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August 29, 2006  
L-06-131

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

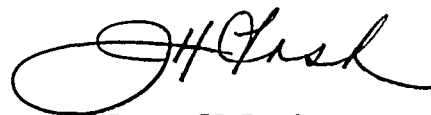
**Subject: Beaver Valley Power Station, Unit No. 1  
Docket No. 50-334, License No. DPR-66  
Core Operating Limits Report, COLR 18-1  
Pressure and Temperature Limits Report, Revision 2**

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 as required by section 6.9 of the BVPS Technical Specifications. COLR 18-1 is included as Enclosure 1 and Revision 2 of the PTLR is included as Enclosure 2.

The revisions to the COLR and the PTLR are associated with the implementation of License Amendment No. 275 and a mid-cycle 3% power uprate, effective August 11, 2006.

No regulatory commitments are contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager - FENOC Fleet Licensing, at (330) 315-7243.

Sincerely,



James H. Lash

A001

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

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Enclosure:

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

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

**Enclosure 1**

**Beaver Valley Power Station**

**Unit No.1**

**Core Operating Limits Report**

**COLR 18-1**

## LICENSING REQUIREMENTS MANUAL

## 4.1 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 6.9.5.

Specification 3.1.3.5 Shutdown Rod Insertion Limits

The shutdown rods shall be withdrawn to at least 225 steps.\*

Specification 3.1.3.6 Control Rod Insertion Limits

Control Banks A and B shall be withdrawn to at least 225 steps.\*

Control Banks C and D shall be limited in physical insertion as shown in Figure 4.1-1.\*

Specification 3.2.1 Axial Flux Difference

The Axial Flux Difference (AFD) acceptable operation limits are provided in Figure 4.1-2.

Specification 3.2.2 Heat Flux Hot Channel Factor -  $F_Q(Z)$  Limits

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

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

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

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

$K(Z)$  = the function obtained from Figure 4.1-3.

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

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

The  $W(Z)$  values are provided in Table 4.1-1.

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 4.1-2.

\* As indicated by the group demand counter

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### LICENSING REQUIREMENTS MANUAL

