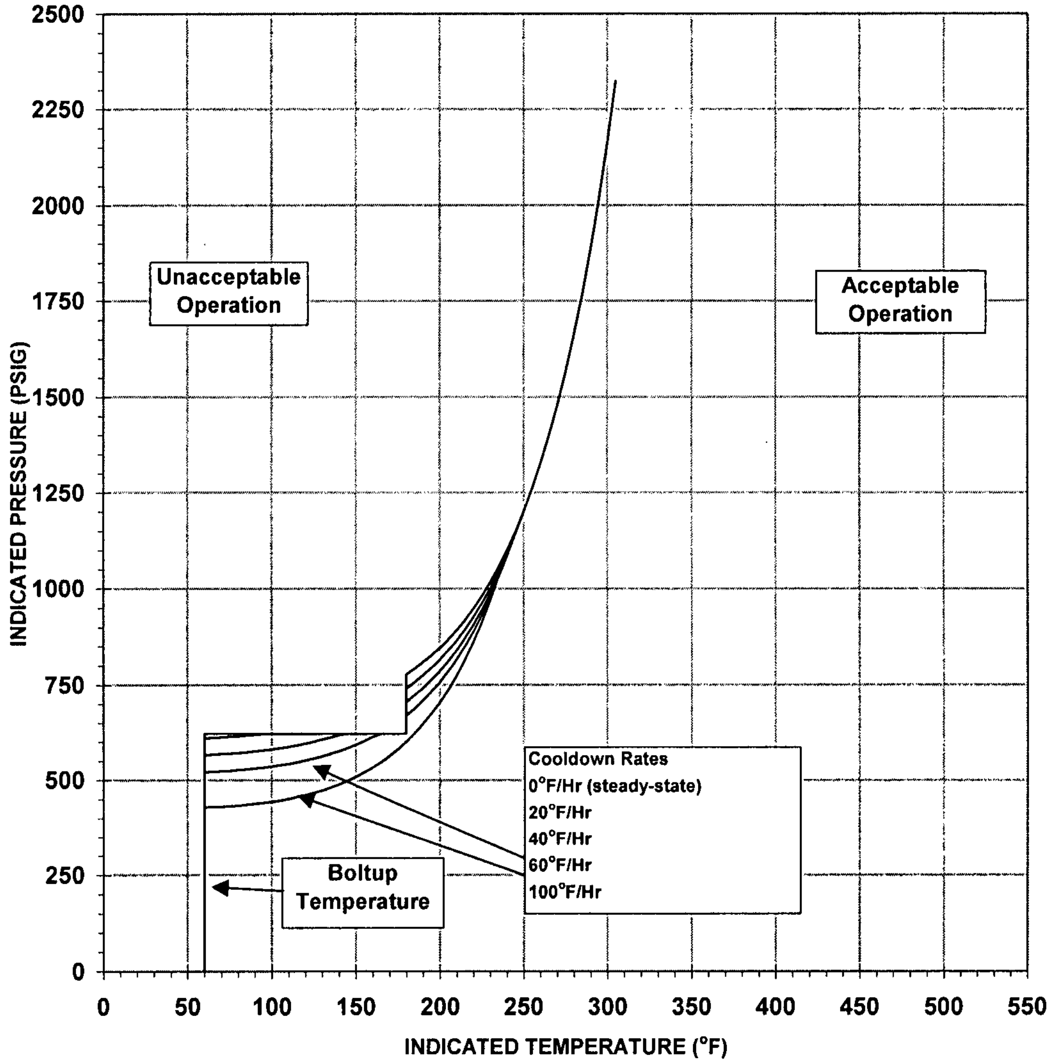

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown
Limitations Applicable for the First 21 EFPY (LCO 3.4.3)

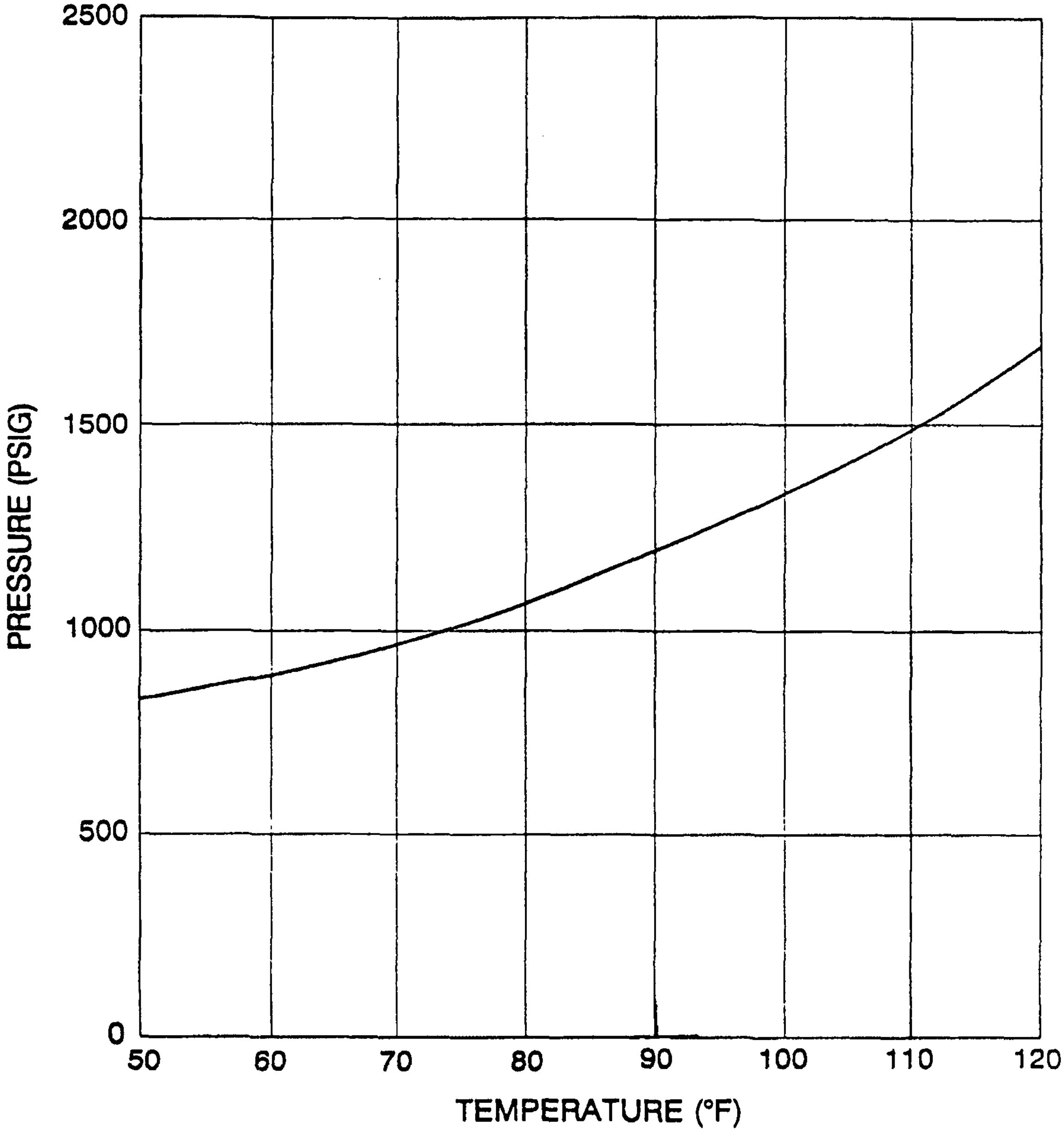

Figure 5.2-3 (Page 1 of 1)
Isolated Loop Pressure - Temperature Limit Curve (LCO 3.4.3)

Table 5.2-1 (Page 1 of 1)
Heatup Curve Data Points for 22 EFPY (LCO 3.4.3)

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 5.2-2 (Page 1 of 2)
Cooldown Curve Data Points for 22 EFPY (LCO 3.4.3)

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 5.2-2 (Page 2 of 2)
Cooldown Curve Data Points for 22 EFPY (LCO 3.4.3)

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		

Table 5.2-3 (Page 1 of 1)

Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)

FUNCTION	SETPOINT
OPPS Enable Temperature	343°F
PORV Setpoint	≤ 403 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) ÷ (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) ÷ (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 $f^{(b)}$	$FF^{(c)}$	$\Delta RT_{NDT}^{(d)}$	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
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = (246.7) \div (2.91) = 84.8°F$					
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
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = (603.3) \div (2.695) = 223.9°F$					

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 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 Weld Metal Heat 90136	97°	0.2291	0.0699	546.7	0.627	72.3
	104°	0.2291	0.0699	546.7	0.909	67.4
	284°	0.2291	0.0699	546.7	1.41	68.0
Fort Calhoun Weld Metal Heat 305414	W-225	0.35	0.60	527	0.553	238
	W-265	0.35	0.60	534	0.771	221
	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 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 5.2-7 (Page 1 of 1)

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

(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 5.2-10 (Page 1 of 1)

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.

Enclosure 3

Beaver Valley Power Station

Unit No. 2

Core Operating Limits Report

COLR 13-2

5.0 ADMINISTRATIVE CONTROLS

5.1 Core Operating Limits Report

This Core Operating Limits Report provides the cycle specific parameter limits developed in accordance with the NRC approved methodologies specified in Technical Specification Administrative Control 5.6.3.

