ENCLOSURE 2

WATTS BAR UNIT 1

RCS PRESSURE AND TEMPERATURE LIMITS REPORT

REVISION 2

APPENDIX "A"

TO RCS SYSTEM DESCRIPTION N3-68-4001

WATTS BAR UNIT 1

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

REVISION 2

Prepared by : Chris Morgan 12-19-94

Checked by: [illegible] 12-19-94

Approved by : [illegible] for WLE 12-19-94

RCS PRESSURE AND TEMPERATURE LIMITS REPORT FOR WATTS BAR UNIT 1

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This PTLR for Watts Bar Unit 1 has been prepared in accordance with the requirements of Technical Specification 5.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications affected by this report are listed below:

LCO 3.4.3. RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

2.0 RCS Pressure and Temperature Limits

The limits for LCO 3.4.3 are presented in the subsection which follows. These limits have been developed (Ref. 1, 4) using the NRC-approved methodologies specified in Specification 5.9.6.

2.1 RCS Pressure and Temperature (P/T) Limits (LCO 3.4.3)

2.1.1 The RCS temperature rate-of-change limits are (Ref. 1):

a. A maximum heatup Rate 100°F per hour.
b. A maximum cooldown Rate 100°F per hour.
c. A maximum temperature change of ≤ 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

2.1.2 The RCS P/T limits for heatup, cooldown, inservice hydrostatic and leak testing, and criticality are specified by Figures 2.1-1 and 2.1-2 (Ref. 1).

NOTE: The heat-up and cool-down curves are based on beltline conditions and do not compensate for pressure differences between the pressure transmitter and reactor midplane/beltline nor instrument inaccuracies. Refer to Table 2.1-3 for pressure differences (Ref. 2).

3.0 Cold Overpressure Mitigation System (LCO 3.4.12)

The lift setting limits for the pressurizer Power Operated Relief Valves (PORVs) are presented in the subsection which follows. These lift setting limits have been developed using the NRC-approved methodologies specified in Specification 5.9.6.

3.1 Pressurizer PORV Lift Setting Limits

The pressurizer PORV lift setting limits are specified by Figure 3.1-1 (Ref. 2).

NOTE: These setpoints include allowance for pressure difference between the pressure transmitter and reactor midplane, and also includes 63 psig pressure channel uncertainty.

4.0 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 4.0-1. The results of these examinations shall be used to update Figures 2.1-1, 2.1-2, and 3.1-1 through 3.1-4.

The pressure vessel steel surveillance program (Ref. 3) is in compliance with Appendix H to 10 CFR 50, entitled "Reactor Vessel Material Surveillance Program Requirements". The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. 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 III of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82. The removal schedule is provided in Table 4.0-1.

5.0 Supplemental Data Tables

Table 5.1 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2, predictions. This table was intentionally left blank since no capsules were removed to date.

Table 5.2 shows calculations of the surveillance material chemistry factors using surveillance capsule data. This table was intentionally left blank since no capsules were removed to date.

Table 5.3 provides the required Watts Bar Unit 1 reactor vessel toughness data. The bolt-up temperature is also included in this table.

Table 5.4 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves.

Table 5.5 provides a summary of the adjusted reference temperature (ART) values of the Watts Bar Unit 1 reactor vessel beltline materials at the 1/4-T and 3/4-T locations for 7 EFPY.

Table 5.6 shows example calculations of the adjusted reference temperature (ART) values at 7 EFPY for the limiting Watts Bar Unit 1 reactor vessel material (Intermediate Shell Forging 05).

Table 5.7 provides a summary of the fluence values used in the PTS evaluation.

Table 5.8 provides RT_{PTS} values for Watts Bar Unit 1 for 32 EFPY.

Table 5.9 provides RT_{PTS} values for Watts Bar Unit 1 for 48 EFPY.

REFERENCES

1. WCAP-13829, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 1", August 1993.

2. Westinghouse Letter to TVA, WAT-D-9448, "Revised COMS PORV Setpoints," August 27, 1993.

3. WCAP-9298, Revision 1, "Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program", April 1993.

4. Westinghouse Letter to TVA, WAT-D-9526, "COMS".

5. WCAP-14040, Revision 1, "Methodology Used To Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", December 1994.

6. Westinghouse Letter to TVA, WAT-D-9871, "Watts Bar Nuclear Plant Units 1 & 2 Pressure/Temperature Limits," December 16, 1994.

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
INITIAL RT_{NDT} 47 °F
LIMITING ART AT 7 EFPY: 1/4-T, 181.1 °F
3/4-T, 147.7 °F

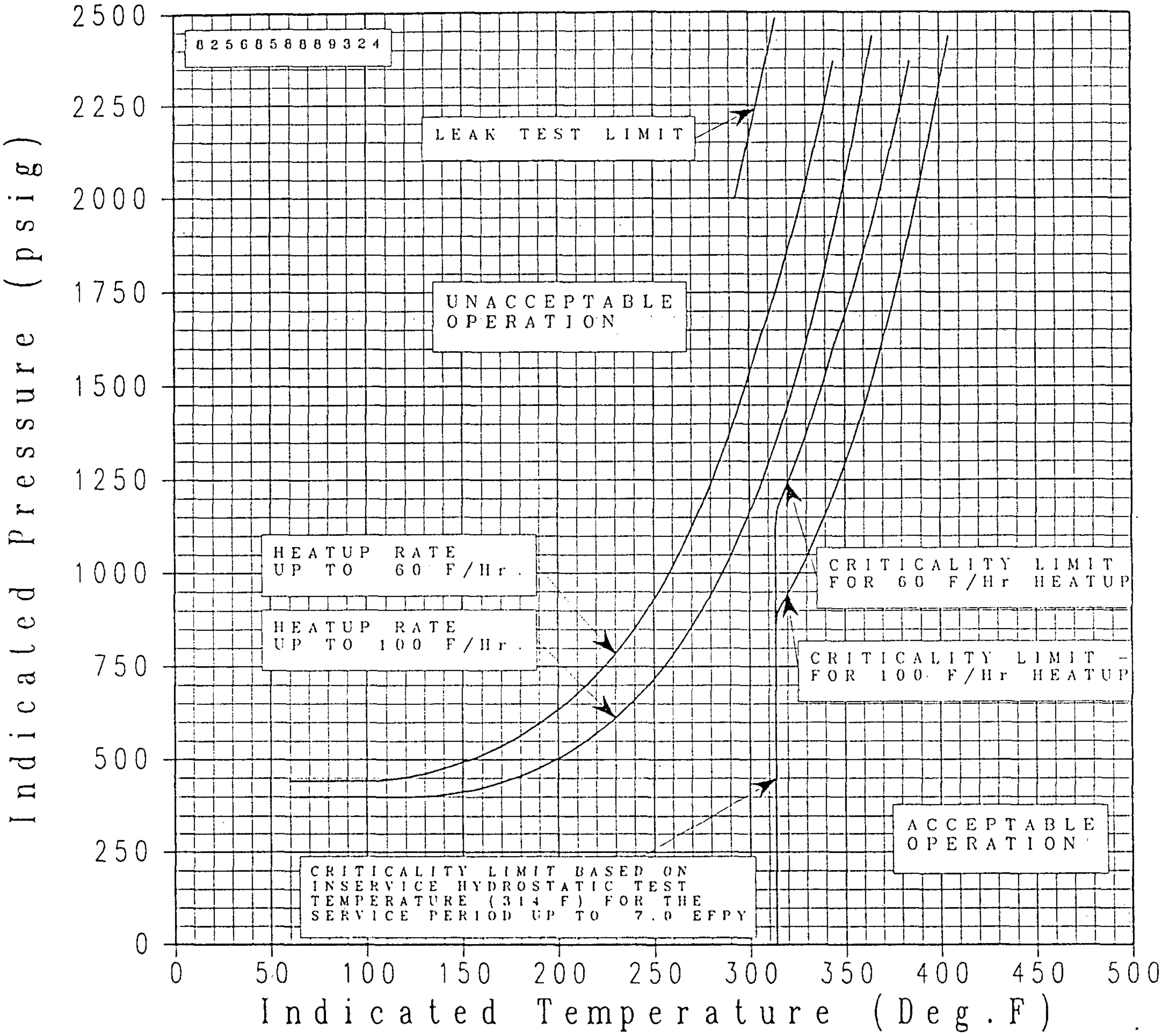

Figure 2.1-1

Watts Bar Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates of 60 and 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)

(Plotted Data (Ref. 1) provided on Table 2.1-1)

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data points plotted on Figure 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMITS (60 °F/HR)	CRITICALITY LIMITS (100 °F/HR)
60	443.05	401.48			
65	443.05	401.48			
70	443.05	401.48			
75	443.05	401.48			
80	443.05	401.48			
85	443.05	401.48			
90	443.05	401.48			
95	443.05	401.48			
100	443.05	401.48			
105	444.11	401.48			
110	446.29	401.48			
115	449.34	401.48			
120	453.33	401.48			
125	458.10	401.48			
130	463.72	402.34			
135	470.07	404.05			
140	477.24	406.56			
145	485.17	409.93			
150	493.84	414.14			
155	503.41	419.16			
160	513.86	425.01			
165	525.20	431.68			
170	537.41	439.16			
175	550.72	447.59			
180	565.12	456.94			

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data points plotted on Figure 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMITS (60 °F/HR)	CRITICALITY LIMITS (100 °F/HR)
185	580.63	467.24			
190	597.27	478.55			
195	615.30	490.90			
200	634.60	504.29			
205	655.51	518.93			
210	678.00	534.82			
215	702.06	551.92			
220	727.92	570.53			
225	755.91	590.51			
230	785.79	612.21			
235	818.05	635.46			
240	852.61	660.68			
245	889.70	687.66			
250	929.52	716.88			
255	972.27	748.15			
260	1018.19	781.76			
265	1067.49	818.06			
270	1120.42	856.92			
275	1177.15	898.66			
280	1237.98	943.47			
285	1303.13	991.58			
290	1373.09	1043.20			
293			2000		
295	1447.99	1098.55			
300	1527.97	1157.76			

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data points plotted on Figure 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMITS (60 °F/HR)	CRITICALITY LIMITS (100 °F/HR)
305	1612.31	1221.51			
310	1686.16	1289.68			
314			2485	0 to 1120.42	0 to 856.92
315	1765.15	1362.69		1177.15	898.66
320	1849.89	1440.81		1237.98	943.47
325	1940.35	1524.60		1303.13	991.58
330	2037.07	1614.19		1373.09	1043.20
335	2140.54	1709.92		1447.99	1098.55
340	2250.90	1812.16		1527.97	1157.76
345	2368.62	1921.54		1612.31	1221.51
350		2038.25		1686.16	1289.68
355		2162.68		1765.15	1362.69
360		2295.42		1849.89	1440.81
365		2436.49		1940.35	1524.60
370				2037.07	1614.19
375				2140.54	1709.92
380				2250.90	1812.16
385				2368.62	1921.54
390					2038.25
395					2162.68
400					2295.42
405					2436.49

MATERIAL PROPERTY BASIS
LIMITING MATERIAL: INTERMEDIATE SHELL FORGING 05
LIMITING ART AT 7 EFPY: 1/4-T, 181.1 °F
3/4-T, 147.7 °F

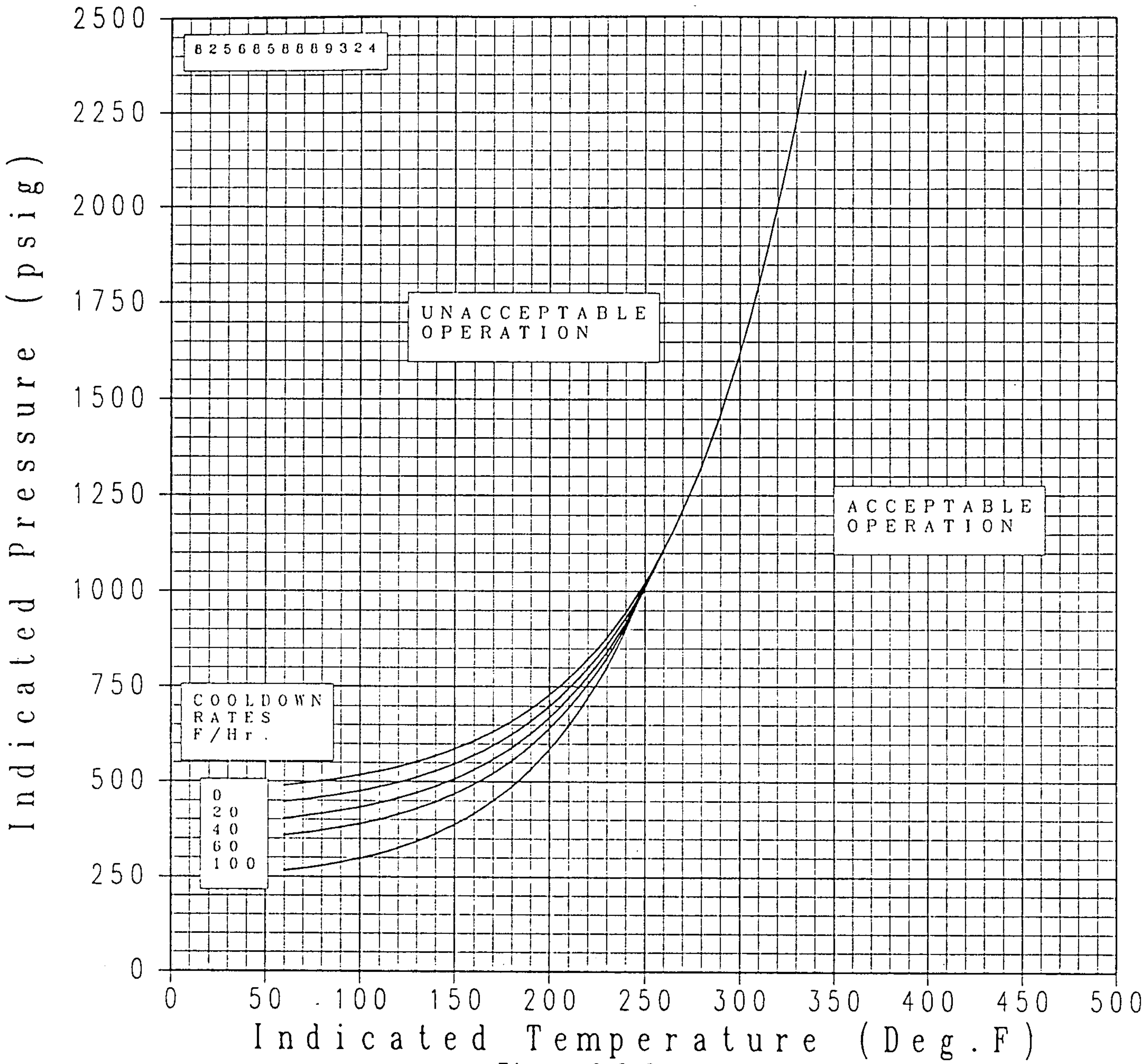

Figure 2.1-2

Watts Bar Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)