#### Specification 3.2.3 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:  $CF_{\Delta H} = 1.62$

$PF_{\Delta H} = 0.3$

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

## LICENSING REQUIREMENTS MANUAL

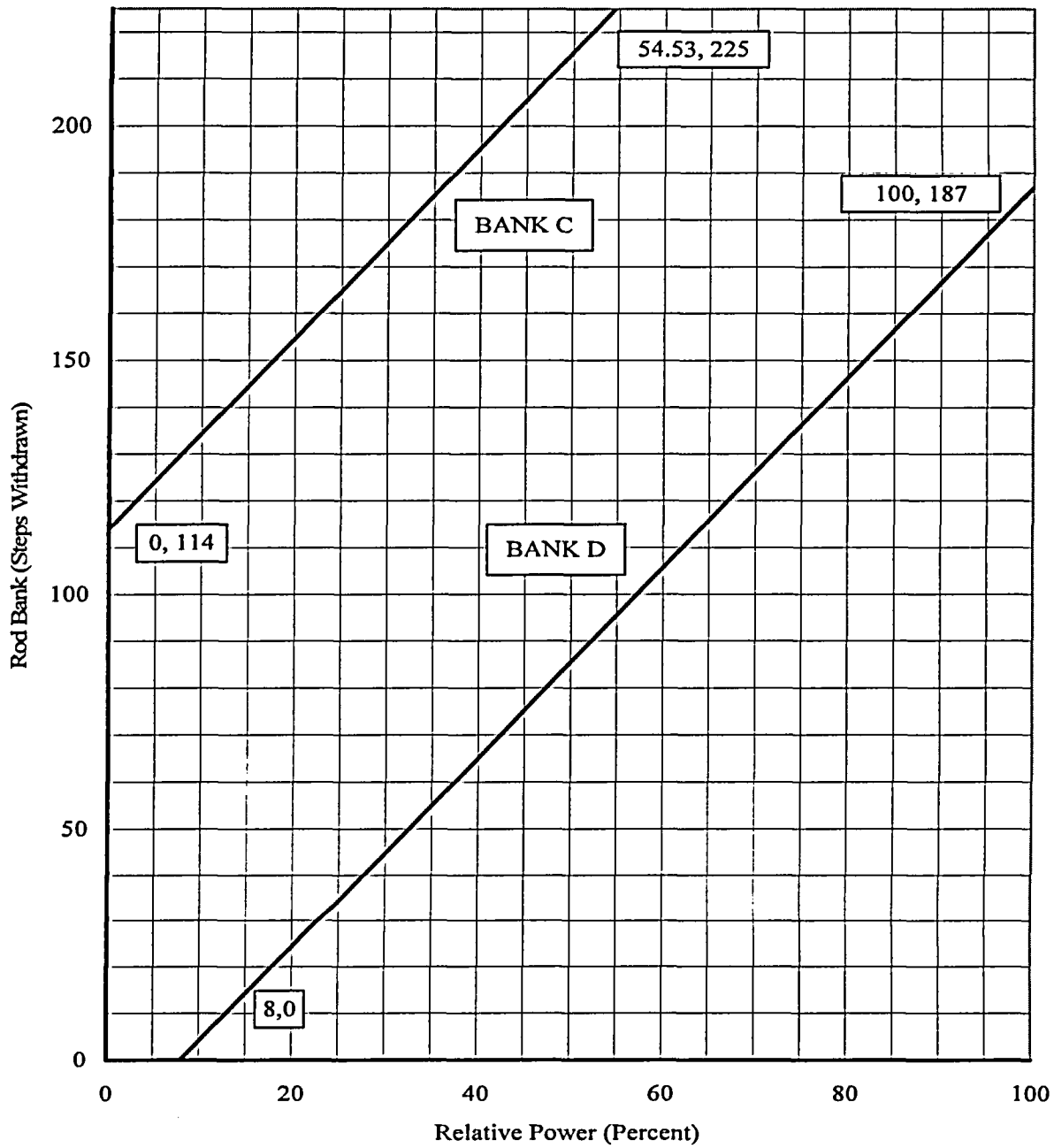
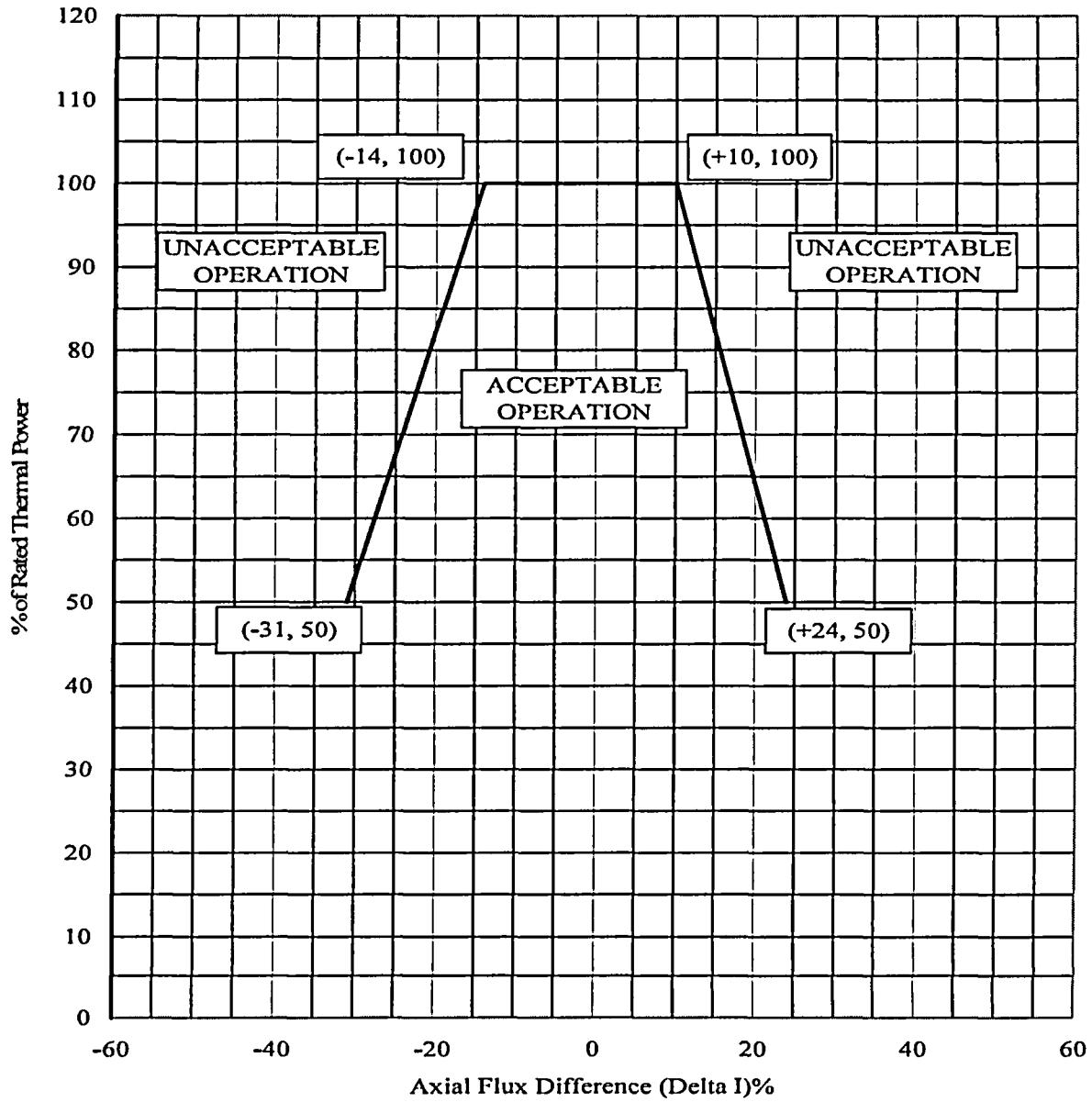