5.1.1 SL 2.1.1 Reactor Core Safety Limits

See Figure 5.1-1.

5.1.2 SHUTDOWN MARGIN (SDM)

a. In MODES 1, 2, 3, and 4, SHUTDOWN MARGIN shall be ≥ 1.77% Δk/k.(1)

b. Prior to manually blocking the Low Pressurizer Pressure Safety Injection Signal, the Reactor Coolant System shall be borated to ≥ the MODE 5 boron concentration and shall remain ≥ this boron concentration at all times when this signal is blocked.

c. In MODE 5, SHUTDOWN MARGIN shall be ≥ 1.0% Δk/k.

5.1.3 LCO 3.1.3 Moderator Temperature Coefficient (MTC)

a. Upper Limit - MTC shall be maintained within the acceptable operation limit specified in Technical Specification Figure 3.1.3-1.

b. Lower Limit - MTC shall be maintained less negative than -4.4×10^{-4} Δk/k/°F at RATED THERMAL POWER.

c. 300 ppm Surveillance Limit: (- 36 pcm/°F)

d. 60 ppm Surveillance Limit: (- 42 pcm/°F)

5.1.4 LCO 3.1.5 Shutdown Bank Insertion Limits

The Shutdown Banks shall be withdrawn to at least 225 steps.(2)

5.1.5 LCO 3.1.6 Control Bank Insertion Limits

a. Control Banks A and B shall be withdrawn to at least 225 steps.(2)

b. Control Banks C and D shall be limited in physical insertion as shown in Figure 5.1-2.(2)

c. Sequence Limits - The sequence of withdrawal shall be A, B, C and D bank, in that order.

d. Overlap Limits(2) - Overlap shall be such that step 129 on banks A, B, and C corresponds to step 1 on the following bank. When C bank is fully withdrawn, these limits are verified by confirming D bank is withdrawn at least to a position equal to the all-rods-out position minus 128 steps.

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(1) The MODE 1 and MODE 2 with $k_{eff} \geq 1.0$ SDM requirements are included to address SDM requirements (e.g., MODE 1 Required Actions to verify SDM) that are not within the applicability of LCO 3.1.1, SHUTDOWN MARGIN (SDM).

(2) As indicated by the group demand counter

5.1 Core Operating Limits Report

5.1.6 LCO 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

The Heat Flux Hot Channel Factor - $F_Q(Z)$ limit is defined by:

$$F_Q(Z) \leq \left[\frac{CFQ}{P}\right] * K(Z) \qquad \text{for } P > 0.5$$

$$F_Q(Z) \leq \left[\frac{CFQ}{0.5}\right] * K(Z) \qquad \text{for } P \leq 0.5$$

Where: $CFQ = 2.40$ $\qquad P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

K(Z) = the function obtained from Figure 5.1-3.

$$F_Q^C(Z) = F_Q^M(Z) * 1.0815$$

$$F_Q^W(Z) = F_Q^C(Z) * W(Z)$$

W(Z) values are provided in Table 5.1-1.

The $F_Q(Z)$ penalty function, applied when the analytic $F_Q(Z)$ function increases from one monthly measurement to the next, is provided in Table 5.1-2.

5.1.7 LCO 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

$$F_{\Delta H}^N \leq CF_{\Delta H} * (1 + PF_{\Delta H}(1 - P))$$

Where: $CF_{\Delta H} = 1.62$

$PF_{\Delta H} = 0.3$

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

5.1.8 LCO 3.2.3 Axial Flux Difference (AFD)

The AFD acceptable operation limits are provided in Figure 5.1-4.

5.1 Core Operating Limits Report

5.1.9 LCO 3.3.1 Reactor Trip System Instrumentation - Overtemperature and Overpower ΔT Parameter Values from Table Notations 3 and 4

a. Overtemperature ΔT Setpoint Parameter Values:

Parameter	Value
Overtemperature ΔT reactor trip setpoint	$K1 \leq 1.239$
Overtemperature ΔT reactor trip setpoint Tavg coefficient	$K2 \geq 0.0183/°F$
Overtemperature ΔT reactor trip setpoint pressure coefficient	$K3 \geq 0.001/psia$
Tavg at RATED THERMAL POWER	$T' \leq 576.2°F$[1]
Nominal pressurizer pressure	$P' \geq 2250$ psia
Measured reactor vessel ΔT lead/lag time constants (* The response time is toggled off to meet the analysis value of zero.)	$\tau_1 = 0$ sec* $\tau_2 = 0$ sec*
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 6$ secs
Measured reactor vessel average temperature lead/lag time constants	$\tau_4 \geq 30$ secs $\tau_5 \leq 4$ secs
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 2$ secs

f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -37% and +15%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

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(1) T' represents the cycle-specific Full Power Tavg value used in core design.

5.1 Core Operating Limits Report

(ii) For each percent that the magnitude of (q_t - q_b) exceeds -37%, the ΔT trip setpoint shall be automatically reduced by 2.52% of its value at RATED THERMAL POWER.

(iii) For each percent that the magnitude of (q_t - q_b) exceeds +15%, the ΔT trip setpoint shall be automatically reduced by 1.47% of its value at RATED THERMAL POWER.

b. <u>Overpower ΔT Setpoint Parameter Values</u>:

Parameter	Value
Overpower ΔT reactor trip setpoint	$K4 \leq 1.094$
Overpower ΔT reactor trip setpoint Tavg rate/lag coefficient	$K5 \geq 0.02/°F$ for increasing average temperature $K5 = 0/°F$ for decreasing average temperature
Overpower ΔT reactor trip setpoint Tavg heatup coefficient	$K6 \geq 0.0021/°F$ for $T > T''$ $K6 = 0/°F$ for $T \leq T''$
Tavg at RATED THERMAL POWER	$T'' \leq 576.2°F$[(1)]
Measured reactor vessel ΔT lead/lag time constants	$\tau_1 = 0$ sec* $\tau_2 = 0$ sec*

(* The response time is toggled off to meet the analysis value of zero.)

Parameter	Value
Measured reactor vessel ΔT lag time constant	$\tau_3 \leq 6$ secs
Measured reactor vessel average temperature lag time constant	$\tau_6 \leq 2$ secs
Measured reactor vessel average temperature rate/lag time constant	$\tau_7 \geq 10$ secs

(1) T" represents the cycle-specific Full Power Tavg value used in core design.

5.1 Core Operating Limits Report

5.1.10 LCO 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

Parameter	Indicated Value
Reactor Coolant System Tavg	Tavg ≤ 579.8°F(1)
Pressurizer Pressure	Pressure ≥ 2214 psia(2)
Reactor Coolant System Total Flow Rate	Flow ≥ 267,300 gpm(3)

(1) The Reactor Coolant System (RCS) indicated Tavg value is determined by adding the appropriate allowances for rod control operation and verification via control board indication (3.6°F) to the cycle specific full power Tavg used in the core design.

(2) The pressurizer pressure value includes allowances for pressurizer pressure control operation and verification via control board indication.

(3) The RCS total flow rate includes allowances for normalization of the cold leg elbow taps with a beginning of cycle precision RCS flow calorimetric measurement and verification on a periodic basis via control board indication.

5.1 Core Operating Limits Report

5.1.11 LCO 3.9.1 Boron Concentration (MODE 6)

The boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained ≥ 2400 ppm. This value includes a 50 ppm conservative allowance for uncertainties.

5.1 Core Operating Limits Report

5.1.12 References

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (Westinghouse Proprietary).

2. WCAP-8745-P-A, "Design Bases for the Thermal Overtemperature ΔT and Thermal Overpower ΔT Trip Functions," September 1986.

3. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998 (Westinghouse Proprietary).

4. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control-F_Q Surveillance Technical Specification," February 1994.

5. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

6. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

7. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicating Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," April 1999.

8. Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997.

9. Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ System," Revision 0, May 2000.