(Plotted Data (Ref. 1) provided on Table 2.1-2)

Table 2.1-2
Watts Bar Unit 1 Cooldown Limits
(Data plotted on Figure 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
60	266.59	359.39	404.02	447.48	490.13
65	269.45	362.11	406.70	450.24	492.77
70	272.56	365.11	409.67	453.18	495.72
75	276.04	368.41	412.91	456.38	498.88
80	279.82	371.97	416.40	459.81	502.28
85	283.98	375.89	420.21	463.54	505.94
90	288.53	380.12	424.32	467.55	509.87
95	293.54	384.74	428.79	471.89	514.09
100	298.99	389.73	433.61	476.55	518.64
105	304.97	395.18	438.77	481.60	523.52
110	311.45	401.06	444.40	487.03	528.77
115	318.55	407.40	450.52	492.81	534.42
120	326.20	414.30	457.10	499.12	540.39
125	334.59	421.80	464.24	505.93	546.91
130	343.66	429.89	471.92	513.26	553.93
135	353.56	438.60	480.25	521.17	561.47
140	364.21	448.06	489.20	529.67	569.58
145	375.84	458.32	498.80	538.75	578.31
150	388.41	469.38	509.23	548.62	587.56
155	402.06	481.35	520.50	559.26	597.63
160	416.74	494.17	532.63	570.71	608.47
165	432.73	508.12	545.63	582.92	620.12
170	449.91	523.16	559.73	596.19	632.50
175	468.61	539.30	574.96	610.49	645.97
180	488.79	556.81	591.21	625.85	660.44
185	510.54	575.72	608.89	642.29	675.98
190	534.09	595.96	627.89	660.09	692.57
195	559.46	617.94	648.27	679.12	710.56
200	586.80	641.47	670.32	699.74	729.70

Table 2.1-2
Watts Bar Unit 1 Cooldown Limits
(Data plotted on Figure 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
205	616.49	667.02	693.97	721.92	750.49
210	648.35	694.35	719.52	745.65	772.78
215	682.74	723.96	746.96	771.30	796.64
220	719.95	755.76	776.41	798.79	822.23
225	760.01	789.97	808.37	828.31	849.99
230	803.10	826.78	842.54	860.26	879.55
235	849.59	866.62	879.31	894.47	911.51
240	899.62	909.35	918.85	931.19	945.77
245	953.53	955.36	961.60	970.69	982.53
250	1011.36	1004.81	1007.39	1013.15	1022.00
255		1057.86	1056.69	1058.81	1064.40
260			1109.45	1107.90	1109.98
265					1158.95
270					1211.57
275					1268.05
280					1328.61
285					1393.44
290					1463.21
295					1537.95
300					1618.11
305					1703.92
310					1795.88
315					1894.39
320					1999.72
325					2112.46
330					2232.88
335					2361.41