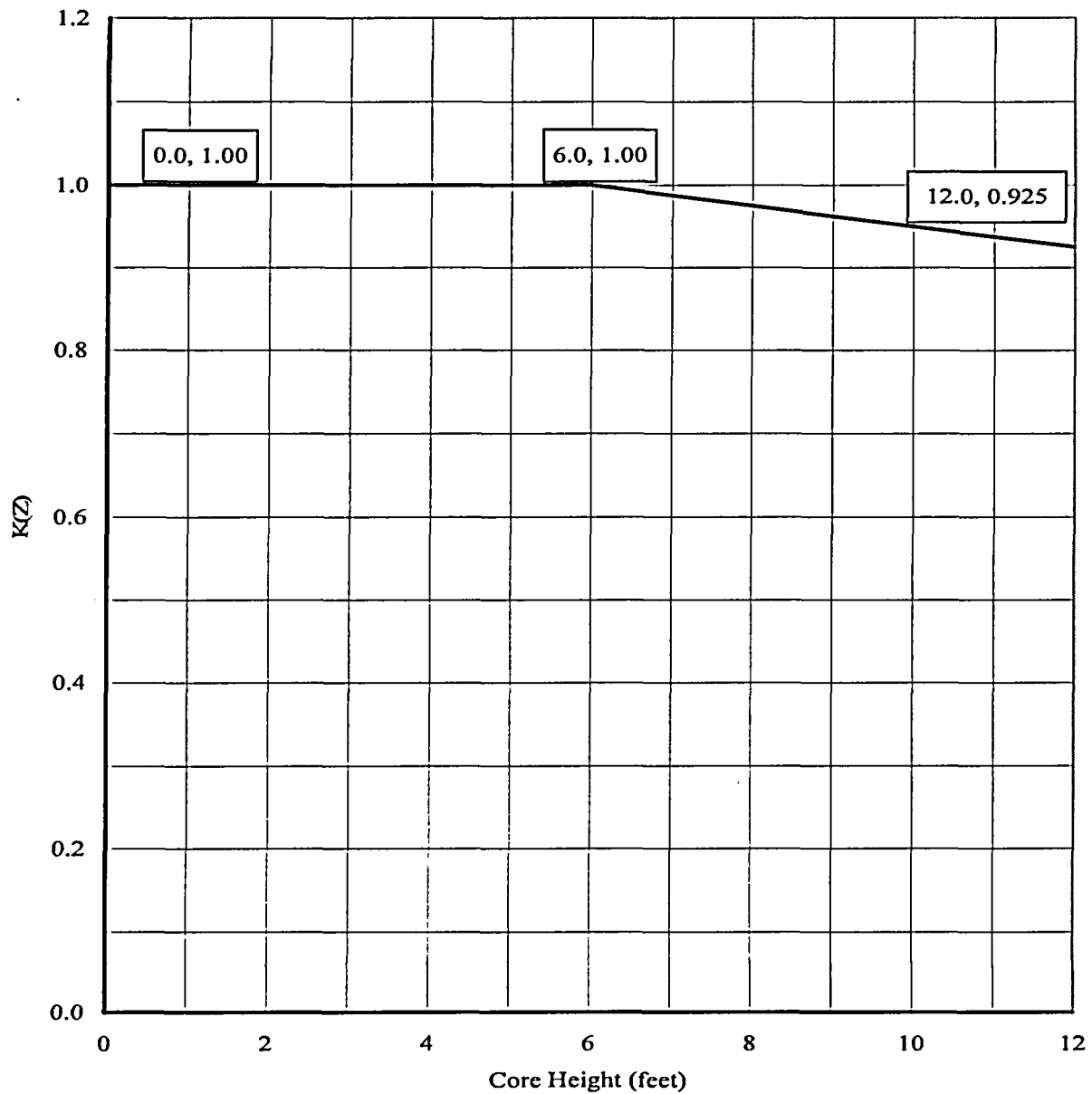
FIGURE 4.1-1  
CONTROL ROD INSERTION LIMITS

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**FIGURE 4.1-2  
AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF  
PERCENT OF RATED THERMAL POWER FOR RAOC**

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LICENSING REQUIREMENTS MANUAL



**FIGURE 4.1-3**  
**F<sub>0</sub>T NORMALIZED OPERATING ENVELOPE,  $K(Z)$**



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Specification 3.3.1.1 Reactor Trip System Instrumentation Setpoints, Table 3.3-1 Table Notations A and B

Overtemperature  $\Delta T$  Setpoint Parameter Values:

<u>Parameter</u>	<u>Value</u>
Overtemperature $\Delta T$ reactor trip setpoint	$K1 \leq 1.242$
Overtemperature $\Delta T$ reactor trip setpoint $T_{avg}$ coefficient	$K2 \geq 0.0183/^{\circ}F$
Overtemperature $\Delta T$ reactor trip setpoint pressure coefficient	$K3 \geq 0.001/psia$
$T_{avg}$ at RATED THERMAL POWER	$T' \leq 580.0^{\circ}F^{(1)}$
Nominal Pressurizer Pressure	$P' \geq 2250 \text{ psia}$
Measured reactor vessel average temperature lead/lag time constants	$\tau_1 \geq 30 \text{ secs}$ $\tau_2 \leq 4 \text{ secs}$
Measured reactor vessel $\Delta T$ lag time constant	$\tau_4 \leq 6 \text{ secs}$
Measured reactor vessel average temperature lag time constant	$\tau_5 \leq 2 \text{ secs}$

$f(\Delta I)$  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 percent and +15 percent,  $f(\Delta I) = 0$  (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).
- (ii) for each percent that the magnitude of  $(q_t - q_b)$  exceeds -37 percent, the  $\Delta T$  trip setpoint shall be automatically reduced by 2.52 percent of its value at RATED THERMAL POWER.
- (iii) for each percent that the magnitude of  $(q_t - q_b)$  exceeds +15 percent, the  $\Delta T$  trip setpoint shall be automatically reduced by 1.47 percent of its value at RATED THERMAL POWER.

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(1)  $T'$  represents the cycle-specific Full Power  $T_{avg}$  value used in core design (576.2°F for 2770 MWt and 577.9°F for 2900 MWt).

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## Overpower $\Delta T$ Setpoint Parameter Values:

<u>Parameter</u>	<u>Value</u>
Overpower $\Delta T$ reactor trip setpoint	$K4 \leq 1.085$
Overpower $\Delta T$ reactor trip setpoint $T_{avg}$ rate/lag coefficient	$K5 \geq 0.02/^{\circ}\text{F}$ for increasing average temperature $K5 = 0/^{\circ}\text{F}$ for decreasing average temperature
Overpower $\Delta T$ reactor trip setpoint $T_{avg}$ heatup coefficient	$K6 \geq 0.0021/^{\circ}\text{F}$ for $T > T''$ $K6 = 0/^{\circ}\text{F}$ for $T \leq T''$
$T_{avg}$ at RATED THERMAL POWER	$T'' \leq 580.0^{\circ}\text{F}^{(1)}$
Measured reactor vessel average temperature rate/lag time constant	$\tau_3 \geq 10$ secs
Measured reactor vessel $\Delta T$ lag time constant	$\tau_4 \leq 6$ secs
Measured reactor vessel average temperature lag time constant	$\tau_5 \leq 2$ secs

## Specification 3.2.5 DNB Parameters

<u>Parameter</u>	<u>Indicated Value</u>
Reactor Coolant System $T_{avg}$	$T_{avg} \leq 583.6^{\circ}\text{F}^{(2)}$
Pressurizer Pressure	Pressure $\geq 2218$ psia <sup>(3)</sup>
Reactor Coolant System Total Flow Rate	Flow $\geq 267,300$ gpm <sup>(4)</sup>

- (1)  $T''$  represents the cycle-specific Full Power  $T_{avg}$  value used in core design ( $576.2^{\circ}\text{F}$  for 2770 MWt and  $577.9^{\circ}\text{F}$  for 2900 MWt).
- (2) The Reactor Coolant System (RCS)  $T_{avg}$  value includes allowances for rod control operation and verification via control board indication.
- (3) The pressurizer pressure value includes allowances for pressurizer pressure control operation and verification via control board indication.
- (4) 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.