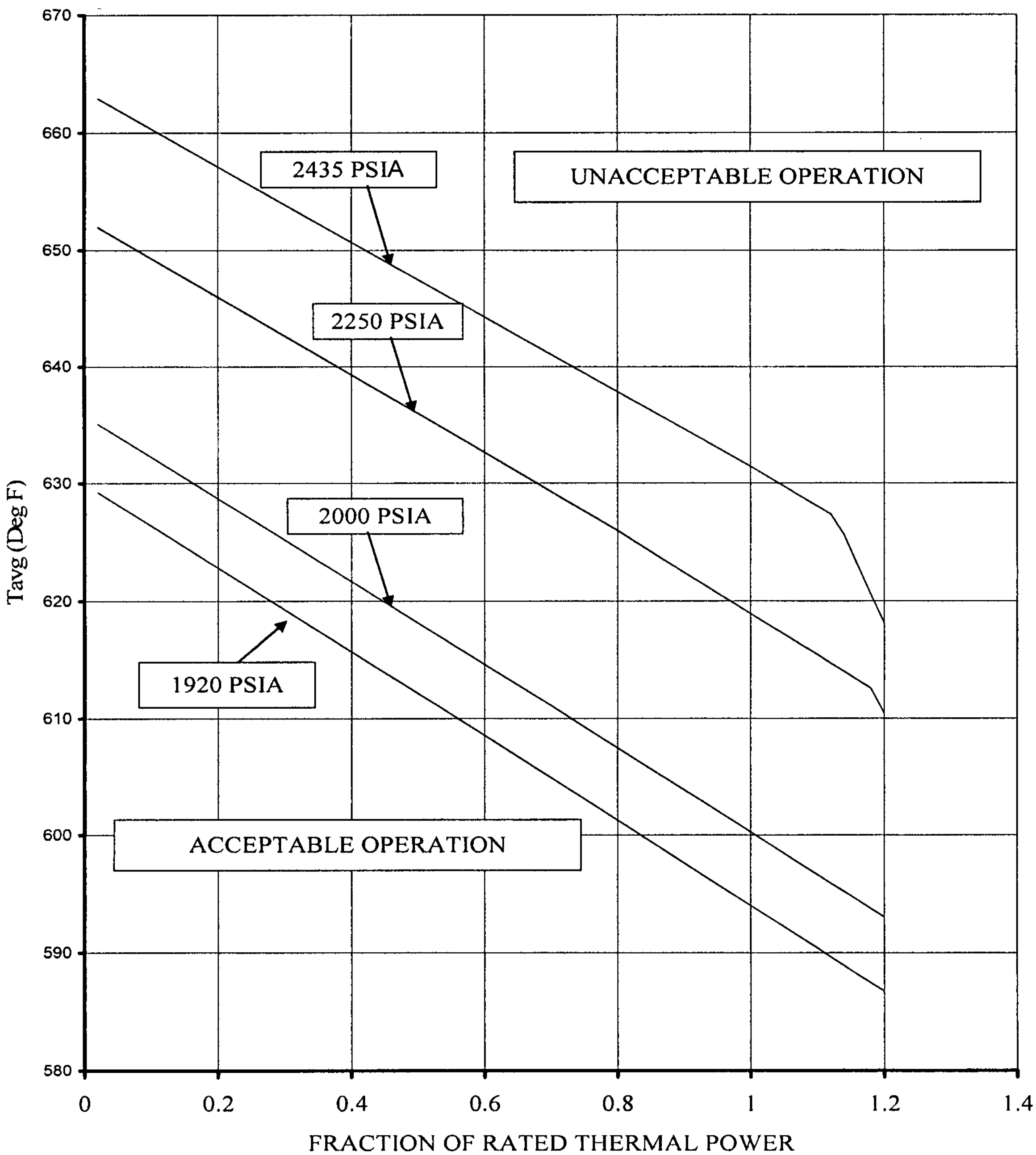

Figure 5.1-1 (Page 1 of 1)

REACTOR CORE SAFETY LIMIT
THREE LOOP OPERATION

(Technical Specification Safety Limit 2.1.1)

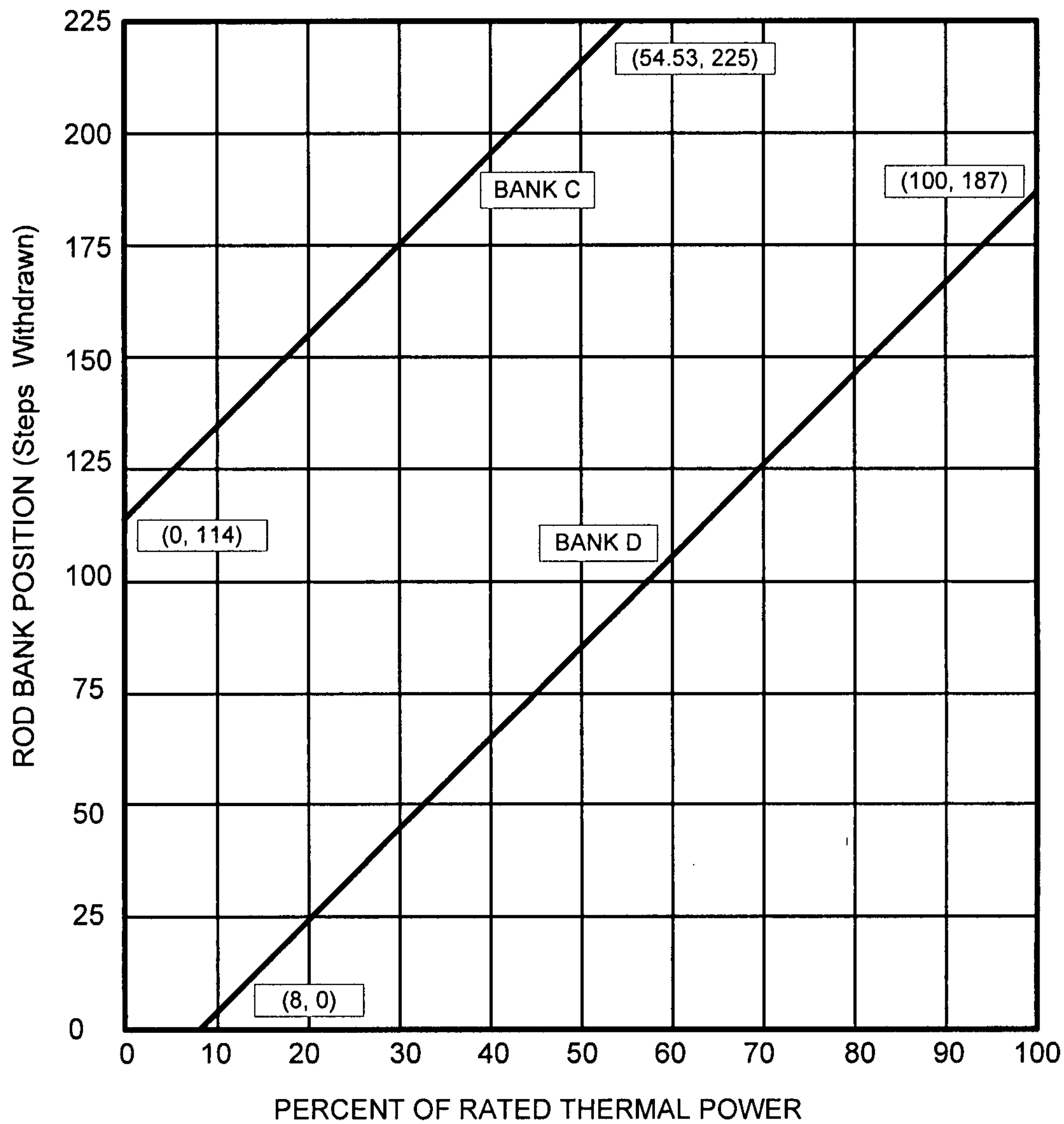

Figure 5.1-2 (Page 1 of 1)