Table 2.1-3

Pressure Differentials

Number of Pumps	Delta P (psi)
0	5.2
1	31.0
2	38.0
3	52.0
4	74.0

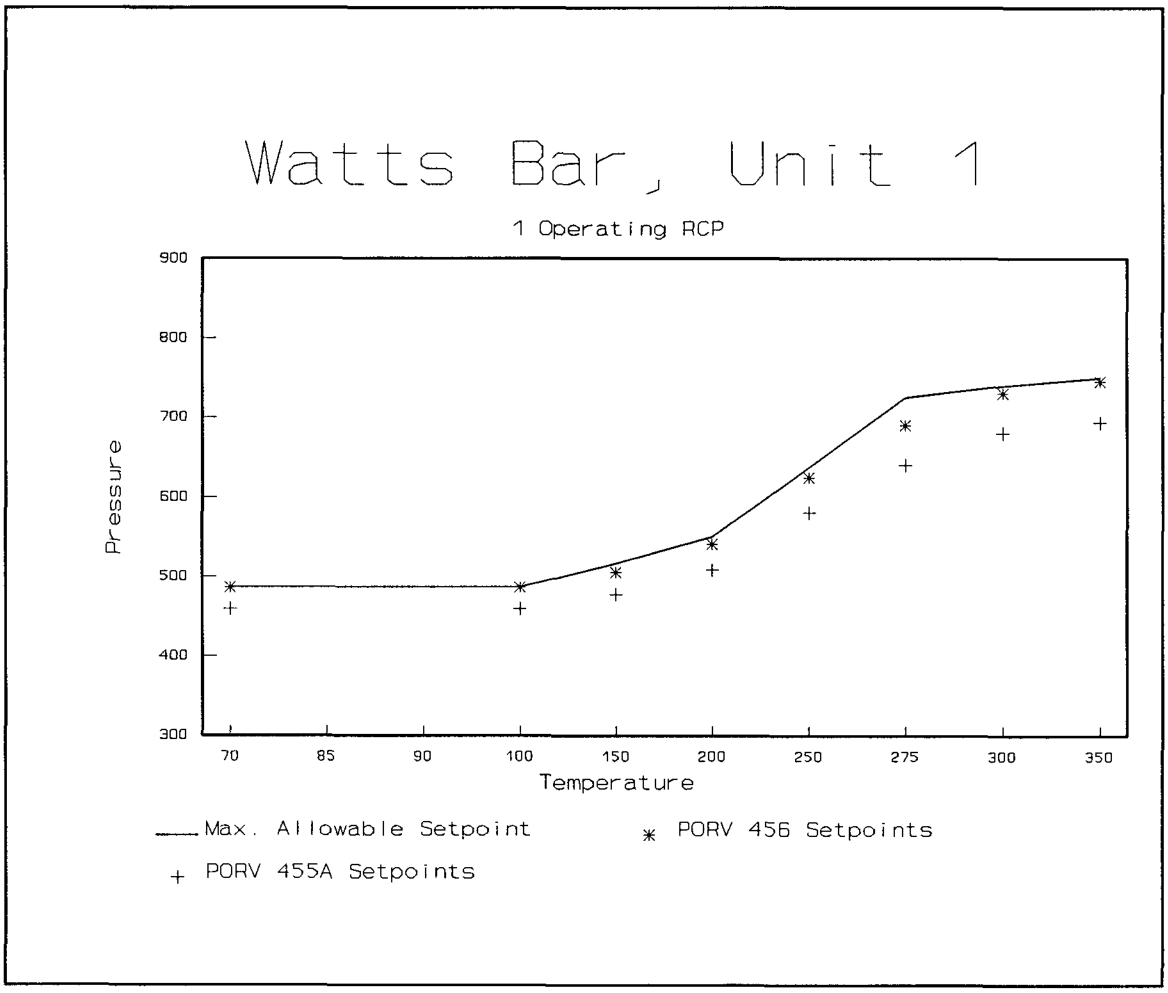

Figure 3.1-1
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

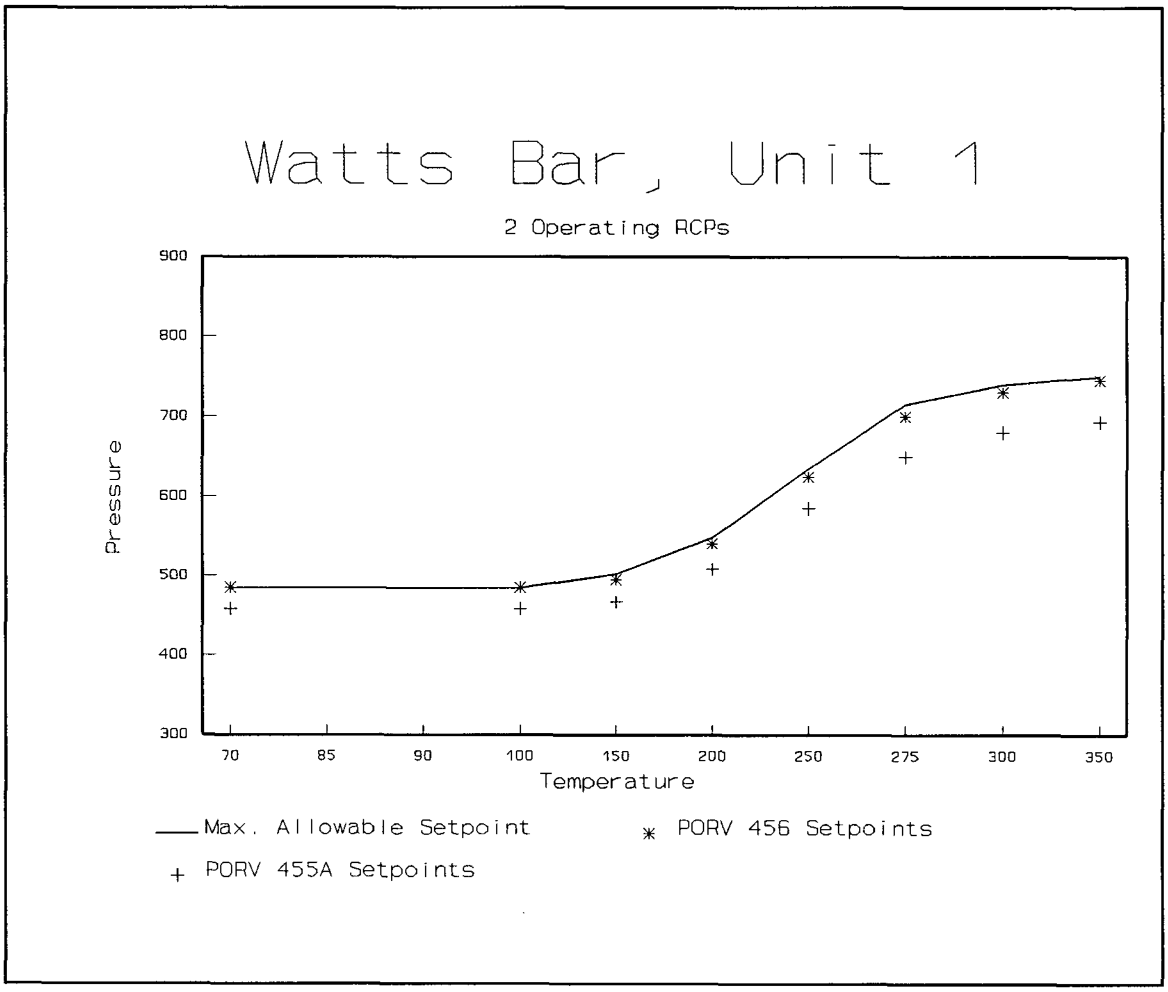

Figure 3.1-2
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

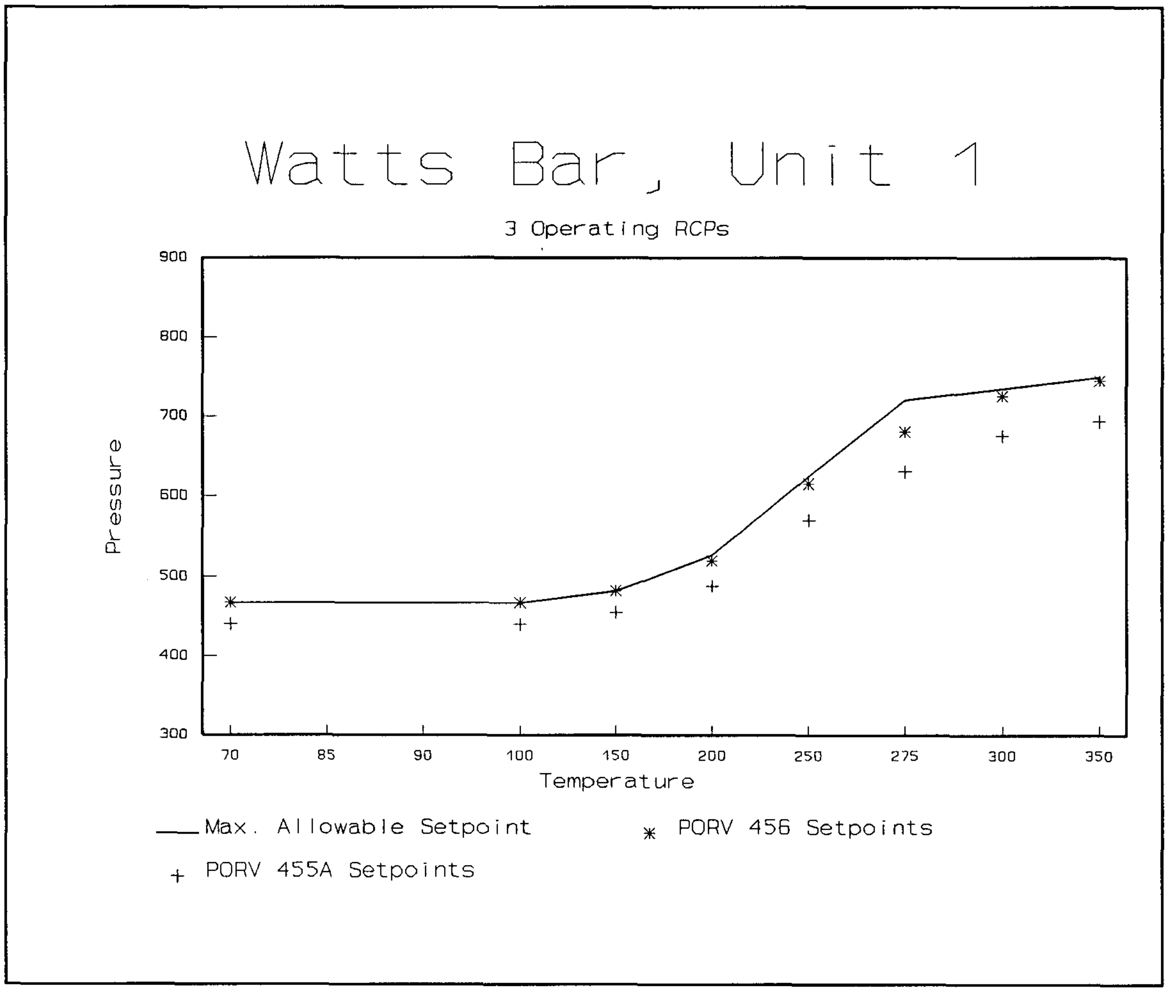

Figure 3.1-3
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

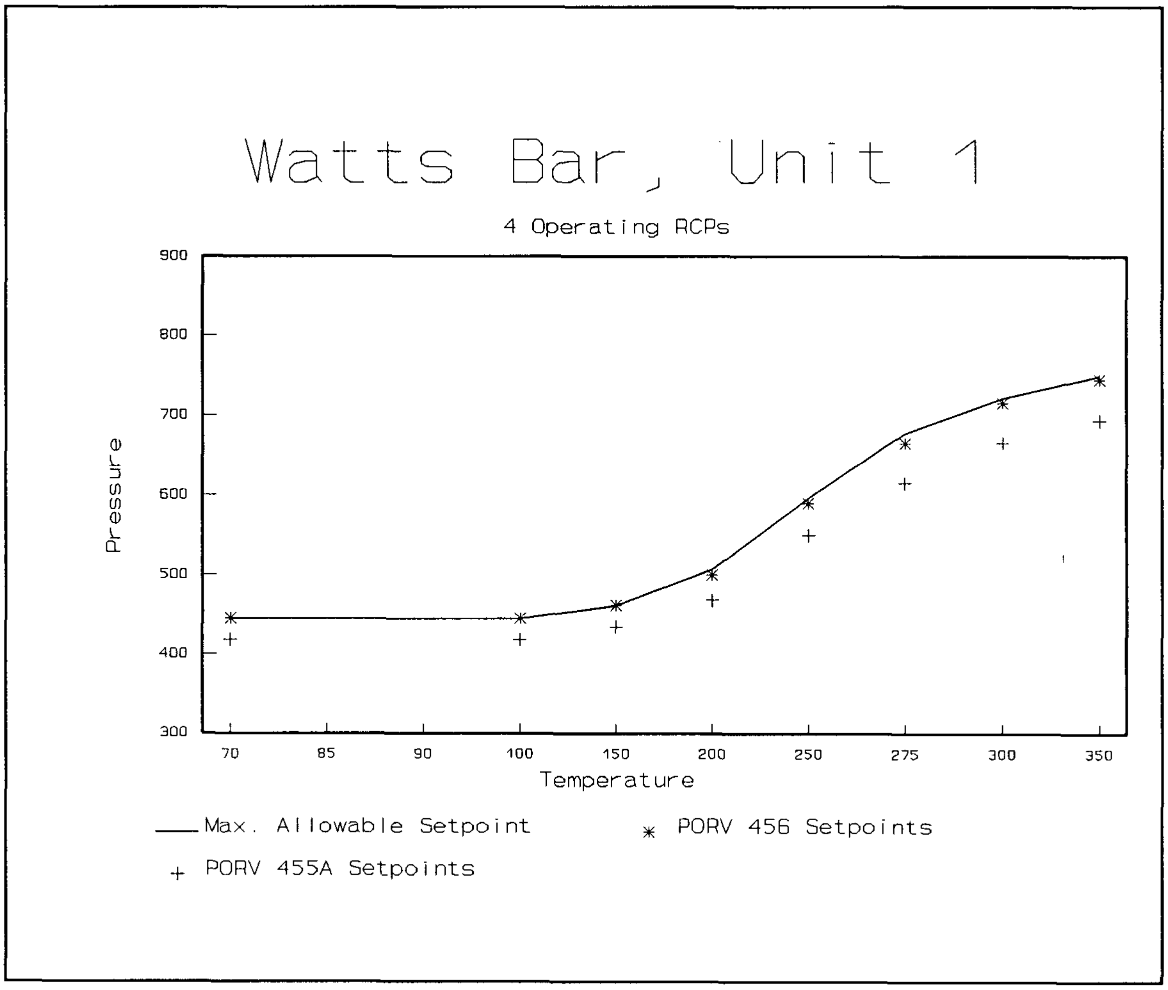

Figure 3.1-4
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

NOTE: Westinghouse PORV Numbers 456 and 455A
Correspond to TVA PORV Numbers 334 and 340A

Table 3.1-1
Watts Bar Unit 1 PORV Setpoints vs Temperature
(Data (Ref. 3) plotted on Figures 3.1-1 through 3.1-4)

TEMP (°F)	SETPOINTS (PSIG)							
	1 RCS PUMP OPERATING		2 RCS PUMPS OPERATING		3 RCS PUMPS OPERATING		4 RCS PUMPS OPERATING	
	PORV-334	PORV-340A	PORV-334	PORV-340A	PORV-334	PORV-340A	PORV-334	PORV-340A
70	486	459	485	458*	467	440*	445*	418*
100	486	459	485	458*	467	440*	445*	418*
150	505	477	495	467	482	455*	462	435*
200	540	508	540	508	520	488	500	468
250	625	580	625	585	615	570	590	550
275	690	640	700	650	680	630	665	615
300	730	680	730	680	725	675	715	665
350	745	690	745	690	745	690	745	690
450	2350	2350	2350	2350	2350	2350	2350	2350

* Setpoint violates pump seal limit which includes 63 psig pressure channel uncertainty

Table 4.0-1
Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Time[a][b][d]	Estimated Capsule Fluence (n/cm^2)[c]
U	56.0°	3.6	1st Refueling Outage	3.60×10^{18}
W	124.0°	3.6	5.4	1.90×10^{19}
X	236.0°	3.6	8.9	3.19×10^{19}
Z	304.0°	3.6	17.8	6.38×10^{19}
V	58.5°	3.6	Stand-By	----
Y	238.5°	3.6	Stand-By	----

(a) Effective Full Power Years (EFPY) from plant startup.

(b) Removal times are based on not-to-exceed criteria of E185-82, Section 7.6.2. Capsules should be removed on the last cycle prior to reaching the indicated time.

(c) Based on design basis fluence of 3.18×10^{19} n/cm^2 ($E > 1MeV$).

(d) Withdraw two capsules before the vessel exceeds 5.4 EFPY. The results of the capsule analysis will be reviewed and should an amended removal schedule be required, two standby capsules are available for additional monitoring.[1] If the results of capsule testing predict an end of life use of < 50 ft-lb, TVA will perform the necessary analysis required by Appendix G, IV.A.1 to ensure adequate safety margins.[2]

TABLE 5.1

Comparison of the Watts Bar Unit 1 Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Decrease with Regulatory Guide 1.99, Revision 2, Predictions

Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	30 ft-lb Transition Temperature Shift		Upper Shelf Energy Decrease	
			Predicted[a] (°F)	Measured (°F)	Predicted[a] (%)	Measured (%)
Intermediate Shell Forging 05 (tangential)						
Intermediate Shell Forging 05 (axial)						
Weld Metal						
HAZ Metal						

(a) Based on Regulatory Guide 1.99, Revision 2, methodology using average weight percent values of Cu an Ni.

NOTE: No capsules have been removed from the Watts Bar Unit 1 reactor vessel at this time.

TABLE 5.2

Watts Bar Unit 1 Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Fluence (n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 05 (Tangential)						
Intermediate Shell Forging 05 (Axial)						
				Sum:		
	Chemistry Factor =					
Weld Metal						
				Sum:		
	Chemistry Factor =					

NOTE: No capsules have been removed from the Watts Bar Unit 1 reactor vessel at this time.

TABLE 5.3			
Watts Bar Unit 1 Reactor Vessel Toughness Table (Unirradiated)			
Material Description	Cu (%)[a]	Ni (%)[a]	Initial RT_{NDT} (°F)[b]
Closure Head Flange	0.13	0.75	-42
Vessel Flange	--	0.92	-40[c]
Intermediate Shell Forging 05	0.17	0.80	47
Lower Shell Forging 04	0.08	0.83	5
Circumferential Weld	0.05	0.70	-43

NOTES:

a) Average values of copper and nickel weight percent.
b) Initial RT_{NDT} values are measured values.
c) Used in the consideration of flange requirements for heatup/cooldown curves. Per methodology given in WCAP-14040, the minimum boltup temperature is 60°F.

TABLE 5.4					
Watts Bar Unit 1 Reactor Vessel Surface Fluence Values at 7 EFPY (n/cm^2, E > 1.0 MeV)					
Azimuthal	0°	15°	25°	35°	45°
Surface	4.13×10^{18}	6.15×10^{18}	6.96×10^{18}	5.67×10^{18}	6.50×10^{18}

TABLE 5.5		
Summary of ARTs for the Watts Bar Unit 1 Reactor Vessel Beltline Materials at the 1/4-T and 3/4-T Locations for 7 EFPY		
Component	7 EFPY[a]	
	1/4-T (°F)	3/4-T (°F)
Intermediate Shell Forging 05	181.11[b]	147.70[b]
Lower Shell Forging 04	77.68	56.54
Circumferential Weld	60.14	25.72

<u>NOTES:</u>
(a) Calculated using the peak vessel fluence of 6.96×10^{18} n/cm^2 (E > 1.0 MeV).
(b) Used to generate the heatup/cooldown curves.

TABLE 5.6		
Calculation of Adjusted Reference Temperatures at 7 EFPY for the Limiting Watts Bar Unit 1 Reactor Vessel Material (Intermediate Shell Forging 05)		
Parameter	Values	
Operating Time	7 EFPY	
Material	Inter. Shell Forging 05	Inter. Shell Forging 05
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F	132	132
Fluence (f), $\div 10^{19}$ n/cm^2 (E > 1.0 MeV) [(a)]	0.4188	0.1517
Fluence Factor (FF)[(b)]	0.758	0.505
ΔRT_{NDT} = CF x FF, °F	100.1	66.7
Initial RT_{NDT} (I), °F	47	47
Margin (M), °F [(c)]	34	34
ART = I + (CFxFF) + M, °F per Regulatory Guide 1.99, Revision 2	181.1	147.7

NOTES:

(a) Fluence, f, is based upon f_{surf} = 6.96 x 10^{18} n/cm^2. The Watts Bar Unit 1 reactor vessel wall thickness is 8.465 inches at the beltline region.

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

(c) Margin is calculated as $M = (\sigma_i^2 + \sigma_\Delta^2)^{0.5}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is 0°F since the initial RT_{NDT} value is a measured value. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for the forging, except that σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

TABLE 5.7					
Watts Bar Unit 1 Reactor Vessel Surface Fluence Values at 32 and 48 EFPY (n/cm^2, E > 1.0 MeV)					
EFPY	0°	15°	25°	35°	45°
32	1.89×10^{19}	2.81×10^{19}	3.18×10^{19}	2.59×10^{19}	2.97×10^{19}
48	2.84×10^{19}	4.22×10^{19}	4.77×10^{19}	3.89×10^{19}	4.46×10^{19}

TABLE 5.8

RT_{PTS} Values for Watts Bar Unit 1 for 32 EFPY

Material	CF (°F)	Surface Fluence (n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (CF x FF) (°F)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Forging 05	132	3.18×10^{19}	1.30	171.6	47	34	253
Lower Shell Forging 04	51	3.18×10^{19}	1.30	66.3	5	34	106
Circ. Weld	68	3.18×10^{19}	1.30	88.4	-43	56	102

TABLE 5.9

RT_{PTS} Values for Watts Bar Unit 1 for 48 EFPY

Material	CF (°F)	Surface Fluence (n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (CF x FF) (°F)	I (°F)	M (°F)	RT_{PTS} (°F)
Inter. Shell Forging 05	132	4.77×10^{19}	1.39	185.5	47	34	265
Lower Shell Forging 04	51	4.77×10^{19}	1.39	70.9	5	34	110
Circ. Weld	68	4.77×10^{19}	1.39	94.5	-43	56	108

SOURCE NOTES

1. NCO820285003
2. NCO820285004

ENCLOSURE 3

WESTINGHOUSE WCAP-14040

METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES

REVISION 1

~~9412300015~~ 42 pp

WESTINGHOUSE CLASS 3 (Non-Proprietary)

WCAP-14040

METHODOLOGY USED TO DEVELOP COLD OVERPRESSURE MITIGATING SYSTEM SETPOINTS AND RCS HEATUP AND COOLDOWN LIMIT CURVES

WOG Program
MUHP-3024
Revision 1

J. D. Andrachek
S. M. DiTommaso
M. J. Malone
M. C. Rood

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WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Technology Division
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

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- An assumed flaw in the wall of the reactor vessel has a depth equal to 1/4 of the thickness of the vessel wall and a length equal to 1-1/2 times the vessel wall thickness,

- A factor of 2 is applied to the membrane stress intensity factor (K_{IM}),

- The limiting toughness is based upon a reference value (K_{Ia}), which is a lower bound of the dynamic crack initiation or arrest toughnesses, and

- 2-sigma margins are applied in determining the adjusted reference temperature (ART).

This section describes the methodology used by Westinghouse Electric Corporation to develop the allowable pressure-temperature relationships for normal plant heatup and cooldown rates that are included in the Pressure-Temperature Limits Report (PTLR). First, the methodology describing how the neutron fluence is calculated for the reactor vessel beltline materials is provided. Next, sections describing fracture toughness properties, adjusted reference temperature calculation, criteria for allowable pressure-temperature relationships, and pressure-temperature curve generation are provided.

2.2 NEUTRON FLUENCE METHODOLOGY

In performing the fast neutron exposure evaluations for a reactor vessel, two distinct sets of transport calculations are typically carried out. The first set, a computation in the conventional forward mode, is used to obtain the neutron fluence rate and relative neutron energy distributions at all locations between the core and the outside of the reactor vessel. The neutron spectral information is required for the interpretation of neutron dosimetry withdrawn from the surveillance capsules for validation of the plant specific calculations.