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LICENSING REQUIREMENTS MANUAL

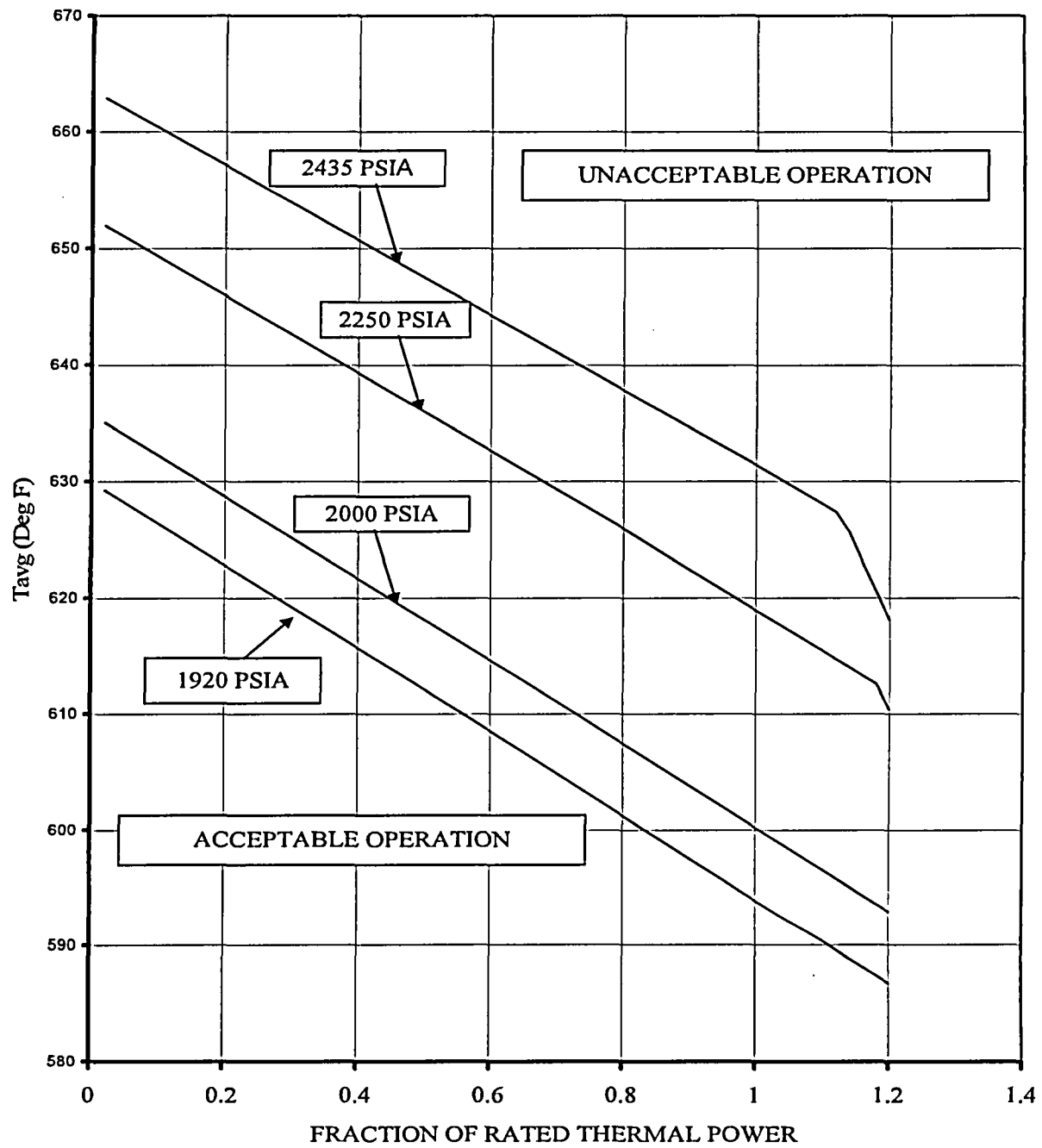


Figure 4.1-5  
REACTOR CORE SAFETY LIMIT  
THREE LOOP OPERATION  
(Technical Specification Safety Limit 2.1.1)

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Table 4.1-1 (Page 1 of 2)  
F<sub>Q</sub> 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

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Table 4.1-1 (Page 2 of 2)  
F<sub>Q</sub> 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

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Table 4.1-2  
 $F_Q(Z)$  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 4.2.2.3, 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 Surveillance Requirement 4.0.2) 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 2**

# BVPS-1

## LICENSING REQUIREMENTS MANUAL

### SECTION 4.2 PRESSURE AND TEMPERATURE LIMITS REPORT

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

BVPS-1 Licensing Requirement to PTLR Cross-Reference			
Licensing Requirement	PTLR		
	Section	Figure	Table
2.2	N/A	N/A	4.2-3
2.4	N/A	N/A	4.2-3



LICENSING REQUIREMENTS MANUAL

## PRESSURE AND TEMPERATURE LIMITS REPORT

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

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

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

- TS 3.4.1.3 Reactor Coolant System – Shutdown,
- TS 3.4.3 Reactor Coolant System – Safety Valve,
- TS 3.4.9.1 Reactor Coolant System - Pressure/Temperature Limits,
- TS 3.4.9.3 Overpressure Protection Systems,
- TS 3.5.2 ECCS Subsystems –  $T_{avg} \geq 350^{\circ}\text{F}$ ,
- TS 3.5.3 ECCS Subsystems –  $T_{avg} < 350^{\circ}\text{F}$ ,
- TS 3.5.4.1.2 Boron Injection Tank  $< 350^{\circ}\text{F}$ ,
- TS 3.10.3 Special Test Exceptions - Pressure/Temperature Limitations -Reactor Criticality,
- LR 2.2 Boration Flow Paths – Operating, and
- LR 2.4 Charging Pumps – Operating.