CONTROL ROD INSERTION LIMITS AS A

FUNCTION OF RATED POWER LEVEL

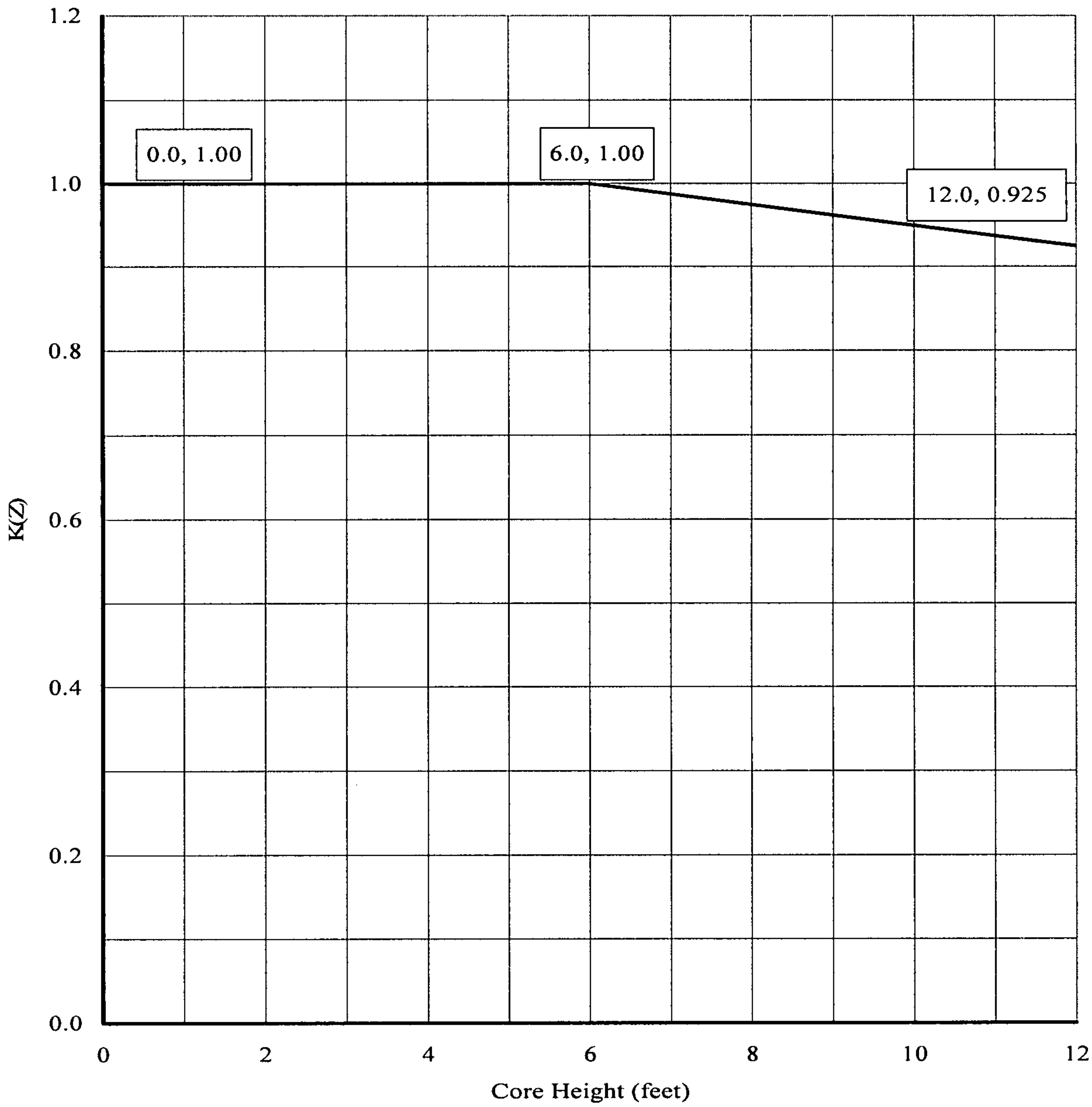

Figure 5.1-3 (Page 1 of 1)

F_QT NORMALIZED OPERATING ENVELOPE, K(Z)

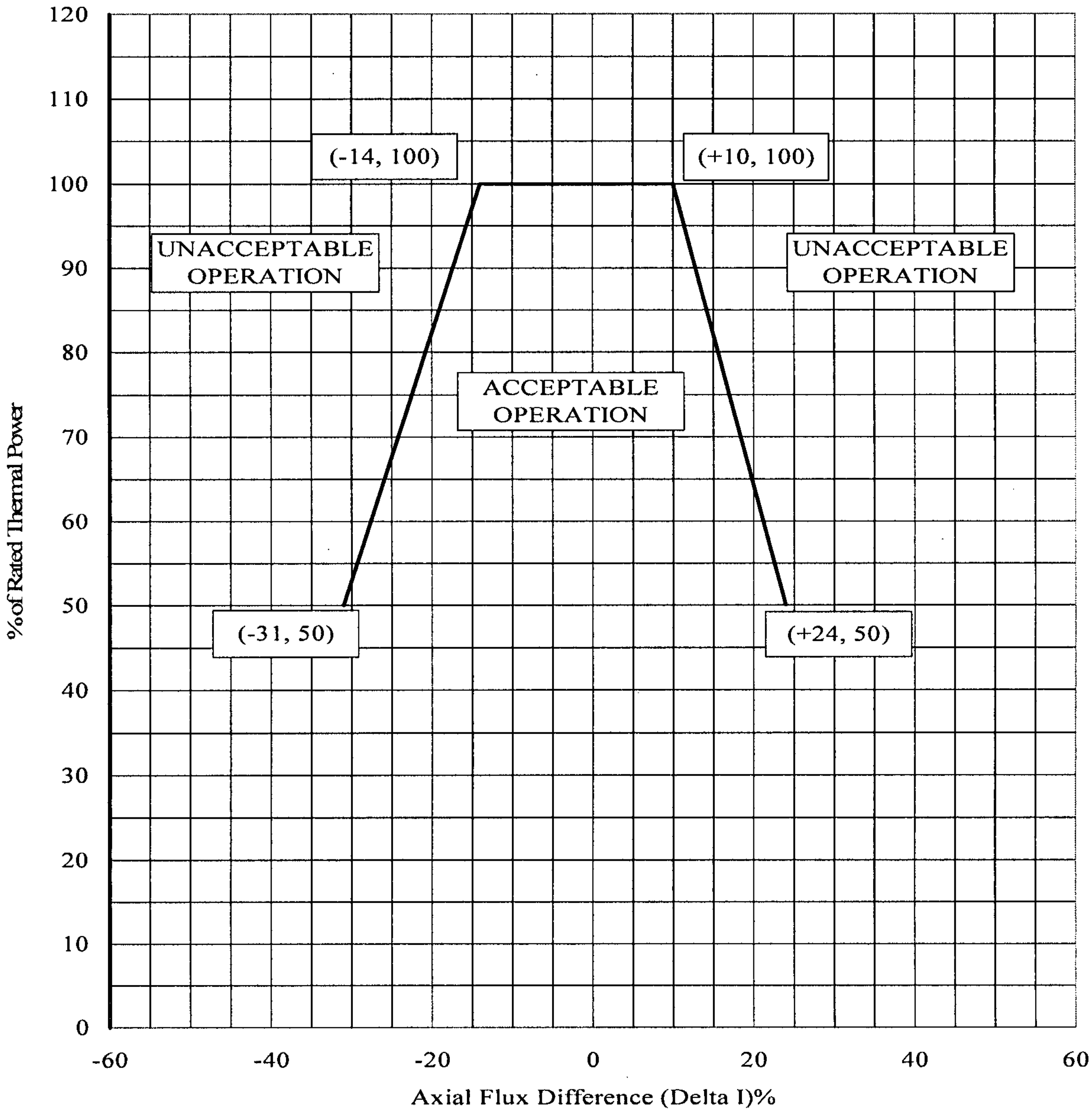

Figure 5.1-4 (Page 1 of 1)

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF

PERCENT OF RATED THERMAL POWER FOR RAOC

Table 5.1-1 (Page 1 of 2)
F_Q Surveillance W(Z) Function versus Burnup

Exclusion Zone	Axial Point	Elevation (feet)	150 MWD/MTU	3000 MWD/MTU	10000 MWD/MTU	16000 MWD/MTU
*	1	12.00	1.0000	1.0000	1.0000	1.0000
*	2	11.80	1.0000	1.0000	1.0000	1.0000
*	3	11.60	1.0000	1.0000	1.0000	1.0000
*	4	11.40	1.0000	1.0000	1.0000	1.0000
*	5	11.20	1.0000	1.0000	1.0000	1.0000
*	6	11.00	1.0000	1.0000	1.0000	1.0000
*	7	10.80	1.0000	1.0000	1.0000	1.0000
	8	10.60	1.2425	1.3243	1.2957	1.2379
	9	10.40	1.2294	1.3089	1.2830	1.2268
	10	10.20	1.2157	1.2912	1.2681	1.2233
	11	10.00	1.2036	1.2737	1.2535	1.2178
	12	9.80	1.1934	1.2603	1.2420	1.2115
	13	9.60	1.1827	1.2557	1.2312	1.2033
	14	9.40	1.1730	1.2534	1.2205	1.1992
	15	9.20	1.1706	1.2464	1.2083	1.1985
	16	9.00	1.1782	1.2374	1.1962	1.2000
	17	8.80	1.1847	1.2348	1.1974	1.2021
	18	8.60	1.1913	1.2342	1.2073	1.2098
	19	8.40	1.1978	1.2317	1.2176	1.2209
	20	8.20	1.2017	1.2267	1.2245	1.2300
	21	8.00	1.2037	1.2194	1.2290	1.2364
	22	7.80	1.2039	1.2152	1.2311	1.2402
	23	7.60	1.2020	1.2110	1.2307	1.2416
	24	7.40	1.1986	1.2045	1.2280	1.2405
	25	7.20	1.1946	1.1972	1.2231	1.2372
	26	7.00	1.1908	1.1905	1.2162	1.2316
	27	6.80	1.1856	1.1825	1.2074	1.2240
	28	6.60	1.1787	1.1728	1.1968	1.2147
	29	6.40	1.1704	1.1619	1.1847	1.2033
	30	6.20	1.1607	1.1498	1.1712	1.1903
	31	6.00	1.1499	1.1362	1.1564	1.1805
	32	5.80	1.1380	1.1237	1.1408	1.1712

Note: Top and Bottom 10% Excluded

TABLE 5.1-1 (Page 2 of 2)
F_Q Surveillance W(Z) Function versus Burnup

Exclusion Zone	Axial Point	Elevation (feet)	150 MWD/MTU	3000 MWD/MTU	10000 MWD/MTU	16000 MWD/MTU
	33	5.60	1.1301	1.1129	1.1274	1.1618
	34	5.40	1.1282	1.1062	1.1201	1.1561
	35	5.20	1.1340	1.1099	1.1188	1.1565
	36	5.00	1.1402	1.1145	1.1198	1.1563
	37	4.80	1.1451	1.1176	1.1218	1.1561
	38	4.60	1.1497	1.1206	1.1229	1.1575
	39	4.40	1.1538	1.1232	1.1237	1.1584
	40	4.20	1.1568	1.1260	1.1242	1.1584
	41	4.00	1.1625	1.1303	1.1241	1.1576
	42	3.80	1.1715	1.1353	1.1252	1.1559
	43	3.60	1.1816	1.1395	1.1310	1.1539
	44	3.40	1.1910	1.1435	1.1360	1.1516
	45	3.20	1.1988	1.1471	1.1387	1.1483
	46	3.00	1.2112	1.1579	1.1485	1.1511
	47	2.80	1.2315	1.1786	1.1675	1.1675
	48	2.60	1.2560	1.2033	1.1874	1.1862
	49	2.40	1.2801	1.2281	1.2074	1.2047
	50	2.20	1.3037	1.2527	1.2271	1.2229
	51	2.00	1.3268	1.2769	1.2462	1.2406
	52	1.80	1.3488	1.3001	1.2647	1.2576
	53	1.60	1.3694	1.3220	1.2819	1.2736
	54	1.40	1.3878	1.3418	1.2975	1.2883
*	55	1.20	1.0000	1.0000	1.0000	1.0000
*	56	1.00	1.0000	1.0000	1.0000	1.0000
*	57	0.80	1.0000	1.0000	1.0000	1.0000
*	58	0.60	1.0000	1.0000	1.0000	1.0000
*	59	0.40	1.0000	1.0000	1.0000	1.0000
*	60	0.20	1.0000	1.0000	1.0000	1.0000
*	61	0.00	1.0000	1.0000	1.0000	1.0000