The second set of calculations consists of a series of adjoint analyses relating the fast neutron flux, $\phi(E > 1.0\ \text{MeV})$, at various vessel positions to neutron source distributions within the reactor core. The source importance functions generated from these adjoint analyses provide the basis for all absolute exposure calculations and comparison with measurement. These importance functions, when combined with fuel cycle specific neutron source distributions, yield absolute predictions of neutron exposure at the locations of interest for each cycle of irradiation. It is important to note that the cycle specific neutron source distributions utilized in these analyses include not only spatial variations of fission rates within the reactor core but also account for the effects of varying neutron yield per fission and fission spectrum introduced by the build-up of plutonium as the burnup of individual fuel assemblies increases. The detailed cycle-by-cycle variation of the neutron flux is generally calculated for the maximum flux position and each weld location on the inside of the reactor vessel. Axial calculations are normally not necessary since the maximum beltline fluence is desired. If axial fluence variation is needed, this can be estimated from the axial core power shape or an R,Z calculation can be carried out.

The forward transport calculation for each reactor model is carried out in R,θ geometry using the DOT two-dimensional discrete ordinates code[6] and the SAILOR cross-section library[7]. The SAILOR library is a 47 energy group ENDF/B-IV based data set produced specifically for light water reactor applications. In these analyses, anisotropic scattering is treated with a P_3 (3rd order Legendre polynomial) expansion of the scattering cross-sections and the angular discretization is modeled with an S_8 order of angular quadrature (48 angular directions). Each model is of a one-eighth core segment containing the surveillance capsules. The source distribution is taken to be an average distribution with all fissions attributed to U-235.

All adjoint calculations are carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the SAILOR library. The adjoint calculations are run in R,θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest, in this case $\phi(E > 1.0\ \text{MeV})$.

Having the importance functions and appropriate core source distributions, the response of interest is calculated as:

$$R(r,\theta) = \int_{r'} \int_{\theta'} \int_{E} I(r',\theta',E)\ S(r',\theta',E)\ r'\ dr'\ d\theta'\ dE \qquad (2.2\text{-}1)$$

where:

$R(r,\theta)$	=	$\phi(E > 1.0\ \text{MeV})$ at radius r and azimuthal angle θ,
$I(r',\theta',E)$	=	Adjoint source importance function at radius r', azimuthal angle, θ', and neutron source energy E for the flux ($E > 1$ MeV) at location r, θ,
$S(r',\theta',E)$	=	Neutron source strength at core location r',θ' for energy E.

Although the adjoint importance functions used in the analysis are based on a response function defined by the threshold neutron flux, $\phi(E > 1.0\ \text{MeV})$, the results of the response calculations indicate that while the variation in fuel loading patterns significantly affects both the magnitude and spatial distribution of the neutron field, changes in the relative neutron energy spectrum are of second order. Thus, the adjoint calculations can be used, together with the neutron spectrum from the forward calculation, to determine other exposure parameters such as displacements per atom (dpa), and threshold dosimetry reaction rates.

In addition to locations at the inner surface of the reactor vessel, adjoint calculations are also carried out for dosimetry locations. These locations are chosen at the geometric center of each surveillance capsule and at cavity dosimetry locations. The capsule center location is used to determine the calculated average exposure of the surveillance specimens. For the Westinghouse capsule design, axial fluence variation is small and is neglected.

The calculated fluence is validated by comparison with benchmark and plant specific measurements, which may include dosimetry from one or more surveillance capsules and, for some plants, reactor cavity dosimetry. On the average, for

Westinghouse plants, surveillance capsule dosimetry indicates that the fluence calculations are biased low. For conservatism, the fluence values used for vessel analyses of pressure-temperature heatup and cooldown limit curves are adjusted by the average bias indicated by the dosimetry measurements for that plant. The methodology used for calculation of neutron fluences is basically consistent with the proposed Regulatory Guide on Reactor Vessel Fluence (DG-1025).

In general, pressure-temperature limits are generated for a particular EFPY (effective full power years) of plant operation. In some cases the fluence at the EFPY of interest is obtained directly from the dosimetry analysis. However, if the fluence is not available from the dosimetry analysis, the peak vessel inner radius fluence at the EFPY of interest is calculated as follows:

$$f = F \times C \times E \qquad (2.2\text{-}2)$$

Where:
- f = the peak vessel inner radius fluence at the EFPY of interest (n/cm^2 (E > 1.0 MeV))
- F = Best estimate peak flux at the pressure vessel inner radius (n/cm^2 - sec (E > 1.0 MeV))
- C = seconds per year = 3.16×10^7 sec/yr
- E = EFPY of interest

2.3 FRACTURE TOUGHNESS PROPERTIES

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of Appendix G, 10 CFR Part 50[4], as augmented by the additional requirements in subsection NB-2331 of Section III of the ASME B&PV Code[8]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 ("Fracture Toughness Requirements")[9] of the NRC Regulatory Standard Review Plan.

These fracture toughness requirements are used to determine the value of the reference nil-ductility transition temperature (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, IRT_{NDT}) and to calculate the adjusted reference temperature (ART) as described in Section 2.4. Two types of tests are required to determine a material's value of IRT_{NDT}: Charpy V-notch impact (C_v) tests and drop-weight tests. The procedure is as follows:

1) Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.
2) At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb absorbed energy. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.
3) If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature T_{Cv} at which they are met. In this case the reference temperature RT_{NDT} = T_{Cv} - 60°F. Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).
4) If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb and 35 mils lateral expansion, a temperature representing a minimum of 50 ft-lb and 35 mils lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed as shown in Figure 2.1.

Plants that do not follow the fracture toughness requirements in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification must be provided for an exemption from the regulations to be granted by the NRC. For example, a study done by B&W Nuclear Technologies[11] used an alternative method to define the IRT_{NDT} value for WF-70 weld metal. Since use of the MTEB 5-2 procedures resulted in a wide scatter of IRT_{NDT} values for WF-70 weld metal, a Linde 80 low upper-shelf material, an alternative method based solely on drop-weight tests was defined for determining

IRT_{NDT} values of WF-70 weld metal. The use of this methodology was approved by the NRC[12] for Commonwealth Edison Company's Zion Units 1 and 2.

2.4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

The adjusted reference temperature (ART) for each material in the beltline region is calculated in accordance with Regulatory Guide 1.99, Revision 2[3]. The most limiting ART values (i.e., highest value at 1/4t and 3/4t locations) are used in determining the pressure-temperature limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (2.4\text{-}1)$$

IRT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[8] and calculated per Section 2.3. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for that class of material can be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the shift in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \qquad (2.4\text{-}2)$$

CF (°F) is the chemistry factor and is a function of copper and nickel content. CF is given in Table 1 of Reference 3 for weld metal and in Table 2 in Reference 3 for base metal (Position 1.1 of Regulatory Guide 1.99, Revision 2). In Tables 1 and 2 of Reference 3 "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in Regulatory Guide 1.99, Revision 2, Paragraph B.4) become available they may be used to calculate the chemistry factor per Position 2.1 of Regulatory Guide 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n} [A_i \times f_i^{(0.28-0.10\log fi)}]}{\sum_{i=1}^{n} [f_i^{(0.28-0.10\log fi)}]^2} \tag{2.4-3}$$

Where "n" is the number of surveillance data points, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point.

If Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a higher value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of Regulatory Guide 1.99, Revision 2, results in a lower value of ART than Position 1.1 of Regulatory Guide 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4t), the following formula is used to attenuate the fast neutron fluence (E> 1 MeV) at the specified depth.

$$f = f_{surface} * e^{(-0.24x)} \tag{2.4-4}$$

where $f_{surface}$(n/cm^2, E> 1 MeV) is the value, calculated per Section 2.2, of the neutron fluence at the base metal surface of the vessel at the location of the postulated defect, and x (in inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then put into equation (2.4-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by equation (2.4-2) plus two standard deviations ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by Appendix G to 10 CFR Part 50[4]. Margin is calculated by the following equation:

$$\text{Margin} = 2\sqrt{(\sigma_I^2 + \sigma_\Delta^2)} \qquad (2.4\text{-}5)$$

σ_I is the standard deviation for IRT_{NDT} and σ_Δ is the standard deviation for ΔRT_{NDT}. If IRT_{NDT} is a measured value, σ_I is estimated from the precision of the test method ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per Regulatory Guide 1.99, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

2.5 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME Code requirements[5] for calculating the allowable pressure-temperature limit curves for various heatup and cooldown rates specify that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ia}, for the metal temperature at that time. K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G, to Section XI of the ASME Code[5]. (Note that in Appendix G, to Section III of the ASME Code, the reference fracture toughness is denoted as K_{IR}, whereas in Appendix G of Section XI, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.) The K_{Ia} curve is given by the following equation:

$$K_{Ia} = 26.78 + 1.223 * \exp[0.0145 (T - RT_{NDT} + 160)] \quad (2.5\text{-}1)$$

where,

K_{Ia} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility transition temperature RT_{NDT}, (ksi $\sqrt{in}$). The value of RTNDT is the adjusted reference temperature (ART) of Section 2.4.

The governing equation for generating pressure-temperature limit curves is defined in Appendix G of the ASME Code[5] as follows:

$$C * K_{IM} + K_{IT} < K_{Ia} \quad (2.5\text{-}2)$$

where,

K_{IM} = stress intensity factor caused by membrane (pressure) stress,

K_{IT} = stress intensity factor caused by the thermal gradients through the vessel wall,

C = 2.0 for Level A and Level B service limits (for heatup and cooldown),

C = 1.5 for hydrostatic and leak test conditions when the reactor core is not critical

(Note: K_{IT} is set to zero for hydrostatic and leak test calculations since these tests are performed at isothermal conditions).

At specific times during the heatup or cooldown transient, K_{Ia} is determined by the metal temperature at the tip of the postulated flaw (the postulated flaw has a depth of one-fourth of the section thickness and a length of 1.5 times the section thickness per ASME Code, Section XI, paragraph G-2120), the appropriate value for RT_{NDT} at the same location, and the reference fracture toughness equation (2.5-1). The thermal stresses resulting from the temperature gradients through the vessel wall and the corresponding (thermal) stress intensity factor, K_{IT}, for the reference flaw are calculated as described in Section 2.6. From Equation (2.5-2), the limiting pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated as described in Section 2.6.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference 1/4t (t = reactor vessel wall thickness) flaw of Appendix G, Section XI to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the vessel wall because the thermal gradients that increase with increasing cooldown rates produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw. Allowable pressure-temperature curves are generated for steady-state (zero rate) and each finite cooldown rate specified. From these curves, composite limit curves are constructed as the minimum of the steady-state or finite rate curve for each cooldown rate specified.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4t vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the temperature difference across the wall developed during cooldown results in a higher value of K_{Ia} at the 1/4t location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ia} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4t location and, therefore, allowable pressures could be lower if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4t flaw at the inside of the wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ia} for the inside 1/4t flaw during heatup is lower than the K_{Ia} for the same flaw during steady-state conditions at the same coolant temperature. However, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ia}'s do not offset each other and the pressure-temperature curve based on finite heatup rates could become limiting. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature, the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained for the inside 1/4t flaw.

The third portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case of a 1/4t outside surface flaw. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and coolant temperature during the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate is analyzed on an individual basis.

Following the generation of the three pressure-temperature curves, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state data and finite heatup rate data for both inside and outside surface flaws. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is not possible to predict which condition is most limiting because of local differences in irradiation (RTNDT), metal temperature and thermal stresses. With the composite curve, the pressure limit is at all times based on analysis of the most critical situation.

Finally, the 1983 Amendment to 10CFR50[4] has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20 percent of the preservice hydrostatic test pressure. In addition, when the core is critical, the pressure-temperature limits for core operation (except for low power physics tests) require that the reactor vessel 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 pressure-temperature curve for heatup and cooldown. These limits are incorporated into the pressure-temperature limit curves wherever applicable.

Figure 2.2 shows an example of a heatup curve using a heatup rate of 60°F/Hr applicable for the first 16 EFPY. Figure 2.3 shows an example of cooldown curves using rates of 0, 20, 40, 60, and 100°F/Hr applicable for the first 16 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 2.2 and 2.3. Note that the step in these curves are due to the previously described flange requirements [4].

2.6 PRESSURE-TEMPERATURE CURVE GENERATION METHODOLOGY

2.6.1 Thermal and Stress Analyses

The time-dependent temperature solution utilized in both the heatup and cooldown analysis is based on the one-dimensional transient heat conduction equation

$$\rho C\frac{\partial T}{\partial t} = K\left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r}\right] \qquad (2.6.1\text{-}1)$$

with the following boundary conditions applied at the inner and outer radii of the reactor vessel,

at $r = r_i$, $$-K\frac{\partial T}{\partial r} = h(T - T_c) \qquad (2.6.1\text{-}2)$$

at $r = r_o$, $$\frac{\partial T}{\partial r} = 0 \qquad (2.6.1\text{-}3)$$

where,

r_i = reactor vessel inner radius
r_o = reactor vessel outer radius
ρ = material density
C = material specific heat
K = material thermal conductivity
T = local temperature
r = radial location
t = time
h = heat transfer coefficient between the coolant and the vessel wall
T_c = coolant temperature

These equations are solved numerically to generate the position and time-dependent temperature distributions, T(r,t), for all heatup and cooldown rates of interest.

With the results of the heat transfer analysis as input, position and time-dependent distributions of hoop thermal stress are calculated using the formula for the thermal stress in a hollow cylinder given by Timoshenko[14].

$$\sigma_\theta(r,t) = \frac{E\alpha}{1-\nu}\frac{1}{r^2}\left[\frac{r^2 + r_i^2}{r_o^2 - r_i^2}\int_{r_i}^{r_o} T(r,t)r\,dr + \int_{r_i}^{r} T(r,t)r\,dr - T(r,t)r^2\right] \qquad (2.6.1\text{-}4)$$

where,

$\sigma_\theta(r,t)$	=	hoop stress at location and time t
E	=	modulus of elasticity
α	=	coefficient of linear expansion
ν	=	Poisson's ratio

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α are evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r)\,r\,dr}{r_o^2 - r_i^2} \qquad (2.6.1\text{-}5)$$

E and α are calculated as a function of this equivalent temperature and the $E\alpha$ product in equation (2.6.1-4) is treated as a constant in the computation of hoop thermal stress.