## 4.2.1 Operating Limits

The PTLR limits for Beaver Valley Power Station (BVPS) Unit 1 were developed using a methodology specified in the Technical Specifications. The methodology listed in Reference 1 was used with two exceptions:

- a) Use of ASME Code Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1”, and
- b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure”.

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

The RCS temperature rate-of-change limits defined in Reference 2 are:

- a. A maximum heatup of  $100^{\circ}\text{F}$  in any one hour period.
- b. A maximum cooldown of  $100^{\circ}\text{F}$  in any one hour period, and
- c. A maximum temperature change of less than or equal to  $5^{\circ}\text{F}$  in any one hour period during inservice hydrostatic testing operations above system design pressure.

The RCS P/T limits for heatup, leak testing, and criticality are specified by Figure 4.2-1 and Table 4.2-1. The RCS P/T limits for cooldown are shown in Figure 4.2-2 and Table 4.2-2. These limits are defined in Reference 2. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 4.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 4.2-1 and 4.2-2 are provided without margins for instrument error. The criticality limit curve specifies pressure-temperature limits for core operation to

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## PRESSURE AND TEMPERATURE LIMITS REPORT

provide additional margin during actual power production as specified in 10 CFR 50, Appendix G. The heatup and cooldown curves also include the effect of the reactor vessel flange.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

Pressure-temperature limit curves shown in Figure 4.2-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-640.

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

The power operated relief valves (PORVs) shall each have maximum lift setting and enable temperature in accordance with Table 4.2-3. The lift setting provided does not impose any reactor coolant pump restrictions.

The PORV setpoint is based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with the methodology described in Reference 1, including the exceptions noted in Section 4.2.1. The PORV lift setting shown in Table 4.2-3 accounts for appropriate instrument error.

#### 4.2.1.3 OPPS Enable Temperature (TS 3.4.9.3)

Two different temperatures are used to determine the OPPS enable temperature, they are the arming temperature and the calculated enable temperature. The arming temperature (when the OPPS rendered operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature is 343°F.

The calculated enable temperature is based on either a RCS temperature of less than 200°F or materials concerns (reactor vessel metal temperature less than  $RT_{NDT} + 50^{\circ}\text{F}$ ), whichever is greater. The calculated enable temperature does not address the piping limit attributed to a water hammer discharge. The calculated enable temperature is 308°F.

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

The calculation method governing the heatup and cooldown of the RCS requires the arming of the OPPS at and below the OPPS enable temperature specified in Table 4.2-3, and

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## PRESSURE AND TEMPERATURE LIMITS REPORT

disarming of the OPPS above this temperature. The OPPS is required to be enabled, i.e., OPERABLE, when any RCS cold leg temperature is less than or equal to this temperature.

From a plant operations viewpoint the terms "armed" and "enabled" are synonymous when it comes to activating the OPPS. As stated in the applicable operating procedure, the OPPS is activated (armed/enabled) manually before entering the applicability of TS 3.4.9.3. This is accomplished by placing two keylock switches (one in each train) into their "automatic" position. Once OPPS is activated (armed/enabled) reactor coolant system pressure transmitters will signal a rise in system pressure above the OPPS setpoint. This will initiate an alarm in the control room and open the OPPS PORVs.

## 4.2.1.4 Reactor Vessel Boltup Temperature (TS 3.4.9.1)

The minimum boltup temperature for the Reactor Vessel Flange shall be  $\geq 60^{\circ}\text{F}$ . Boltup is a condition in which the reactor vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

## 4.2.2 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and analyzed to determine changes in material properties. The capsule withdrawal schedule is provided in Table 4.5-3 of the UFSAR. Also, the results of these analyses shall be used to update Figures 4.2-1 and 4.2-2, and Tables 4.2-1 and 4.2-2. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

The pressure vessel material surveillance program (References 3 and 4) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standards utilize the reference nil-ductility temperature,  $RT_{NDT}$ , which is determined in accordance with ASME, Section III, NB-2331. The empirical relationship between  $RT_{NDT}$  and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to Section XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E 185-82.

Reference 10 is an NRC commitment made by FENOC to use only the calculated vessel fluence values when performing future capsule surveillance evaluations for BVPS Unit 1. This commitment is a condition of license Amendment 256 and will remain in effect until the NRC staff approves an alternate methodology to perform these evaluations. Best-estimate values generated using the FERRET Code may be provided for information only.

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PRESSURE AND TEMPERATURE LIMITS REPORT

4.2.3 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

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

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

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

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

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

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

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

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

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

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### LICENSING REQUIREMENTS MANUAL

#### PRESSURE AND TEMPERATURE LIMITS REPORT

##### 4.2.4 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.
2. WCAP-15570, Revision 2, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, April 2001.
3. WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," C. Brown, et. al., November 2000.
4. WCAP-8475, "Duquesne Light Company, Beaver Valley Unit No. 1 Reactor Vessel Radiation Surveillance Program," J. A. Davidson, October 1974.
5. WCAP-15569, "Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1," C. Brown, et al., November 2000.
6. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.
7. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)
8. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
9. Westinghouse Report, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule", Revision 1, April 2001.
10. FirstEnergy Nuclear Operating Company letter L-01-157, "Supplement to License Amendment Requests Nos. 295 and 167," dated December 21, 2001.