Note: Top and Bottom 10% Excluded

Table 5.1-2 (Page 1 of 1)
$F_Q(Z)$ Penalty Factor versus Burnup

Cycle Burnup (MWD/MTU)	$F_Q(Z)$ Penalty Factor
0 to 4300	1.02
4300 to 5000	1.0205
5000 to EOL	1.02

Note: The Penalty Factor, to be applied to $F_Q(Z)$ in accordance with Technical Specification Surveillance Requirement (SR) 3.2.1.2, is the maximum factor by which $F_Q(Z)$ is expected to increase over a 39 Effective Full Power Day (EFPD) interval (surveillance interval of 31 EFPD plus the maximum allowable extension not to exceed 25% of the surveillance interval per Technical Specification SR 3.0.2) starting from the burnup at which the $F_Q(Z)$ was determined.

Enclosure 4

Beaver Valley Power Station

Unit No. 2

Pressure and Temperature Limits Report

Revision 2

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.

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 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 Pressure and Temperature Limits Report

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.

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 "ARM" position. Once OPPS is activated (armed/enabled) reactor coolant system pressure transmitters will signal a rise in system pressure above the variable OPPS setpoint. This will initiate an alarm in the control room and open the OPPS PORVs.

5.2.1.4 Reactor Vessel Boltup Temperature (LCO 3.4.3)

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

5.2 Pressure and Temperature Limits Report

5.2.2 Reactor Vessel Material Surveillance Program

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

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 4-9 of Reference 2, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 5.2-6, taken from Tables 4-8 and 4-10 of Reference 2, provides the reactor vessel beltline material property table.

Table 5.2-7, taken from Tables 4-15 and 4-16 of Reference 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 22 EFPY.

Table 5.2-8, taken from Tables 4-15 and 4-16 of Reference 2, shows the calculation of ARTs for 22 EFPY.

Table 5.2-9 shows the Reactor Vessel Toughness Data (Unirradiated).

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

<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

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

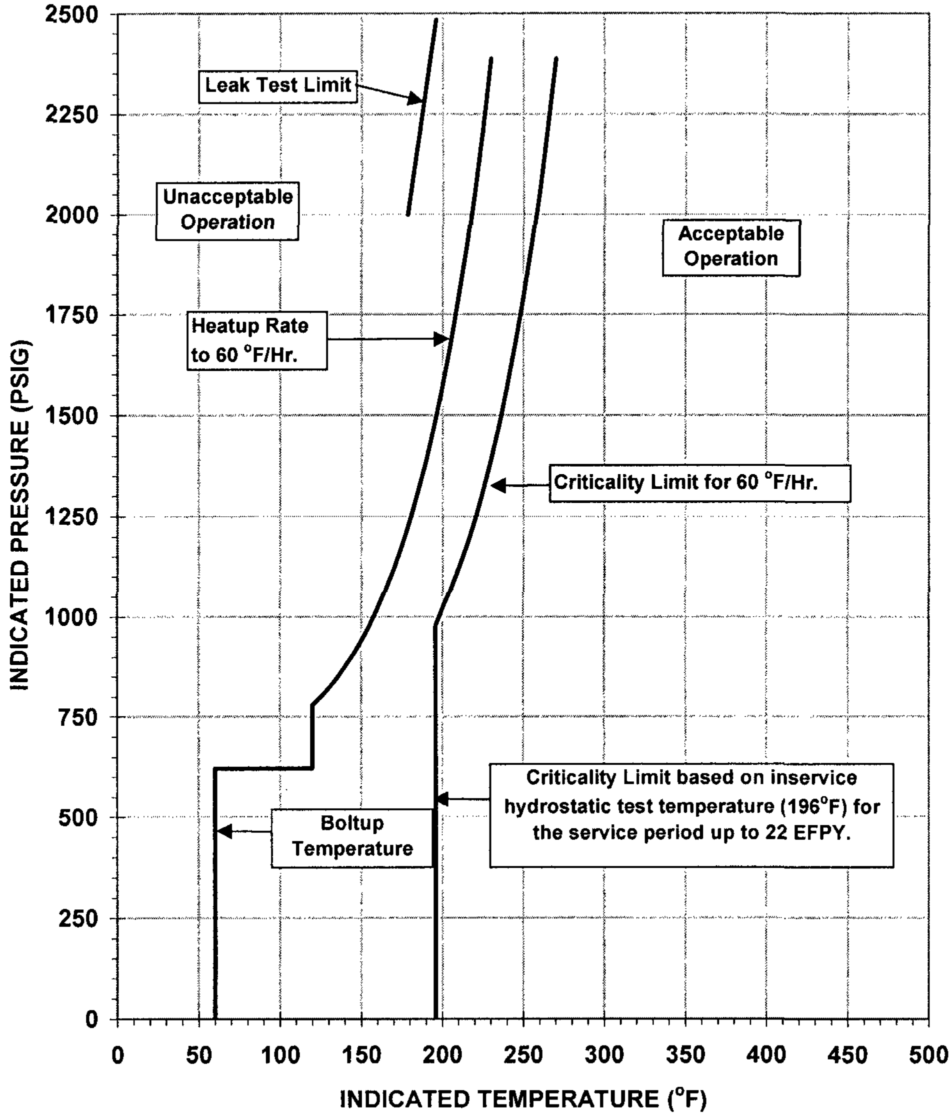

Figure 5.2-1 (Page 1 of 1)
Reactor Coolant System Heatup
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

<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 0°F/HR FOR THE SERVICE PERIOD UP TO 22 EFPY.

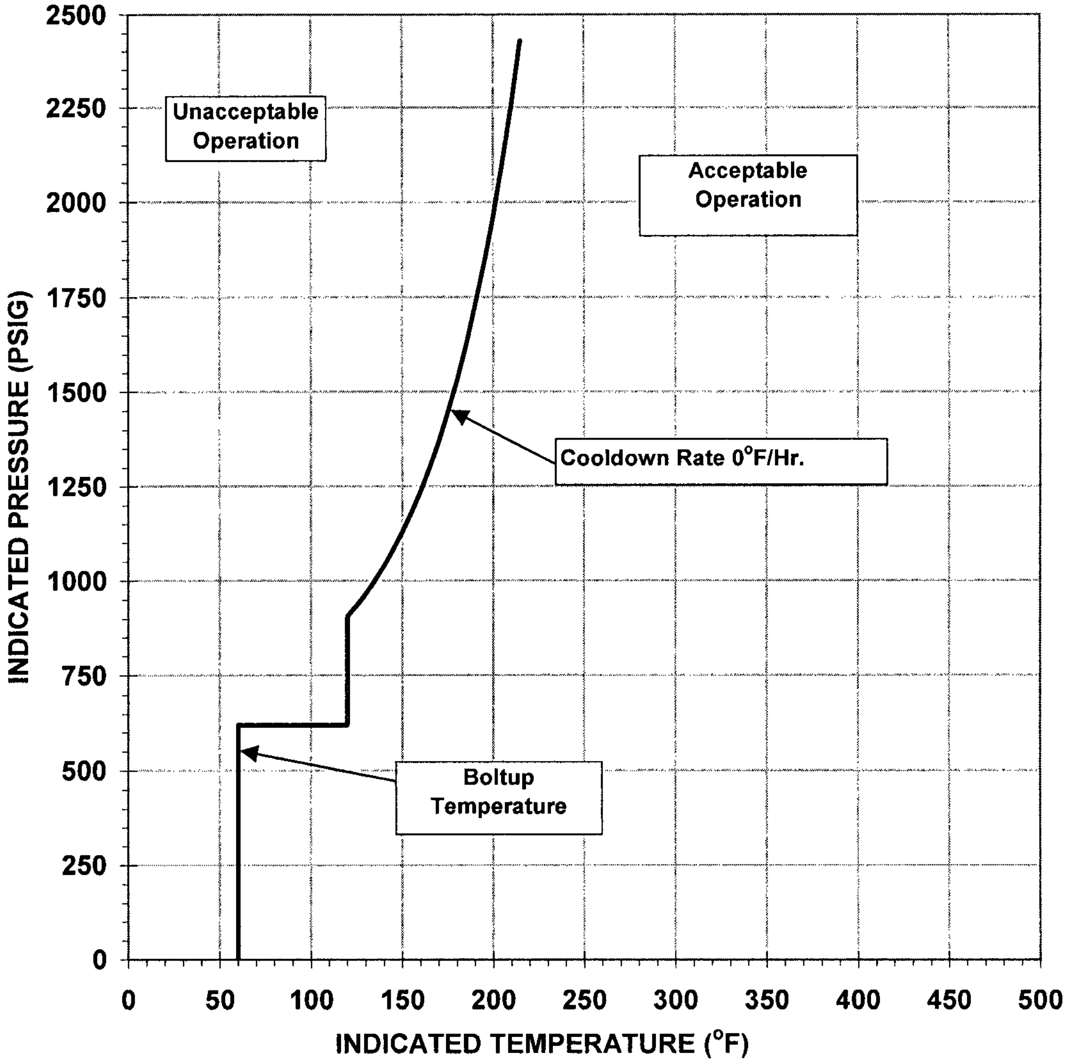

Figure 5.2-2 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 0°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