The linear bending (σ_b) and constant membrane (σ_m) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in Appendix A, to Section XI of the ASME Code[15]. These stress components are used for determining the thermal stress intensity factors, K_{IT}, as described in the following subsection.

2.6.2 Steady-State Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at the 1/4t location where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius.

The allowable pressure $P(T_c)$ is a function of coolant temperature, and the pressure temperature curve is calculated for the steady state case at the assumed 1/4t inside surface flaw. First, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_{Ia}(T - RT_{NDT})_{1/4t}}{2.0} \quad (2.6.2\text{-}1)$$

where,

$K_{Ia}(T\text{-}RT_{NDT})$ = allowable reference stress intensity factor as a function of $T\text{-}RT_{NDT}$ at 1/4t.

Next, the maximum allowable pressure stress is determined using an iterative process and the following three equations:

$$Q = \phi^2 - 0.212\left(\frac{\sigma_p}{\sigma_y}\right)^2 \quad (2.6.2\text{-}2)$$

$$\sigma_p = \frac{K_{IM(max)}}{1.1 M_K \sqrt{\frac{\pi a}{Q}}} \quad (2.6.2\text{-}3)$$

$$K_{IP} = 1.1 M_K \sigma_p \sqrt{\frac{\pi a}{Q}} \quad (2.6.2\text{-}4)$$

where,

Q = flaw shape factor modified for plastic zone size[16],

ϕ = is the elliptical integral of the 2nd kind ($\phi = 1.11376$ for the fixed aspect ratio of 3 of the code reference flaw)[16],

0.212 = plastic zone size correction factor[16],

σ_p = pressure stress,

σ_y = yield stress,

1.1 = correction factor for surface breaking flaws,

M_K = correction factor for constant membrane stress[16], M_K as function of relative flaw depth (a/t) is shown in Figure 2.4,

a = crack depth of 1/4t,

K_{IP} = pressure stress intensity factor.

The maximum allowable pressure stress is determined by incrementing σ_p from an initial value of 0.0 psi until a pressure stress is found that computes a K_{IP} value within 1.0001 of the $K_{IM(max)}$ value. After the maximum allowable σ_p is found, the maximum allowable internal pressure is determined by

$$P(T_c) = \sigma_p \left[\frac{r_o^2 - r_i^2}{r_o^2 + r_i^2} \right] \qquad (2.6.2\text{-}5)$$

where,

$P(T_c)$ = calculated allowable pressure as a function of coolant temperature.

2.6.3 Finite Cooldown Rate Analyses

For each cooldown rate the pressure-temperature curve is calculated at the inside 1/4t location. First, the thermal stress intensity factor is calculated for a coolant temperature at a given time using the following equation from the Welding Research Council[16]:

$$K_{IT} = \left[\sigma_m 1.1\, M_K + \sigma_b M_B\right] \sqrt{\frac{\pi a}{Q}} \qquad (2.6.3\text{-}1)$$

where,

σ_m = constant membrane stress component from the linearized thermal hoop stress distribution,

σ_b = linear bending stress component from the linearized thermal hoop stress distribution,

M_K = correction factor for membrane stress[16], (see Figure 2.4),

M_B = correction factor for bending stress[16], M_B as a function of relative flaw depth (a/t) is shown in Figure 2.5.

The flaw shape factor Q in equation (2.6.2-6) is calculated from[16]:

$$Q = \phi^2 - 0.212\left(\frac{\sigma_m + \sigma_b}{\sigma_y}\right)^2 \qquad (2.6.3\text{-}2)$$

Once K_{IT} is computed, the maximum allowable membrane (pressure) stress intensity factor is determined using the factor of 2.0 from equation (2.5-2) and the following equation:

$$K_{IM(max)} = \frac{K_{Ia}(T - RT_{NDT})_{1/4t} - K_{IT}(T_c)_{1/4t}}{2.0} \qquad (2.6.3\text{-}3)$$

From $K_{IM(max)}$, the maximum allowable pressure is determined using the iterative process described above and equations (2.6.2-2) through (2.6.2-5).

The steady-state pressure-temperature curve of Section 2.6.2 is compared to the cooldown curves for the 1/4t inside surface flaw at each cooldown rate. At any time, the allowable pressure is the lesser of the two values, and the resulting curve is called the composite cooldown limit curve.

Finally, the 10 CFR Part 50[4] rule for closure flange regions is incorporated into the cooldown composite curve as described in Section 2.5.

2.6.4 Finite Heatup Rate Analyses

Using the calculated beltline metal temperature and the metal reference nil-ductility transition temperature, the reference stress intensity factor (K_{Ia}) is determined in Equation (2.5-1) at both the 1/4t and 3/4t locations where "t" represents the vessel wall thickness. At the 1/4t location, a 1/4 thickness flaw is assumed to originate at the vessel inside radius. At the 3/4t location, a 1/4t flaw is assumed to originate on the outside of the vessel.

For each heatup rate a pressure-temperature curve is calculated at the 1/4t and 3/4t locations. First, the thermal stress intensity factor is calculated at the 1/4t and 3/4t locations for a coolant temperature at a given time using equation (2.6.3-1) from Section 2.6.3.

Once K_{IT} is computed, the maximum allowable membrane (pressure) stress intensity factors at the 1/4t and 3/4t locations are determined using the following equations:

At 1/4t, $$K_{IM(max)1/4t} = \frac{K_{Ia}(T - RT_{NDT})_{1/4t} - K_{IT}(T_c)_{1/4t}}{2.0} \qquad (2.6.4\text{-}1)$$

At 3/4t, $$K_{IM(max)3/4t} = \frac{K_{Ia}(T - RT_{NDT})_{3/4t} - K_{IT}(T_c)_{3/4t}}{2.0} \qquad (2.6.4\text{-}2)$$

From $K_{IM(max)1/4t}$ and $K_{IM(max)3/4t}$, the maximum allowable pressure at both the 1/4t and 3/4t locations is determined using the iterative process described in Section 2.6.2 and equations (2.6.2-2) through (2.6.2-5).

As was done with the cooldown case, the steady state pressure-temperature curve of Section 2.6.2 is compared with the 1/4t and 3/4t location heatup curves for each heatup rate, with the lowest of the three being used to generate the composite heatup limit curve. The composite curve is then adjusted for the 10 CFR Part 50[4] rule for closure flange requirements.

2.6.5 Hydrostatic and Leak Test Curve Analyses

The minimum inservice hydrostatic leak test curve is determined by calculating the minimum allowable temperature at two pressure values (pressure values of 2000 psig and 2485 psig, approximately 110% of operating pressure, are generally used). The curve is generated by drawing a line between the two pressure-temperature data points. The governing equation for generating the hydrostatic leak test pressure-temperature limit curve is defined in Appendix G, Section XI, of the ASME Code[5] as follows:

$$1.5 * K_{IM} < K_{Ia} \qquad (2.6.5\text{-}1)$$

where, K_{IM} is the stress intensity factor caused by the membrane (pressure) stress and K_{Ia} is the reference stress intensity factor as defined in equation (2.5-1). Note that the thermal stress intensity factor is neglected (i.e. KIT=0) since the hydrostatic leak test is performed at isothermal conditions.

The pressure stress is determined by,

$$\sigma_p = \left[\frac{r_o^2 + r_i^2}{r_o^2 - r_i^2}\right] P \qquad (2.6.5\text{-}2)$$

where,

P = the input pressure (generally 2000 and 2485 psig)

Next, the pressure stress intensity factor is calculated for a 1/4t flaw by,

$$K_{IM} = \left[1.1\, M_K \sqrt{\frac{\pi a}{Q}}\right] \sigma_p \qquad (2.6.5\text{-}3)$$

The K_{IM} result is multiplied by the 1.5 factor of equation (2.5-2) and divided by 1000,

$$K_{HYD} = \frac{1.5\, K_{IM}}{1000} \qquad (2.6.5\text{-}4)$$

Finally, the minimum allowable temperature is determined by setting K_{HYD} to K_{Ia} in equation (2.5-1) and solving for temperature T:

$$T = \frac{\ln\left[\dfrac{(K_{HYD} - 26.78)}{1.223}\right]}{0.0145} + RT_{NDT} - 160.0 \qquad (2.6.5\text{-}5)$$

The 1983 Amendment to 10CFR50[3] has a rule which addresses the test temperature for hydrostatic pressure tests. This rule states that, when there is no fuel in the reactor vessel during hydrostatic pressure tests or leak tests, the minimum allowable test temperature must be 60°F above the adjusted reference temperature of the beltline region material that is controlling. If fuel is present in the reactor vessel during hydrostatic pressure tests or leak tests, the requirements of this section and Section 2.5 must be met, depending on whether the core is critical during the test.

2.7 Minimum Boltup Temperature

The minimum boltup temperature is equal to the material RT_{NDT} of the stressed region. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2. The Westinghouse position is that the minimum boltup temperature be no lower than 60° F. Thus, the minimum boltup temperature should be 60° F or the material RT_{NDT}, whichever is higher.

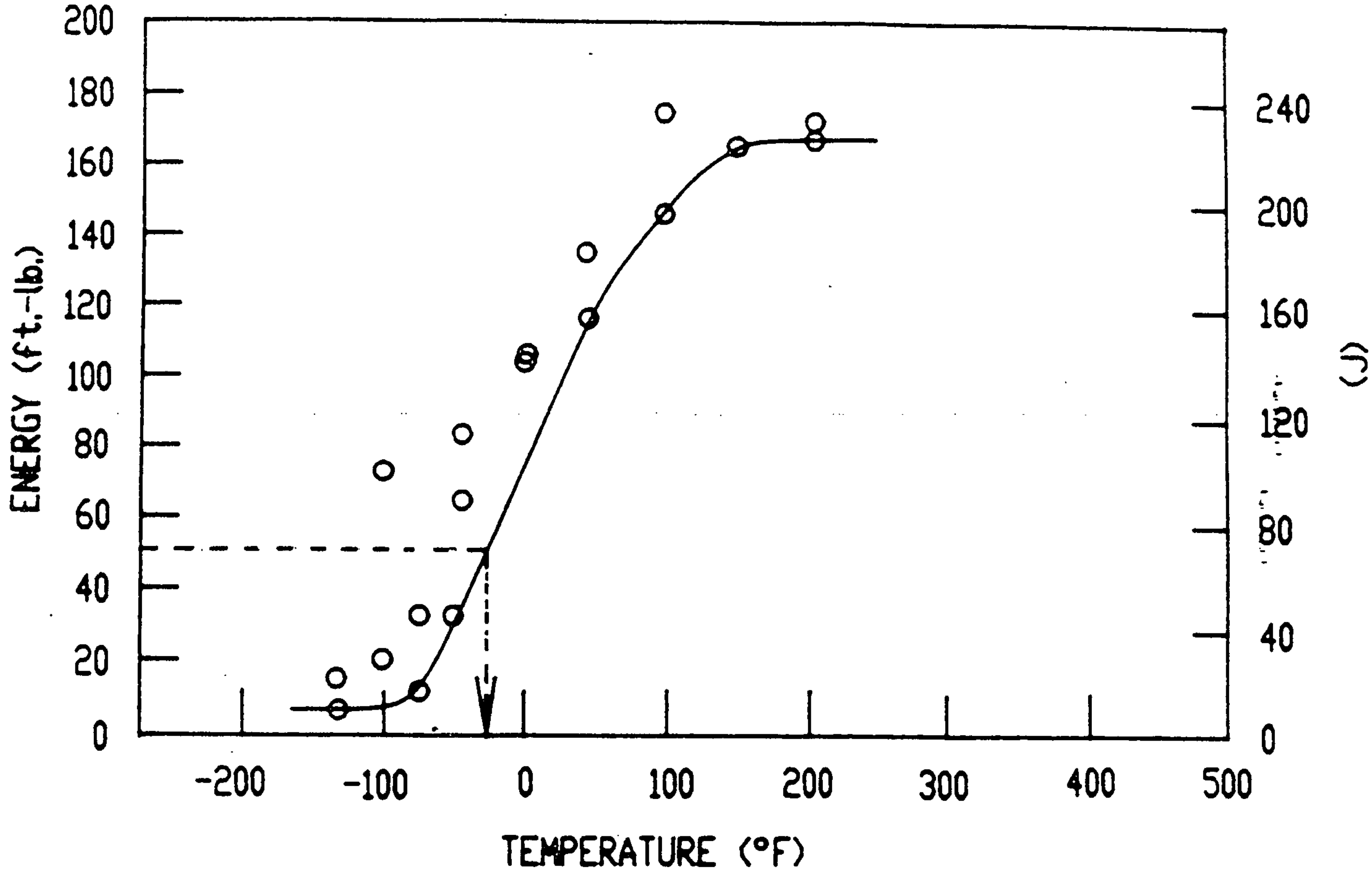


Figure 2.1: Example of a Charpy Impact Energy Curve Used to Determine IRT_{NDT} (Note: 35 mils lateral expansion is required at indicated temperature)