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PRESSURE AND TEMPERATURE LIMITS REPORT

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:

INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 21 EFPY:

1/4T, 233°F

3/4T, 196°F

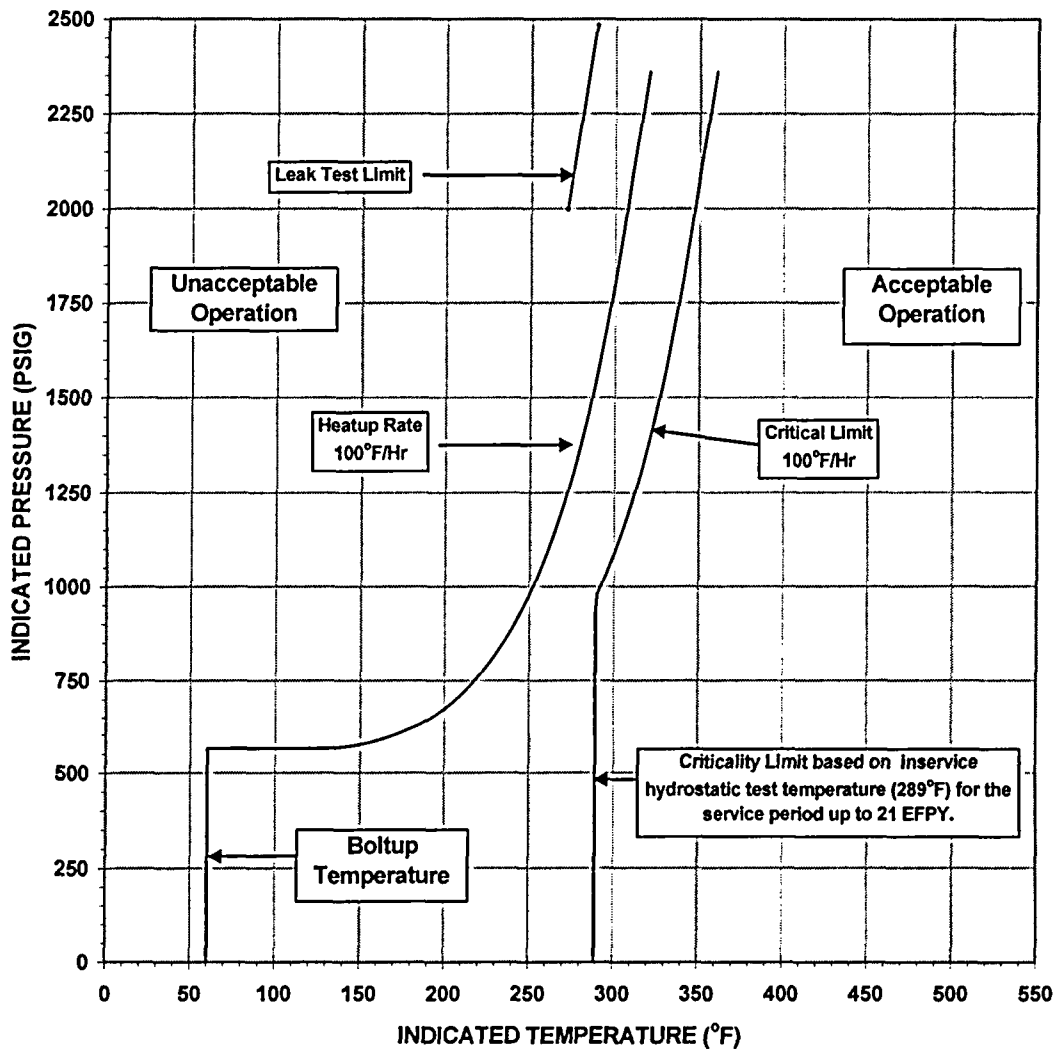


Figure 4.2-1  
Reactor Coolant System Heatup  
Limitations Applicable for the First 21 EFPY (TS 3.4.9.1)

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PRESSURE AND TEMPERATURE LIMITS REPORT

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:

INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 21 EFY:

1/4T, 233°F

3/4T, 196°F

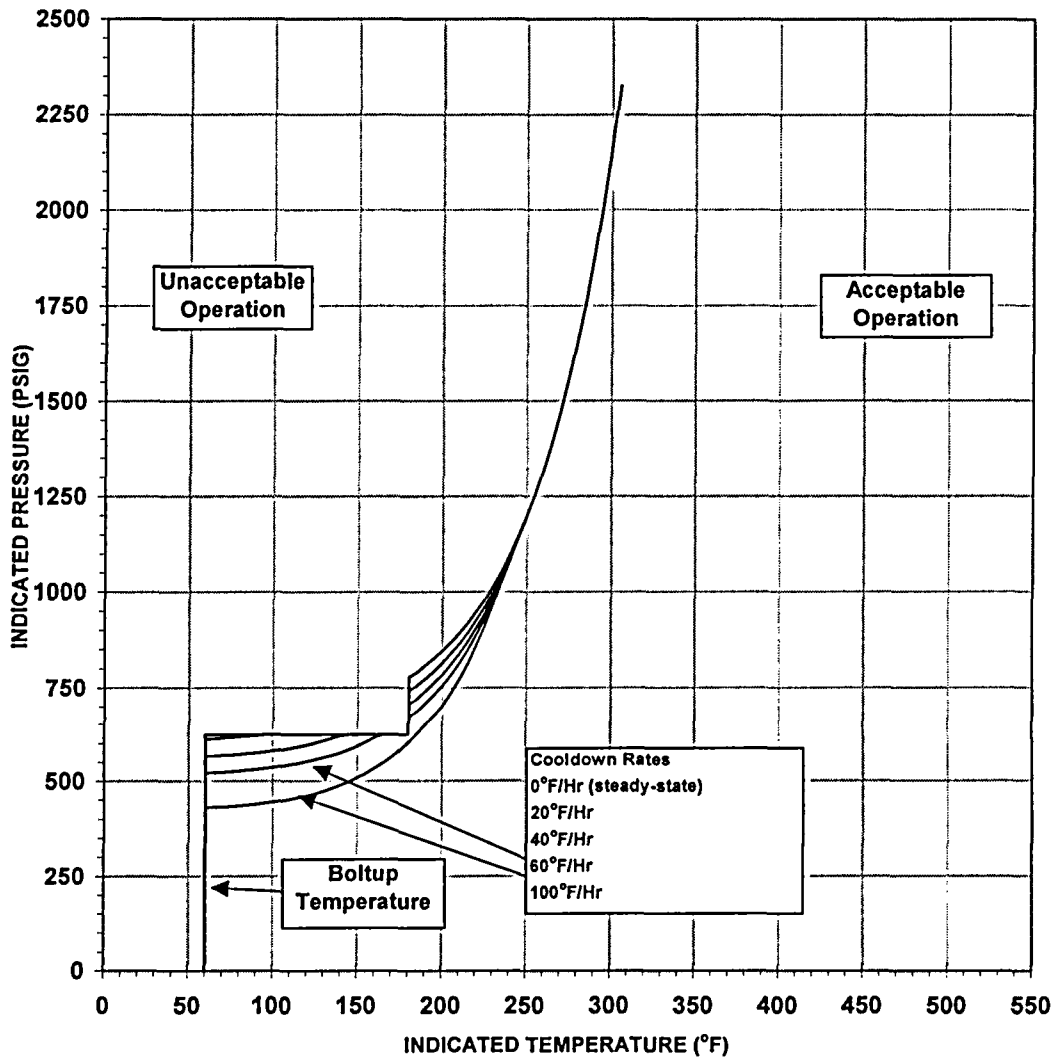


Figure 4.2-2  
Reactor Coolant System Cooldown  
Limitations Applicable for the First 21 EFY (TS 3.4.9.1)

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PRESSURE AND TEMPERATURE LIMITS REPORT

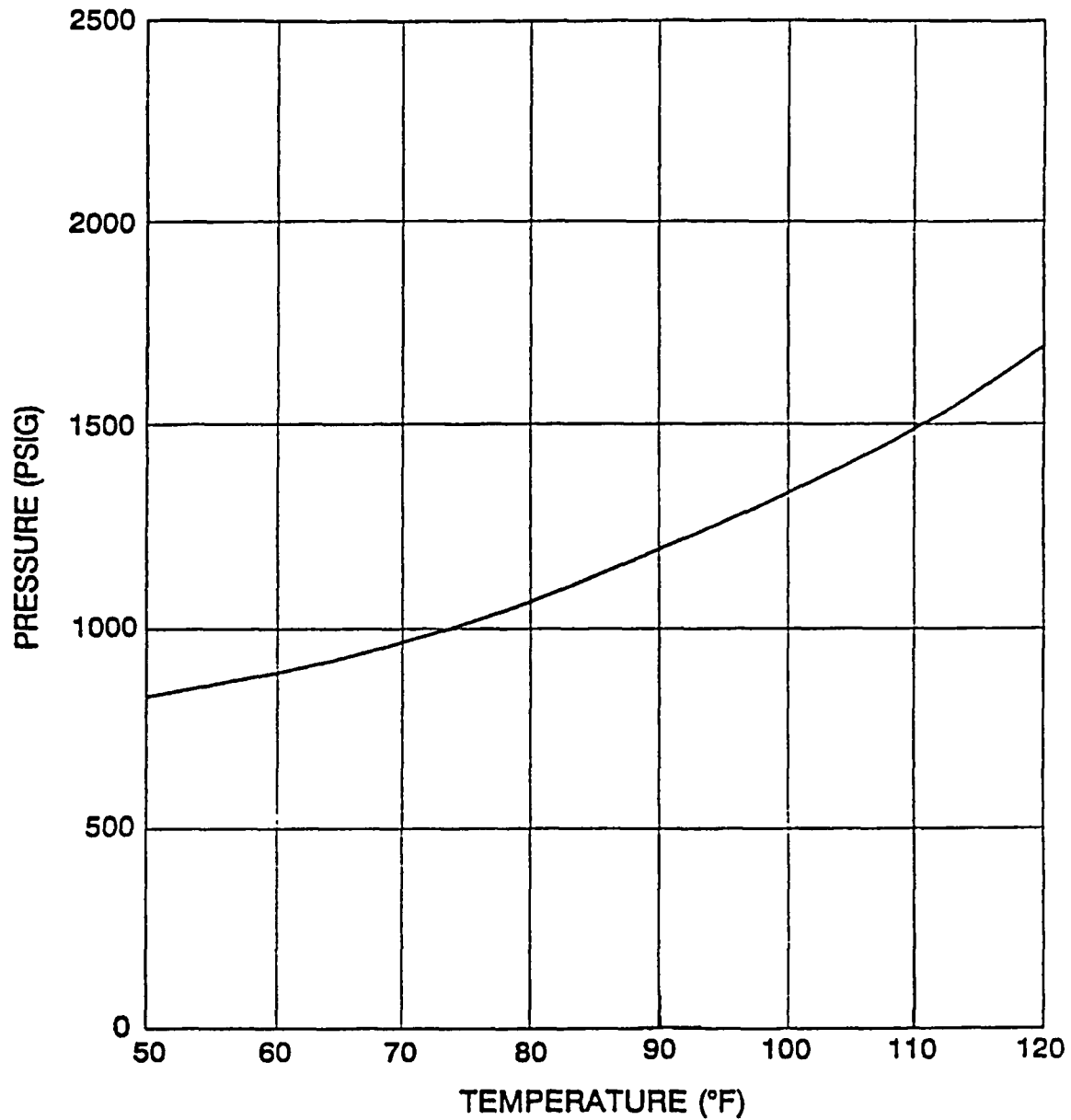


Figure 4.2-3  
Isolated Loop Pressure – Temperature Limit Curve (TS 3.4.9.1)



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## PRESSURE AND TEMPERATURE LIMITS REPORT

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

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

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### PRESSURE AND TEMPERATURE LIMITS REPORT

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

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

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## LICENSING REQUIREMENTS MANUAL