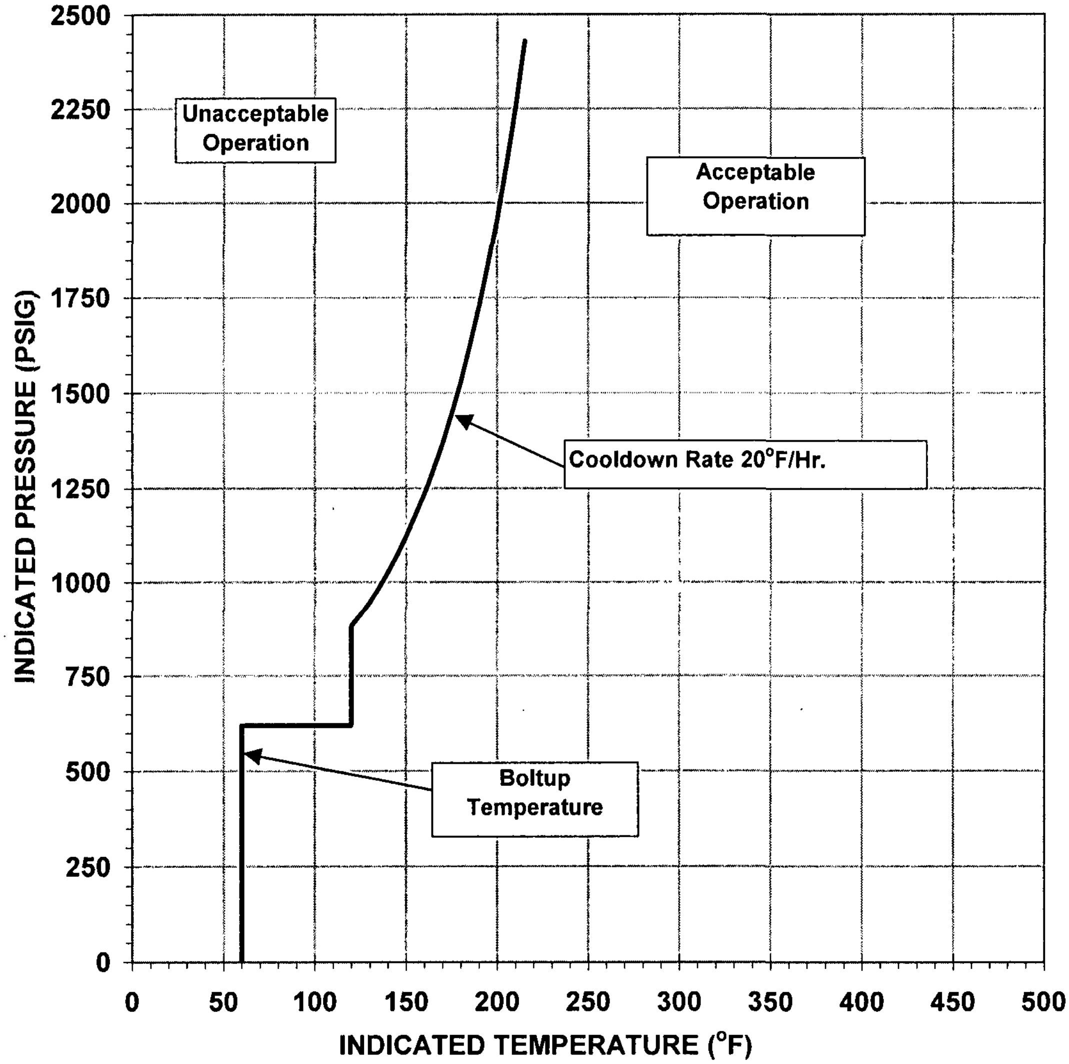

Figure 5.2-3 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 20°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

<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.

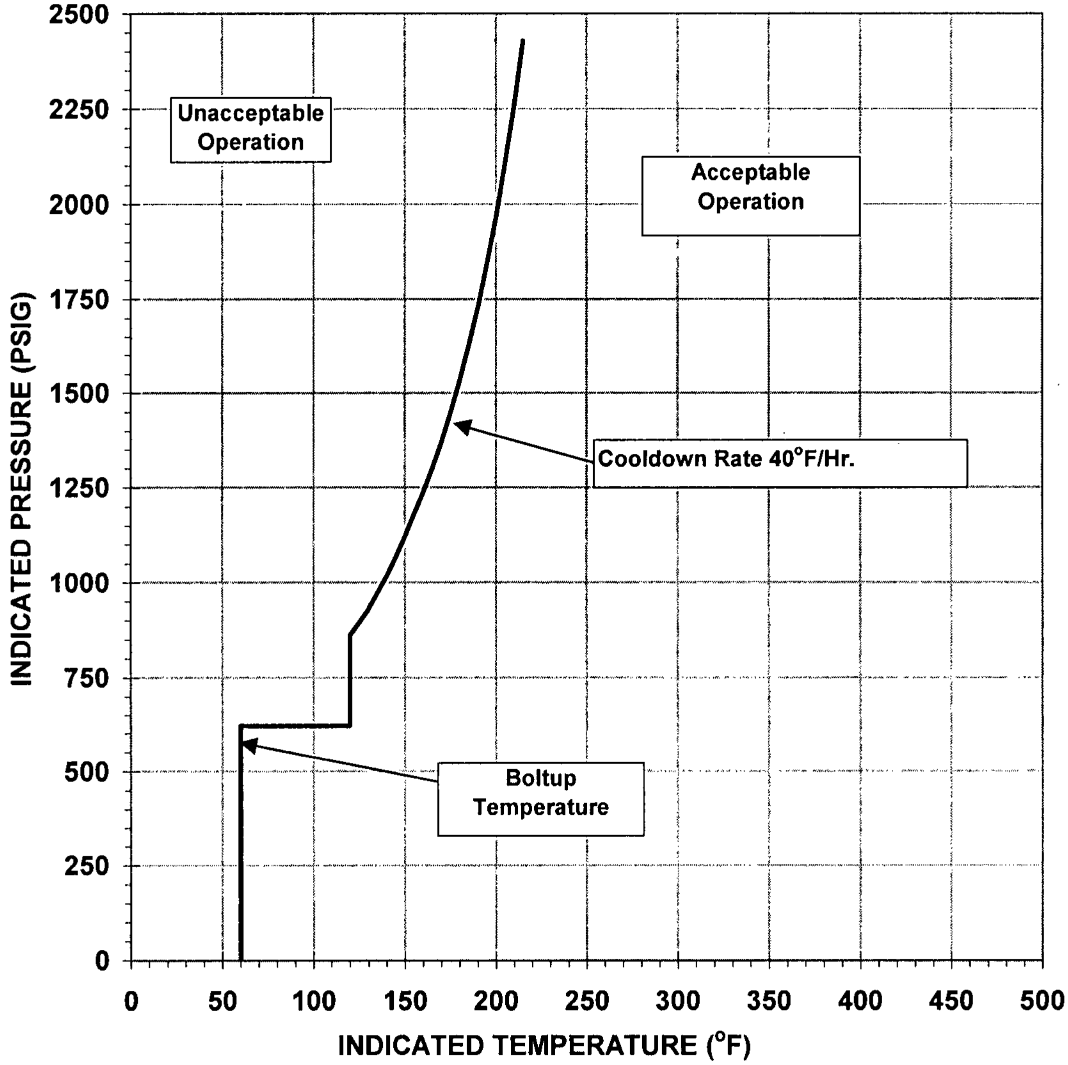

Figure 5.2-4 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 40°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

<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.