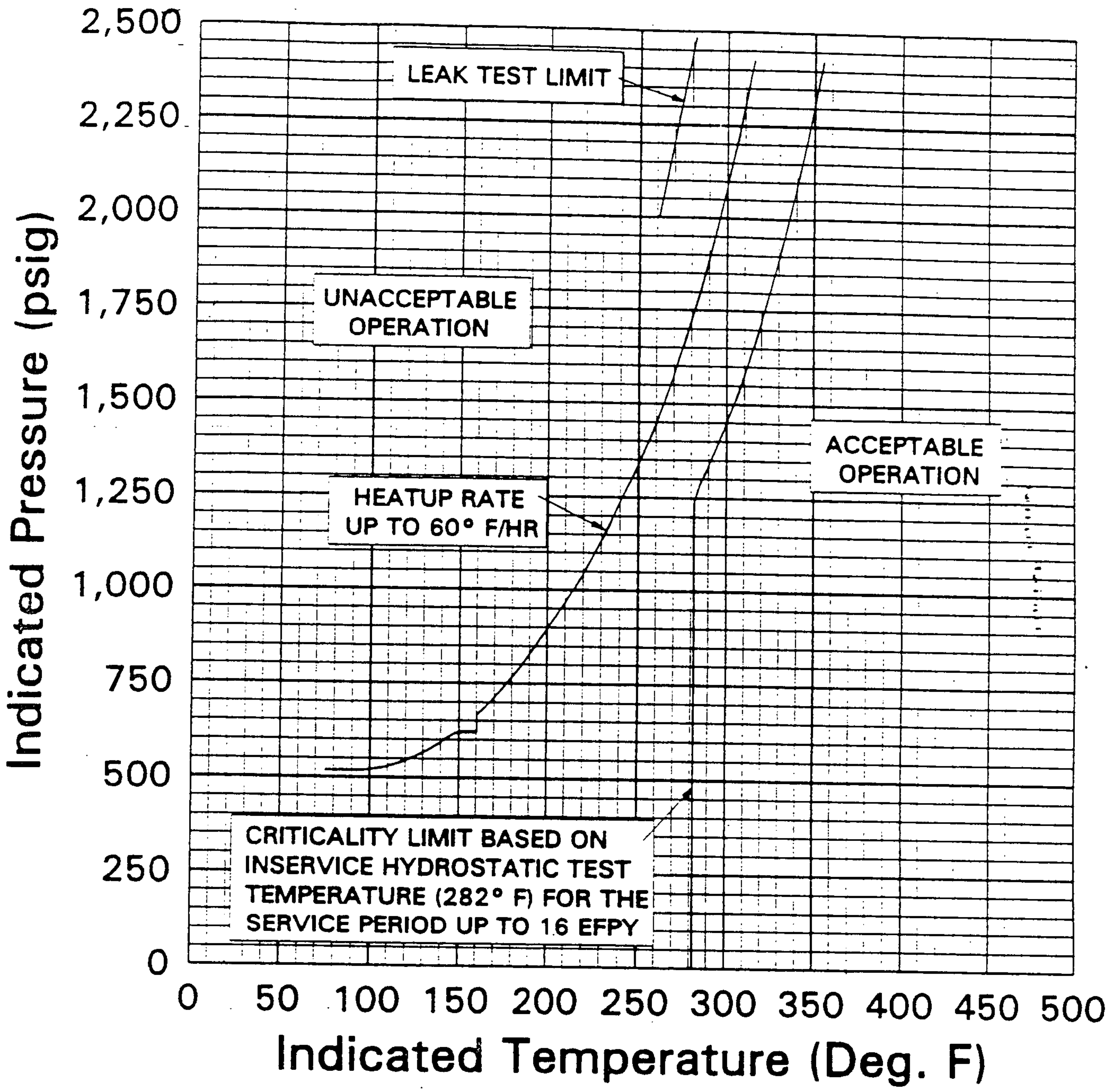


Figure 2.2: Heatup Pressure-Temperature Limit Curve For Heatup Rates up to 60 °F/Hr

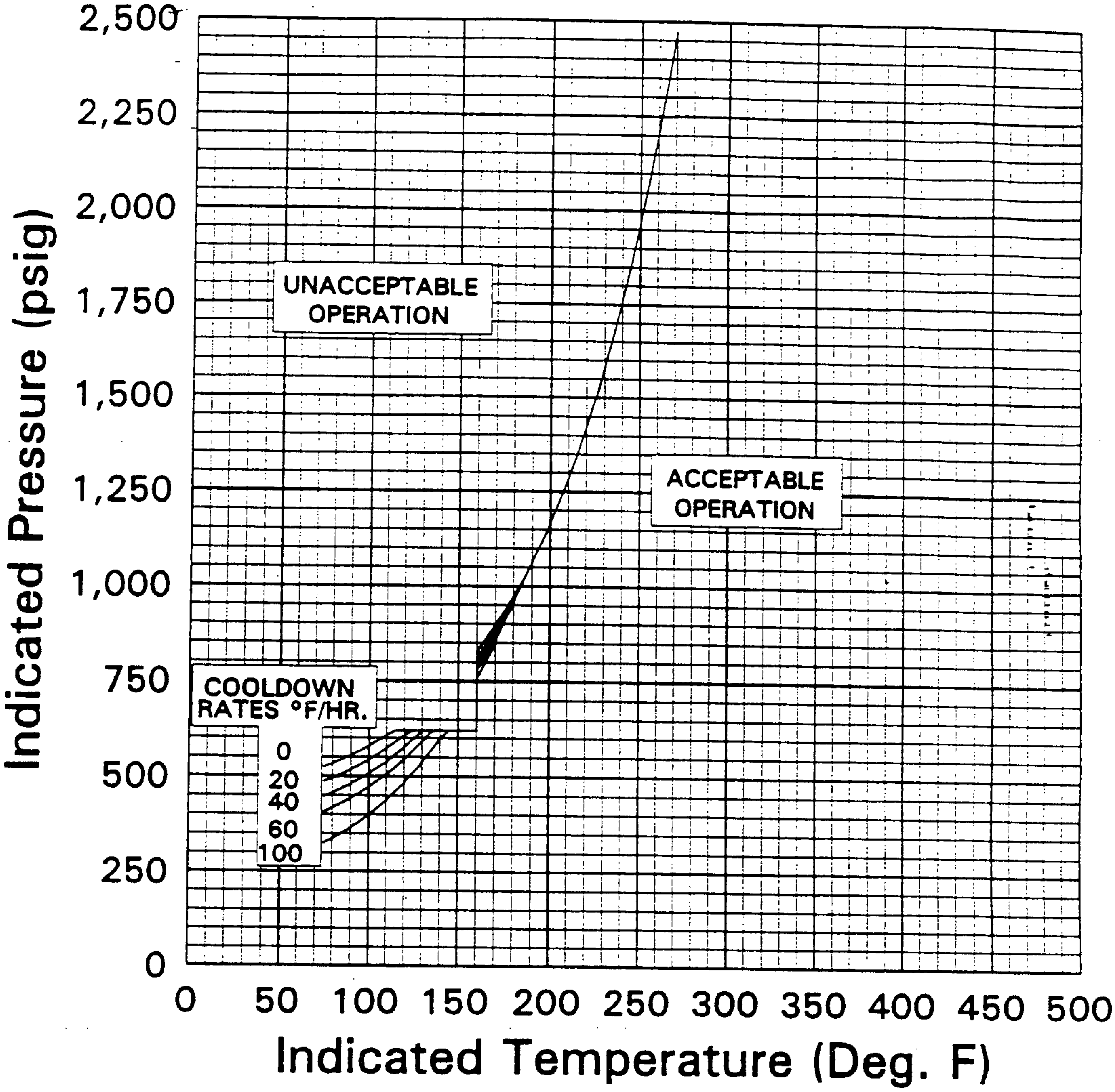


Figure 2.3: Cooldown Pressure-Temperature Limit Curves For Cooldown Rates up to 100°F/Hr

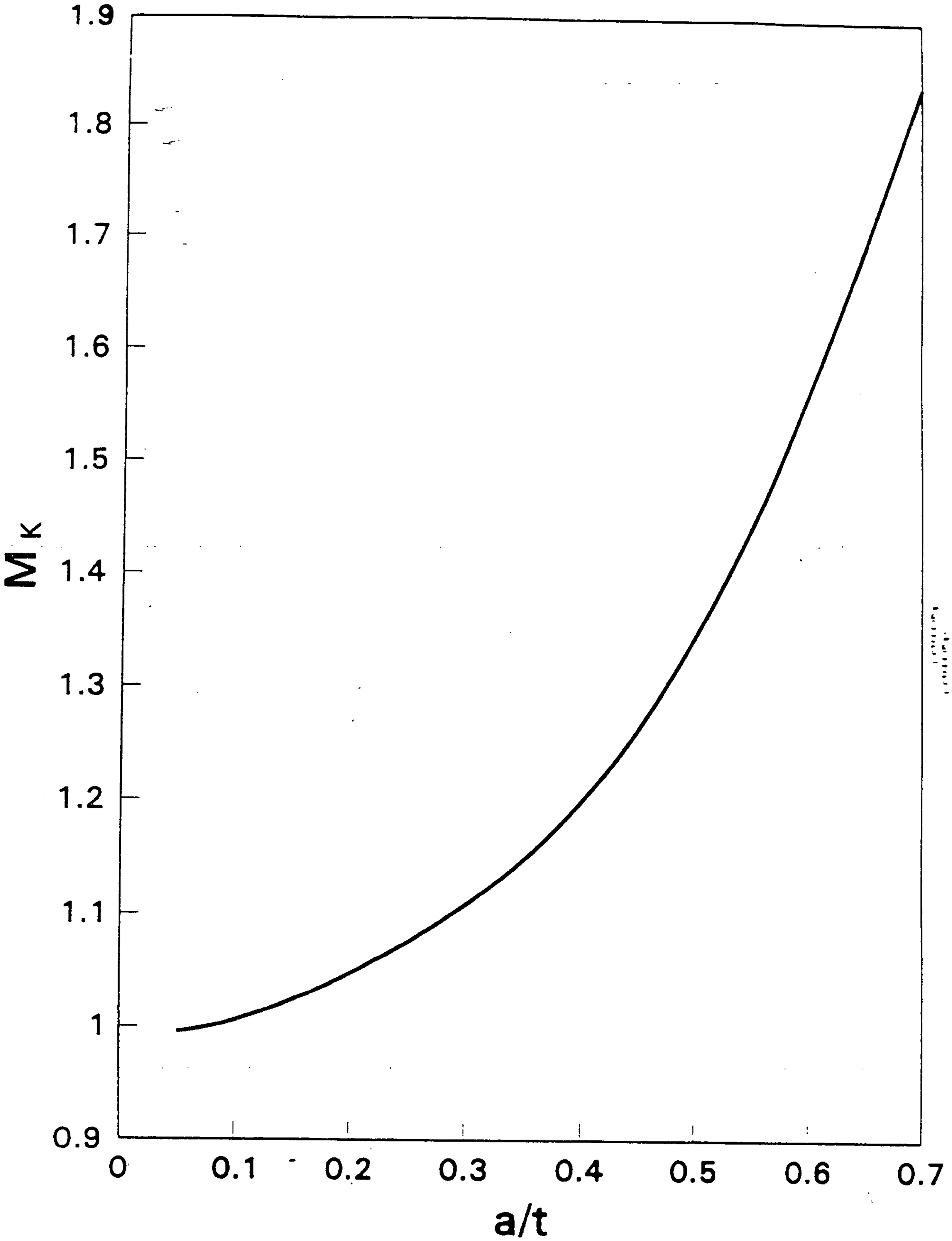


Figure 2.4: Membrane Stress Correction Factor (M_K) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6

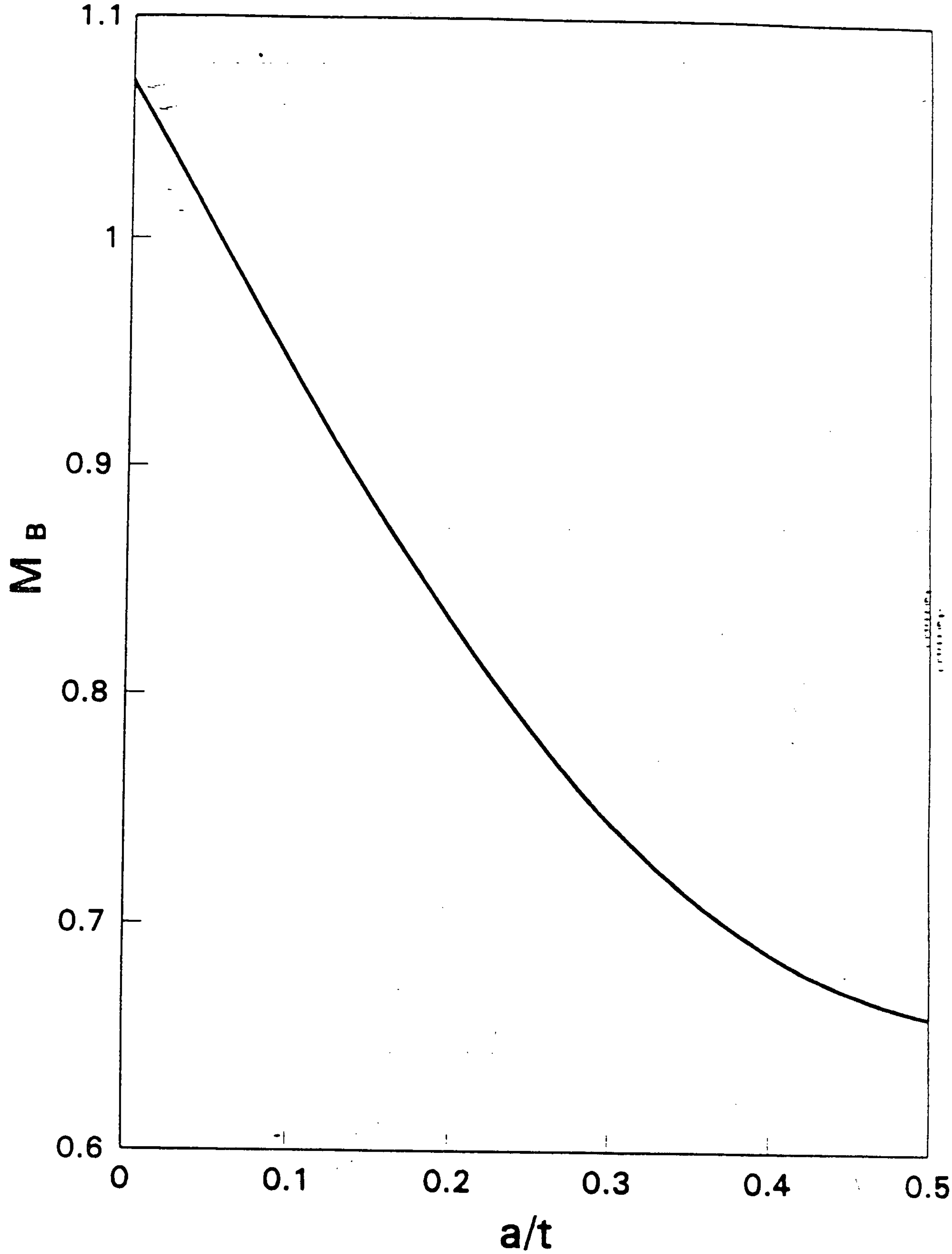


Figure 2.5: Bending Stress Correction Factor (M_B) vs. a/t Ratio for Flaws Having Length to Depth Ratio of 6.