### PRESSURE AND TEMPERATURE LIMITS REPORT

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

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

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-3

Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

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

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-4

Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule $f^{(a)}$	$FF^{(b)}$	$\Delta RT_{NDT}^{(c)}$	$FF * \Delta RT_{NDT}$	$FF^2$
Lower Shell Plate B6903-1 <sup>(d)</sup> (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 <sup>(d)</sup> (Transverse)	V	.323	.689	137.81	94.95	.475
	U	.646	.878	131.84	115.76	.771
	W	.986	.996	179.99	179.27	.992
	Y	2.15	1.21	166.93	201.99	1.464
	SUM:				1104.89	7.404
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (1104.89) \div (7.404) = 149.2^\circ F$					
Beaver Valley	V	.323	.689	169.30	116.65	.475
Surv. Weld Material 305424 <sup>(d)</sup>	U	.646	.878	176.30	154.79	.771
	W	.986	.996	198.99	198.19	.992
	Y	2.15	1.21	189.41	229.19	1.464
	SUM:				698.82	3.702
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (698.82) \div (3.702) = 188.8^\circ F$					

Notes:

- (a)  $F$  = Calculated fluence from Beaver Valley Unit 1 capsule Y dosimetry analysis results, ( $\times 10^{19}$  n/cm<sup>2</sup>,  $E > 1.0$  Mev).
- (b)  $FF$  = fluence factor =  $f^{(0.28 - 0.1 * \log f)}$ .
- (c) The surveillance weld metal  $\Delta RT_{NDT}$  values have been adjusted by a ration factor of 1.06.
- (d) Data not credible.

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-4a

Calculation of Chemistry Factors<sup>(a)</sup>  
(Based on St. Lucie and Fort Calhoun Surveillance Capsule Data)

Material	Capsule	Capsule f <sup>(b)</sup>	FF <sup>(c)</sup>	$\Delta RT_{NDT}$ <sup>(d)</sup>	FF * $\Delta RT_{NDT}$	FF <sup>2</sup>
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
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (246.7) \div (2.91) = 84.8^{\circ}F$					
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
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (603.3) \div (2.695) = 223.9^{\circ}F$					

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 ( $\times 10^{19}$  n/cm<sup>2</sup>, E > 1.0 Mev) from Reference 2.
- (c) FF = fluence factor =  $f^{(0.28 - 0.1 * \log f)}$ .
- (d)  $\Delta RT_{NDT}$  values are the measured 30 ft-lb. shift values taken from Reference 2.

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-4b

St. Lucie and Fort Calhoun Surveillance Weld Data<sup>(a)(b)</sup>

Material	Capsule	Cu	Ni	Irradiated Temperature °F	Fluence 10 <sup>19</sup> n/cm <sup>2</sup>	ΔRT <sub>NDT</sub>
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.

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-5

Reactor Vessel Beltline Material Properties

Material Description	Cu(%)	Ni(%)	Chemistry Factor	Initial $RT_{NDT}(^{\circ}F)$ <sup>(a)</sup>
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.



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Table 4.2-6

Summary of Adjusted Reference Temperature (ARTs) for 22 EFPY

MATERIAL DESCRIPTION	22 EFPY	
	1/4T ART(°F) <sup>(a)</sup>	3/4T ART(°F) <sup>(a)</sup>
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 <sup>(b)</sup>	233	193
Intermediate Shell Longitudinal Weld 19-714A/B	145	102
- Using S/C Data <sup>(b)</sup>	143	100
Intermediate to Lower Shell Circ. Weld 11-714	152	119
- Using S/C Data <sup>(c)</sup>	86	63
Lower Shell Longitudinal Weld 20-714A/B	159	111
- Using S/C Data <sup>(d)</sup>	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  $\sigma_{\Delta}$ .)
- (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  $\sigma_{\Delta}$ .)

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-7

Calculation of Adjusted Reference Temperatures (ARTs) for 22 EFPY

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

Notes:

- (a) Initial RT<sub>NDT</sub> 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  $\sigma_{\Delta}$ .)

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-8  
Reactor Vessel Toughness Data (Unirradiated)

COMPONENT	HEAT NO.	CODE NO.	MATERIAL TYPE	Cu (%)	Ni (%)	P (%)	T <sub>NDT</sub> (°F)	RT <sub>NDT</sub> (°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.

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-9

RT<sub>PTS</sub> Calculation for Beltline Region Materials at EOL (28 EFPY)

Material	Fluence (10 <sup>19</sup> n/cm <sup>2</sup> , E>1.0 MeV)	FF	CF (°F)	Δ RT <sub>PTS</sub> <sup>(a)</sup> (°F)	Margin (°F)	RT <sub>NDT(U)</sub> <sup>(b)</sup> (°F)	RT <sub>PTS</sub> <sup>(c)</sup> (°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 <sup>(e)</sup>	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 <sup>(e)</sup>	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 <sup>(f)</sup>	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 <sup>(d)</sup>	3.53	1.329	84.8	112.3	44	-56	101

Notes:

- (a)  $\Delta RT_{PTS} = CF * FF$ .
- (b) Initial RT<sub>NDT</sub> 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  $\sigma_{\Delta}$ .
- (f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full  $\sigma_{\Delta}$ .

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PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-10

RT<sub>PTS</sub> Calculation for Beltline Region Materials at Life extension (45 EFY)

Material	Fluence (10 <sup>19</sup> n/cm <sup>2</sup> , E>1.0 MeV)	FF	CF (°F)	Δ RT <sub>PTS</sub> <sup>(c)</sup> (°F)	Margin (°F)	RT <sub>NDT(U)</sub> <sup>(a)</sup> (°F)	RT <sub>PTS</sub> <sup>(b)</sup> (°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 <sup>(e)</sup>	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 <sup>(e)</sup>	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 <sup>(f)</sup>	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 <sup>(d)</sup>	5.82	1.43	84.8	121.3	44	-56	109

Notes:

- (a) Initial RT<sub>NDT</sub> values of the plate material are measured values while the weld material values are generic.
- (b) RT<sub>PTS</sub> = RT<sub>NDT(U)</sub> + ΔRT<sub>PTS</sub> + Margin (°F).
- (c) ΔRT<sub>PTS</sub> = 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<sub>Δ</sub>.
- (f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ<sub>Δ</sub>.