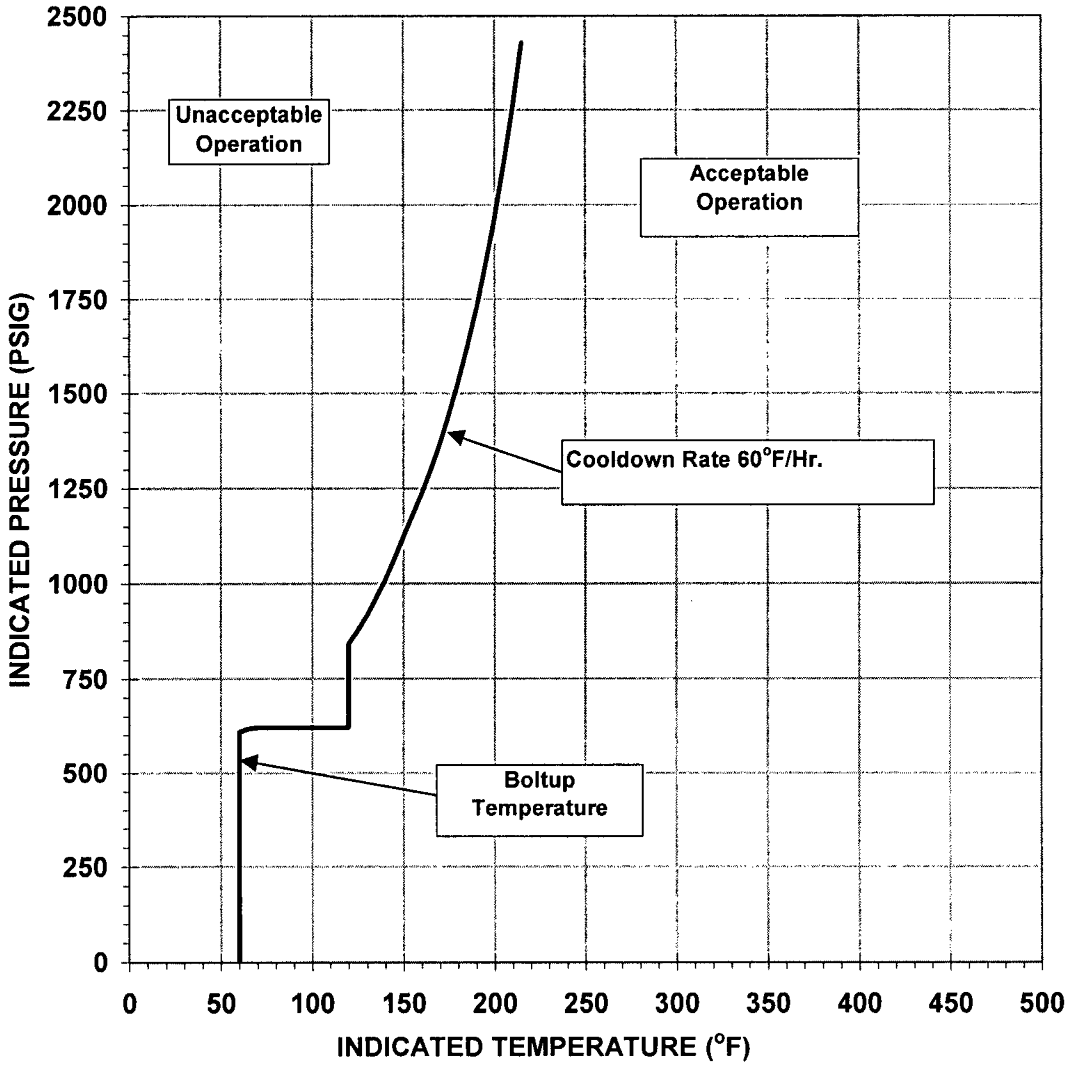

Figure 5.2-5 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 60°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

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.

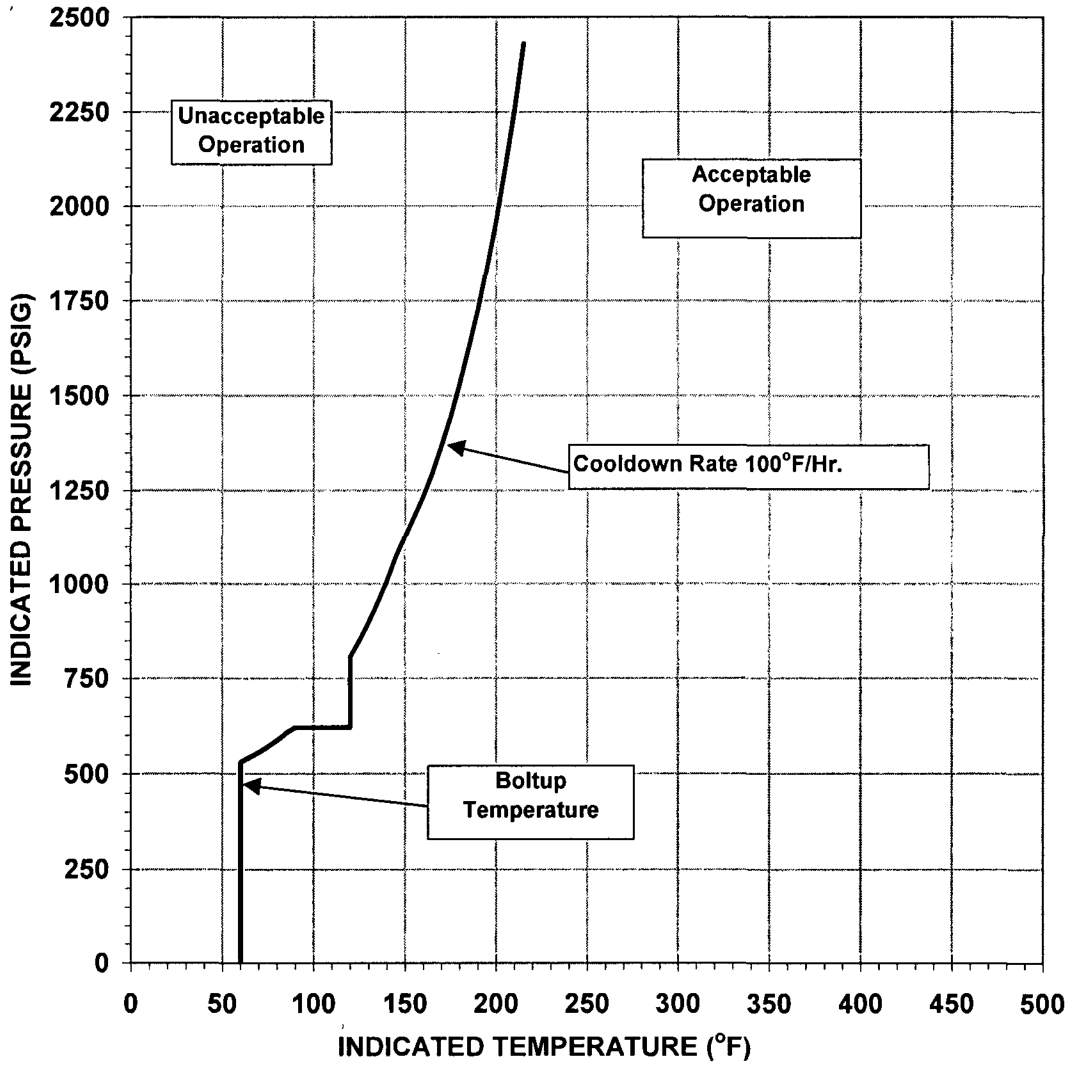

Figure 5.2-6 (Page 1 of 1)
Reactor Coolant System Cooldown (up to 100°F/Hr.)
Limitations Applicable for the First 22 EFPY (LCO 3.4.3)