3.0 COLD OVERPRESSURE MITIGATING SYSTEM (COMS)

3.1 INTRODUCTION

The purpose of the COMS is to supplement the normal plant operational administrative controls and the water relief valves in the Residual Heat Removal System (RHRS) when they are unavailable to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. This has been achieved by conservatively choosing COMS setpoints which prevent exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G[4] requirements. The COMS is desig ned to provide the capability, during relatively low temperature operation (typically less than 350°F), to automatically prevent the RCS pressure from exceeding the applicable limits. Once the system is enabled, no operator action is involved for the COMS to perform its intended pressure mitigation function. Thus, no operator action is modelled in the analyses supporting the setpoint selection, although operator action may be initiated to ultimately terminate the cause of the overpressure event.

The PORVs located near the top of the pressurizer, together with additional actuation logic from the wide-range pressure channels, are utilized to mitigate potential RCS overpressure transients defined below if the RHRS water relief valves are inadvertently isolated from the RCS. The COMS provides the supplemental relief capacity for specific transients which would not be mitigated by the RHRS relief valves. In addition, a limit on the PORV piping is accommodated due to the potential for water hammer effects to be developed in the piping associated with these valves as a result of the cyclic opening and closing characteristics during mitigation of an overpressure transient. Thus, a pressure limit more restrictive than the 10CFR50, Appendix G[4] allowable is imposed above a certain temperature so that the loads on the piping from a COMS event would not affect the piping integrity.

Two specific transients have been defined, with the RCS in a water-solid condition, as the design basis for COMS. Each of these scenarios assumes as an initial condition that the RHRS is isolated from the RCS, and thus the relief capability of the RHRS relief valves is not available. The first transient consists of a heat injection scenario in which a reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature and the RHRS has been inadvertently isolated. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into a water-solid RCS caused by the simultaneous isolation of the RHRS, isolation of letdown and failure of the normal charging flow controls to the full flow condition. Various combinations of charging and safety injection flows may also be evaluated on a plant-specific basis. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

3.2 COMS Setpoint Determination

Westinghouse has developed the following methodology which is employed to determine PORV setpoints for mitigation of the COMS design basis cold overpressurization transients. This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel integrity.

3.2.1 Parameters Considered

The selection of proper COMS setpoints for actuating the PORVs requires the consideration of numerous system parameters including:

a. Volume of reactor coolant involved in transient
b. RCS pressure signal transmission delay
c. Volumetric capacity of the relief valves versus opening position
d. Stroke time of the relief valves (open & close)
e. Initial temperature and pressure of the RCS
f. Mass input rate into RCS
g. Temperature of injected fluid

h. Heat transfer characteristics of the steam generators

i. Initial temperature asymmetry between RCS and steam generator secondary water

j. Mass of steam generator secondary water

k. RCP startup dynamics

l. 10CFR50, Appendix G pressure/temperature characteristics of the reactor vessel

m. Pressurizer PORV piping/structural analysis limitations

n. Dynamic and static pressure difference between reactor vessel midplane and location of wide range pressure transmitter

These parameters are input to a specialized version of the LOFTRAN computer code which calculates the maximum and minimum system pressures.

3.2.2 Pressure Limits Selection

The function of the COMS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing COMS setpoints which prevent exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50[4]. The COMS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. In addition, the COMS also provides for an operational consideration to maintain the integrity of the PORV piping. A typical characteristic 10CFR50 Appendix G curve is shown by Figure 3.1 where the allowable system pressure increases with increasing temperature. This type of curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties. Superimposed on this curve is the PORV piping limit which is conservatively used, for setpoint development, as the maximum allowable pressure above the temperature at which it intersects with the 10CFR50 Appendix G curve.

When a relief valve is actuated to mitigate an increasing pressure transient, the release of a volume of coolant through the valve will cause the pressure increase to be slowed and reversed as described by Figure 3.2. The system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signalled to close. Note that the pressure continues to decrease below the reset pressure as the valve recloses. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the reactor coolant pump #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement and the PORV piping limitations) and the nominal RCP #1 seal performance criteria create a pressure range from which the setpoints for both PORVs may be selected as shown on Figures 3.3 and 3.4.

3.2.3 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signalled to open at a specific pressure setpoint. However, as shown on Figure 3.2, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed. The maximum and minimum pressures reached (P_{MAX} and P_{MIN}) in the transient are a function of the selected setpoint (P_S) as shown on Figure 3.3. The shaded area represents an optimum range from which to select the setpoint based on the particular mass input case. Several mass input cases may be run at various input flow rates to bound the allowable setpoint range.

3.2.4 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e. the mass injection transient is not sensitive to temperature). The shaded area on Figure 3.4 describes the acceptable band for a heat input transient from which to select the setpoint for a particular initial reactor coolant temperature.

3.2.5 Final Setpoint Selection

By superimposing the results of multiple mass input and heat input cases evaluated, (from a series of figures such as 3.3 and 3.4) a range of allowable PORV setpoints to satisfy both conditions can be determined. Each of the two PORVs may have a different pressure setpoint versus temperature specification such that only one valve will open at a time and mitigate the transient (i.e. staggered setpoints). The second valve operates only if the first fails to open on command. This design supports a single failure assumption as well as minimizing the potential for both PORVs to open simultaneously, a condition which may create excessive pressure undershoot and challenge the RCP #1 seal performance criteria. However, each of the sets of staggered setpoints must result in the system pressure staying below the P_{MAX} pressure limit shown on Figures 3.3 and 3.4 when either valve is utilized to mitigate the transient.

The function generator used to program the pressure versus setpoint curves for each valve has a limited number of programmable break points (typically 9). These are strategically defined in the final selection process, with consideration given to the slope of any line segment, which is limited to approximately 24 psi/°F.

The selection of the setpoints for the PORVs considers the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50[4] or the peak RCS pressure based upon piping/structural analysis loads. The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. Since both the upper and lower pressure values are conservatively determined, the uncertainties in the pressure and temperature instrumentation utilized by the COMS are not explicitly accounted for in the selection of the COMS PORV setpoints. Accounting for the effects of instrumentation uncertainty would impose additional unnecessary restrictions on the setpoint development, which is already based on conservative pressure limits (such as a safety factor of 2 on pressure stress, use of a lower bound K_{IR} curve and an assumed ¼T flaw depth with a length equal to 1½ times the vessel wall thickness) as discussed in section 2 of this report, without a commensurate increase in the level of protection afforded to reactor vessel integrity.

3.3 Application of ASME Code Case N-514

ASME Code Case N-514[17] allows low temperature overpressure protection systems (LTOPS, as the code case refers to COMS) to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G, paragraph G-2215, of Section XI of the ASME Code[5]. (Note, that the setpoint selection methodology as discussed in Section 3.2.5 specifically utilizes the steady-state curve.) The application of ASME Code Case N-514 increases the operating margin in the region of the pressure-temperature limit curves where the COMS system is enabled. Code Case N-514 requires LTOPS to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. RT_{NDT} is the highest adjusted reference temperature for weld or base metal in the beltline region at a distance one-fourth of the vessel section thickness from the vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2. Although expected soon, use of Code Case N-514 has not yet been formally approved by the NRC. In the interim, an exemption to the regulations must be granted by the NRC before Code Case N-514 can be used in the determination of the COMS setpoints and enable temperature.

3.4 Enable Temperature for COMS

The enable temperature is the temperature below which the COMS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is also supported by Westinghouse and is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3. This definition is also supported by Westinghouse and can be used by requesting an exemption to the regulations or when ASME Code Case N-514 is formally approved by the NRC.

The RCS cold leg temperature limitation for starting an RCP is the same value as the COMS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the COMS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

FIGURE 3.1

TYPICAL APPENDIX G
P/T CHARACTERISTICS

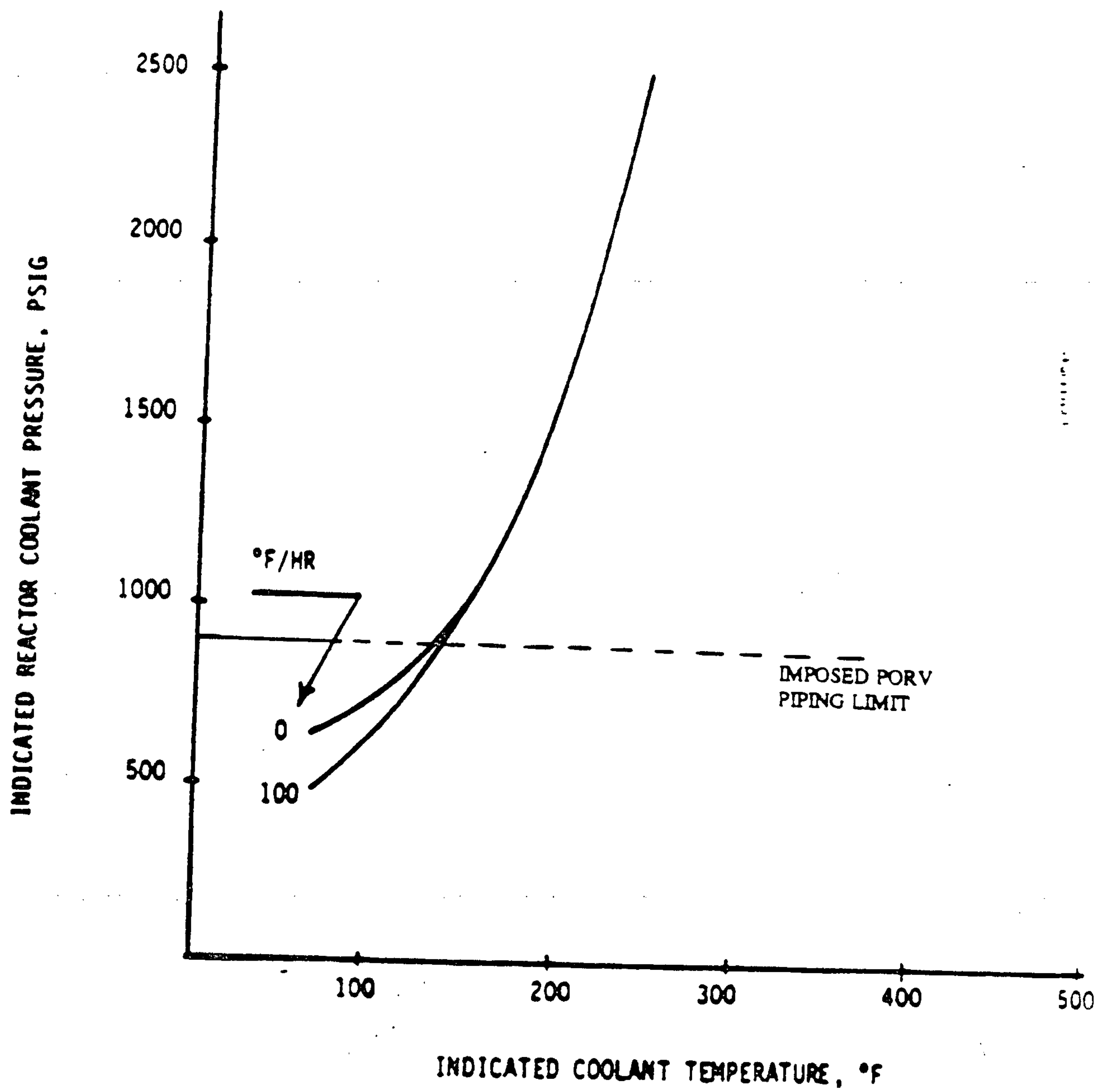


FIGURE 3.2

TYPICAL PRESSURE TRANSIENT

(1 RELIEF VALVE CYCLE)

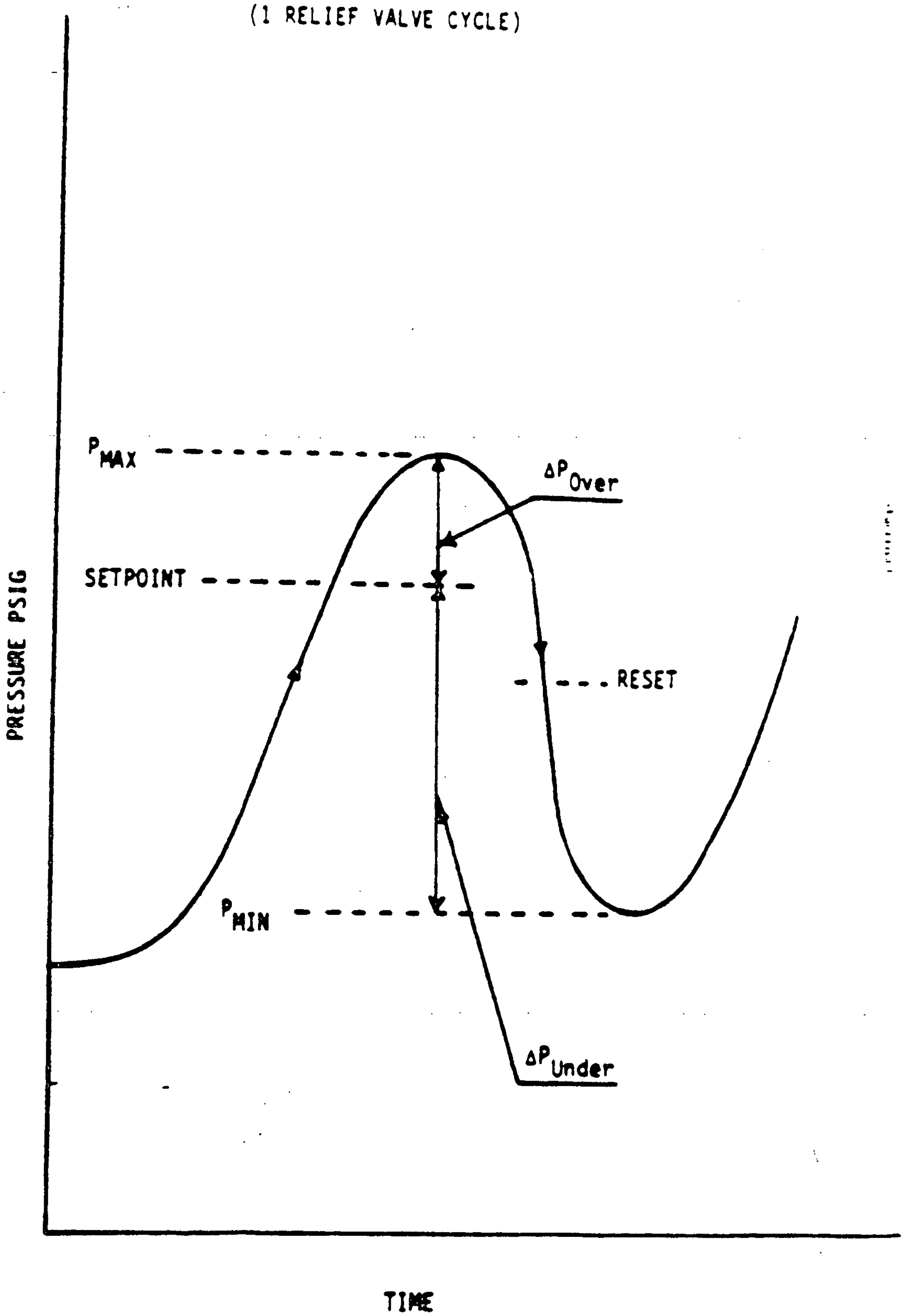
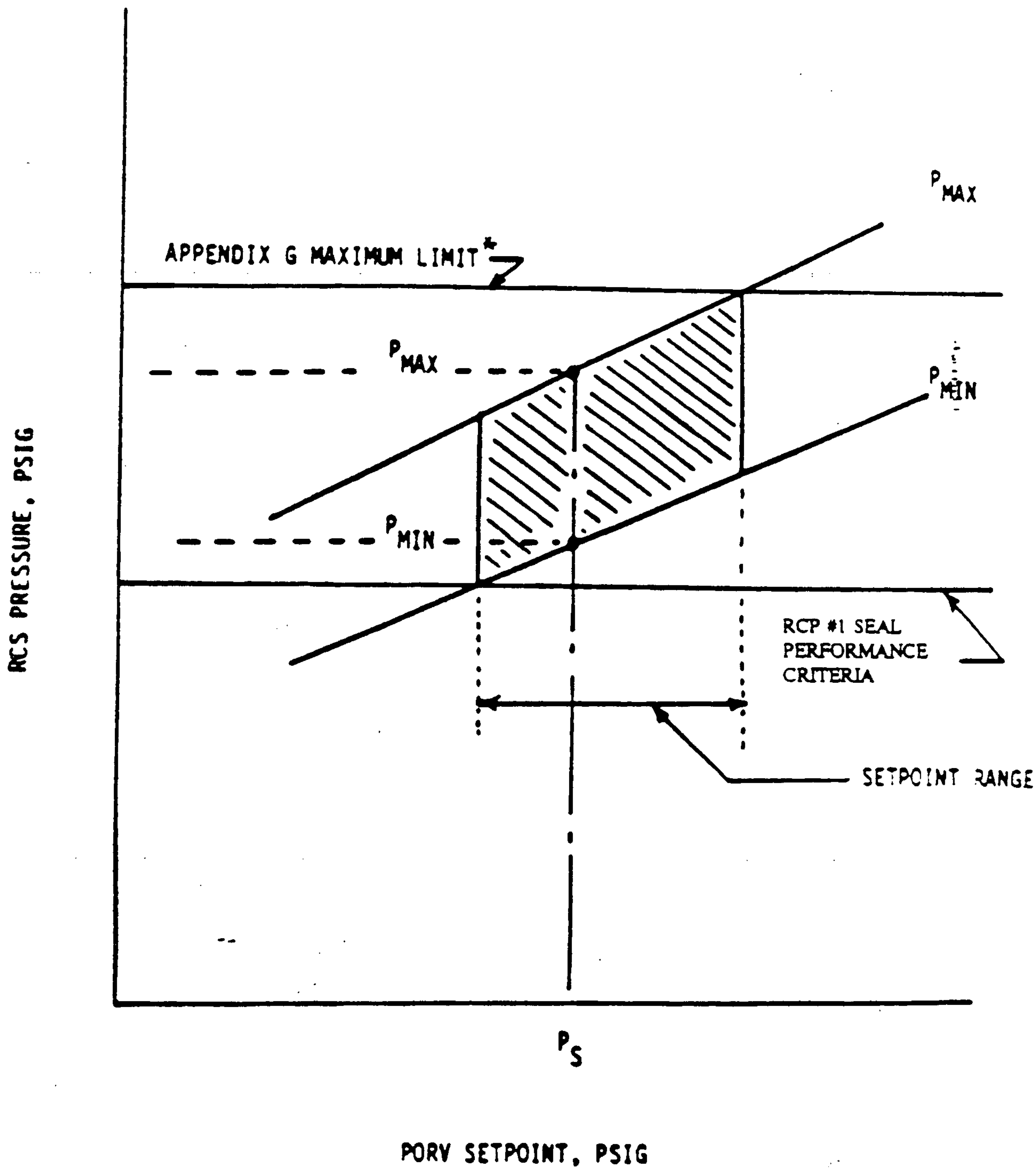


FIGURE 3.3

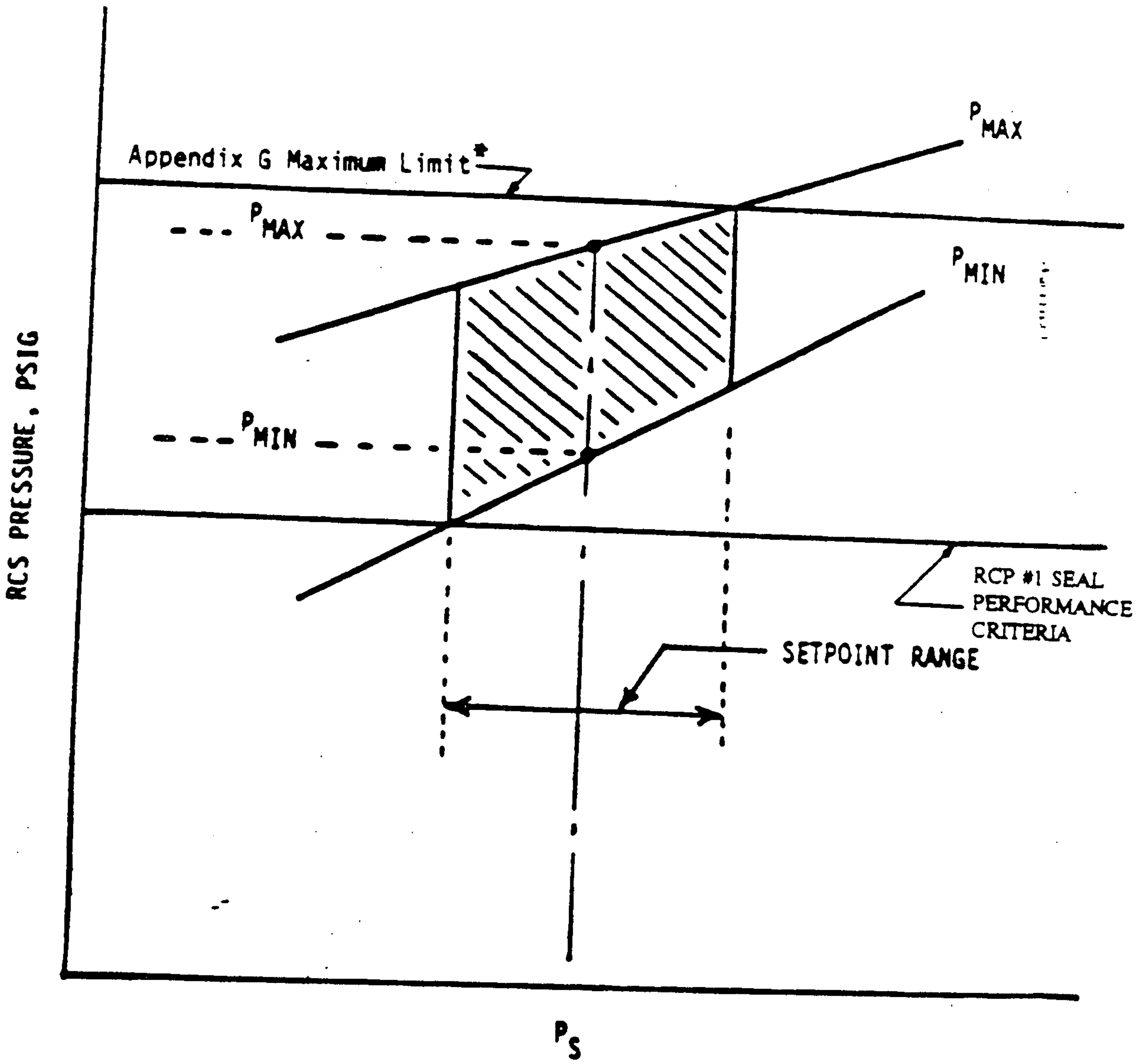
SETPOINT
DETERMINATION
(MASS INPUT)



* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

FIGURE 3.4

SETPOINT DETERMINATION (HEAT INPUT)



PORV SETPOINT, PSIG

* The maximum pressure limit is the minimum of the Appendix G limit or the PORV discharge piping structural analysis limit.

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors", Revision 0, September, 1992.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications", Generic Letter 88-16, October, 1988.

3. U.S. Nuclear Regulatory Commission, Radiation Embrittlement of Reactor Vessel Materials, <u>Regulatory Guide 1.99, Revision 2</u>, May, 1988.

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors", Appendix G, Fracture Toughness Requirements.

5. <u>ASME Boiler and Pressure Vessel Code</u>, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G, Fracture Toughness Criteria For Protection Against Failure.

6. R. G. Soltesz, R. K. Disney, J. Jedruch, and S. L. Ziegler, Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation. Vol. 5--Two-Dimensional Discrete Ordinates Transport Technique, WANL-PR(LL)-034, Vol. 5, August 1970.

7. ORNL RSIC Data Library Collection DLC-76 SAILOR Coupled Self-Shielded, 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors.

8. <u>ASME Boiler and Pressure Vessel Code</u>, Section III, "Rules for Construction of Nuclear Power Plant Components", Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements", NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

11. B&W Owners Group Report BAW-2202, "Fracture Toughness Characterization of WF-70 Weld Material", B&W Owners Group Materials Committee, September 1993.

12. Letter, Clyde Y. Shiraki, Nuclear Regulatory Commission, to D. L. Farrar, Commonwealth Edison Company, "Exemption from the Requirement to Determine the Unirradiated Reference Temperature in Accordance with the Method Specified in 10 CFR 50.61(b) (2) (i) (TAC NOS. M84546 and M84547)", Docket Nos. 50-295 and 50-304, February 22, 1994.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors", Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix A, Analysis of Flaws, Article A-3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials", Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section XI, Division 1, "Low Temperature Overpressure Protection", Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures", NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

ENCLOSURE 4

PROPOSED FSAR CHANGE

(MARKUP OF FSAR PAGE 5.2-39)

5.2.4.3 Operating Limitations During Startup and Shutdown

Startup and shutdown operating limitations are based on the properties of the core region materials of the reactor pressure vessel[6]. Actual material property test data is used. The methods outlined in Appendix G to Section III of the ASME Code will be employed for the shell regions in the analysis of protection against non-ductile failure. The initial operating curves are calculated assuming a period of reactor operation such that the beltline material will be limiting. The heatup and cooldown curves are given in the Pressure and Temperature Limits Report (PTLR) required by the technical specifications. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil-ductility temperature which includes a reference nil-ductility temperature shift (ΔRT_{NDT}).

Predicted ΔRT_{NDT} values are derived using two curves: the effect of fluence and copper and nickel content on the shift of ΔRT_{NDT} for the reactor vessel steels exposed to 550°F temperature curve, and the maximum fluence at 1/4 T (thickness) and 3/4 T locations (tips of the code reference flaw when the flaw is assumed at inside diameter and outside diameter locations, respectively) curve. These curves are presented in the ~~technical specifications~~ PTLR. For a selected time of operation, this shift is assigned a sufficient magnitude so that no unirradiated ferritic materials in other components of the reactor coolant system are limiting in the analysis. Based on copper and nickel content and initial RT_{NDT}, the intermediate forgings are determined to be limiting for Unit 1 and to a fluence of 1.5 x 10^{19} n/cm^2 (E > 1.0 MeV) for Unit 2. Above this fluence value, the circumferential weld seam between the lower and intermediate forging is controlling for Unit 2.

The operating curves including pressure-temperature limitations are calculated in accordance with methodology referenced in the PTLR. ~~10 CFR Part 50, Appendix G and ASME Code Section III, Appendix G, requirements as modified by code case N-514~~. Changes in fracture toughness of the core region forgings, weldments and associated heat affected zones due to radiation damage will be monitored by a surveillance program which is based on ASTM E-185-82, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels" and 10 CFR Part 50, Appendix H. The Reactor Vessel Irradiation Surveillance Program is in compliance with these documents with the exception that four of the six reactor vessel irradiation surveillance capsules will receive a fluence which is 3.6 times the maximum reactor vessel fluence. The above documents require that the capsule to vessel maximum fluence not exceed a lead factor of 3.0. At the time of the design of the surveillance program, all capsules were positioned as near to the vessel wall as possible and were limited to a fluence less than 3 times the vessel fluence. Recently a more accurate method of calculating vessel and capsule fluence has been developed which results in a lead factor of 3.6 for four of the capsules which are in violation of the above documents. This violation is not considered to be of any significant consequence since the test results from the encapsulated specimens will represent the actual behavior of the material in the vessel and, therefore, the evaluation of the effects of radiation on the actual vessel material will not be influenced by the larger lead factor.

The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and 1/2 T compact tension specimens. The post-irradiation testing will be carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core midheight and removable from the vessel at specified intervals.

ENCLOSURE 5

LIST OF COMMITMENTS

1. TVA will revise the Final Safety Analysis Report (FSAR) to address the use of the Pressure Temperature Limits Report (PTLR) in accordance with the Technical Specifications.

2. TVA will evaluate the continued applicability of the LTOP setpoints prior to exceeding 1.5 EFPY.