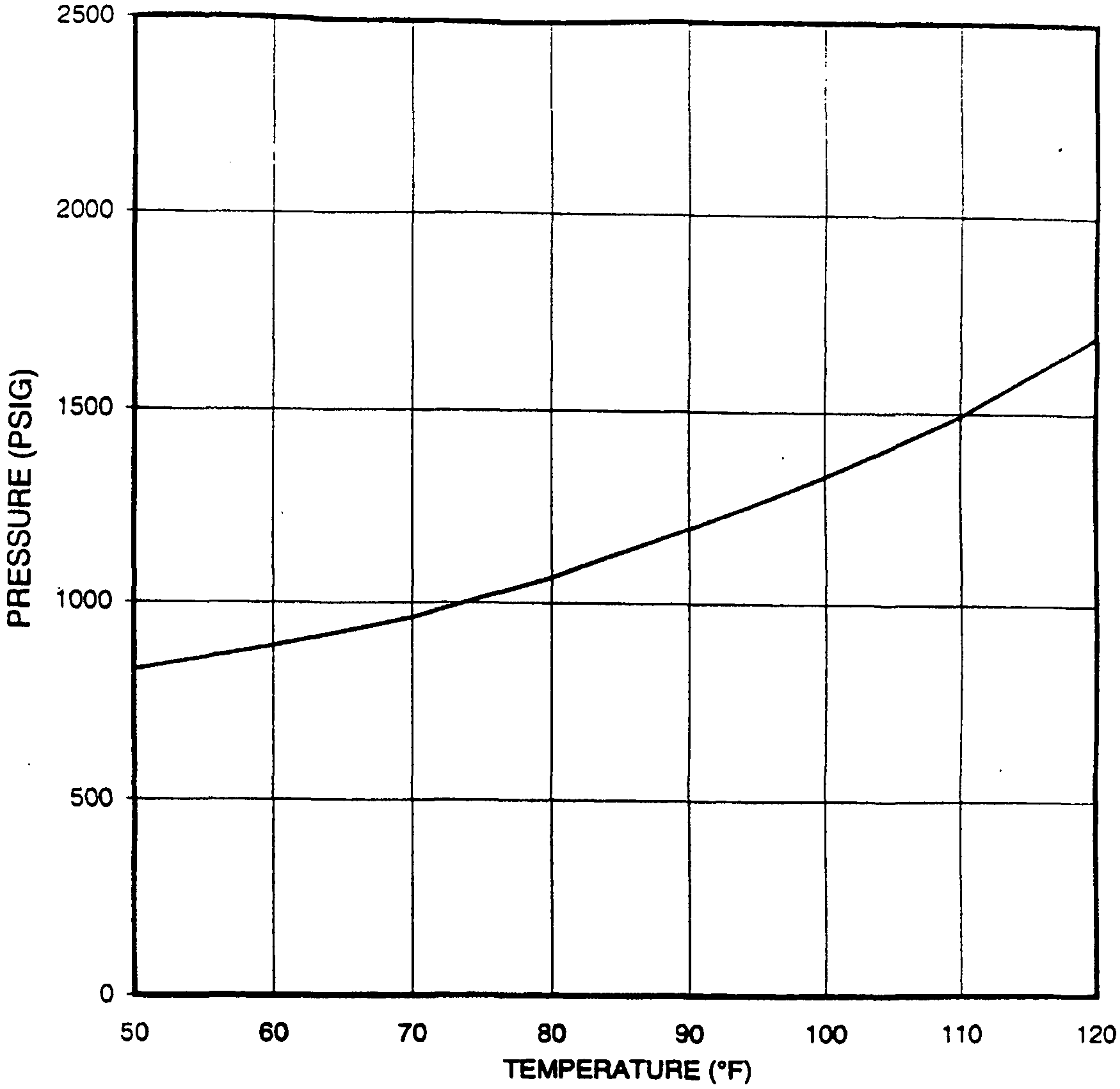

Figure 5.2-7 (Page 1 of 1)
Isolated Loop Pressure – Temperature Limit Curve (LCO 3.4.3)

See Table 5.2-4 for RCP restrictions.

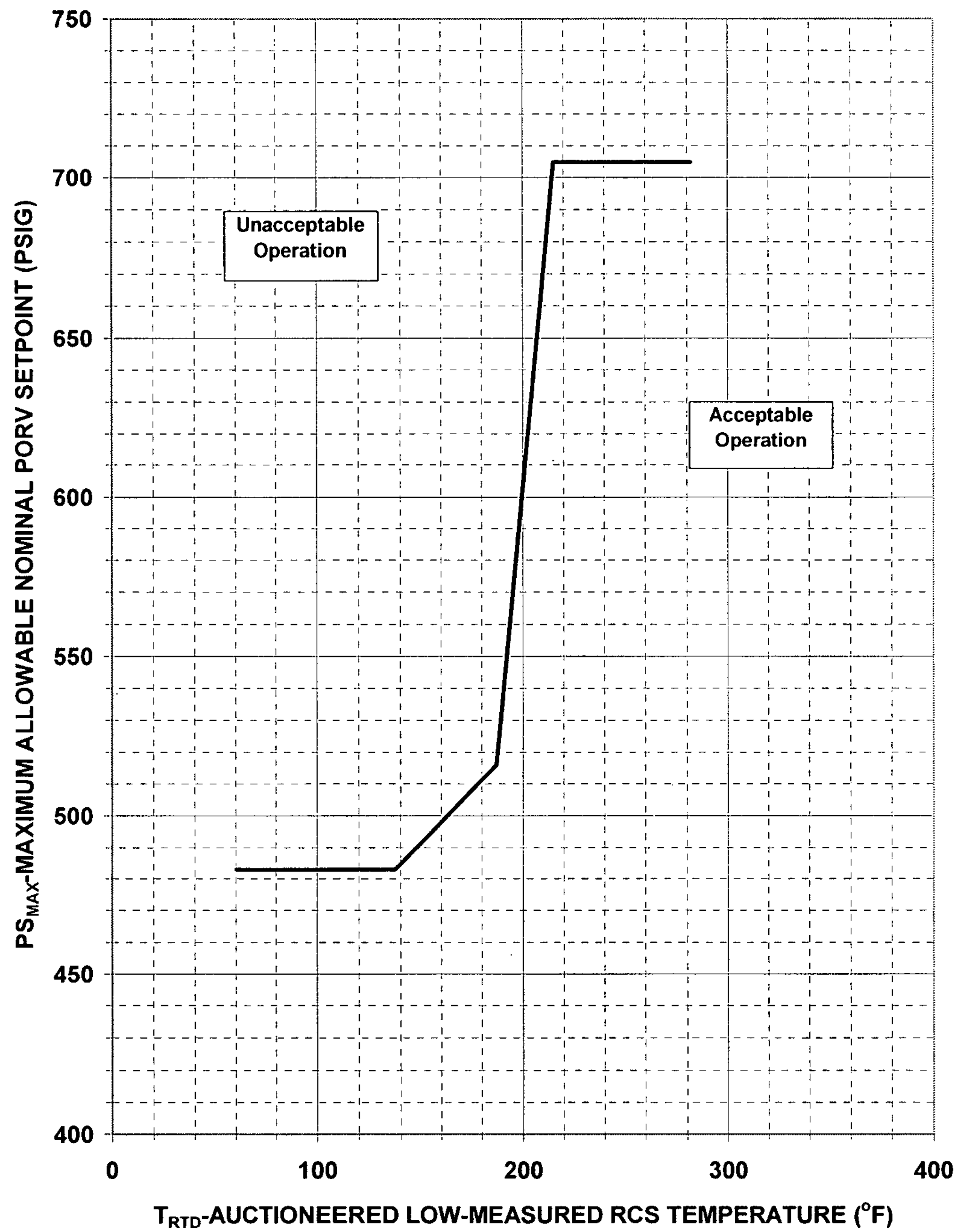

Figure 5.2-8 (Page 1 of 1)
Maximum Allowable Nominal PORV Setpoint for the
Overpressure Protection System (LCO 3.4.12)

Table 5.2-1 (Page 1 of 1)
Heatup Curve Data Points for 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				

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

Table 5.2-3 (Page 1 of 1)

Overpressure Protection System (OPPS) Setpoints (LCO 3.4.12)

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

Table 5.2-4 (Page 1 of 1)

Reactor Coolant Pump Restrictions

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

Table 5.2-5 (Page 1 of 1)
Calculation of Chemistry Factors Using Surveillance Capsule Data[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	0.86	24.26	20.86	0.74
	V	2.63	1.26	55.93	70.47	1.59
	W	3.625	1.335	71.04	94.83	1.78
Intermediate Shell Plate B9004-2 (Transverse)	U	0.608	0.86	17.56	15.10	0.74
	V	2.63	1.26	46.27	58.30	1.59
	W	3.625	1.335	63.39	84.63	1.78
				SUM:	344.19	8.22
					$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$ =	41.9
Weld Metal	U	0.608	0.86	3.64	3.13	0.74
	V	2.64	1.26	25.47	32.09	1.59
	W	3.625	1.335	6.21	8.29	1.78
				SUM:	43.51	4.11
					$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$ =	10.6

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 3).
(c) FF = fluence factor = $f^{(0.28 - 0.1 * \log f)}$.
(d) ΔRT_{NDT} values obtained from CVGRAPH Version 4.1.

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

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	0.74	N/A	N/A	-10
Vessel Flange	N/A	0.73	N/A	N/A	0
Intermediate Shell Plate B9004-1	Position 1.1	0.065	0.55	40.5	60
Intermediate Shell Plate B9004-2	Position 1.1	0.06	0.57	37.0	40
	Position 2.1	N/A	N/A	41.9	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 (Longitudinal & Circumferential Seams)	Position 1.1	0.046	0.086	34.4	-30
	Position 2.1	N/A	N/A	10.6	-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.

Table 5.2-7 (Page 1 of 1)

Summary of Adjusted Reference Temperature (ARTs) for 22 EFPY

MATERIAL DESCRIPTION	Method Used To Calculate the CF[(a)]	22 EFPY ART	
		1/4T ART(°F)	3/4T ART(°F)
Intermediate Shell Plate B9004-1	Position 1.1	140	129
Intermediate Shell Plate B9004-2	Position 1.1	116	106
	Position 2.1	104	94
Lower Shell Plate B9005-1	Position 1.1	120	106
Lower Shell Plate B9005-2	Position 1.1	117	105
Vessel Beltline Welds (b)	Position 1.1	48	30
	Position 2.1	-6	-12

Notes:

(a) Regulatory Guide 1.99, Revision 2.

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

Table 5.2-8 (Page 1 of 1)

Calculation of Adjusted Reference Temperatures (ARTs) for 22 EFPY

PARAMETER	VALUES	
Operating Time	22 EFPY	
Material - Intermediate Shell Plate	B9004-1	B9004-1
Location	1/4T ART	3/4T ART
Chemistry Factor, CF (°F)	40.5	40.5
Fluence, (f), (10^{19} n/cm^2)[(a)]	1.63	0.632
Fluence Factor, FF	1.13	0.87
ΔRT_{NDT} = CF x FF(°F)	45.8	35.2
Initial RT_{NDT}, I(°F)	60	60
Margin, M(°F)	34	34
ART, per Regulatory Guide 1.99, Revision 2	140	129

Notes:

(a) Fluence (f), is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 1.81 at 22 EFPY.

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

Table 5.2-9 (Page 1 of 1)
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 5.2.1.1 for a discussion of EFPY.

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.