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William J. Museler
Site Vice President, Watts Bar Nuclear Plant

MAR 10 1994

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of the Application of	)	Docket Nos. 50-390
Tennessee Valley Authority	)	50-391

WATTS BAR NUCLEAR PLANT (WBN) - WATTS BAR NUCLEAR PLANT (WBN) - REQUEST FOR EXEMPTION FROM 10 CFR 50.60, "ACCEPTANCE CRITERIA FOR FRACTURE PREVENTION MEASURES FOR LIGHTWATER NUCLEAR POWER REACTORS FOR NORMAL OPERATION"

The purpose of this letter is to request, in accordance with the provisions of 10 CFR 50.12, an exemption from certain requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation" for WBN Unit 1 and Unit 2.

Paragraph 10 CFR 50.60 states that all lightwater nuclear power reactors must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary as set forth in Appendices G and H to 10 CFR Part 50. Paragraph 10 CFR 50.60 also specifies that proposed alternatives to the described requirements of 10 CFR Part 50 Appendices G and H may be used when an exemption is granted by NRC under 10 CFR 50.12.

This exemption is requested to allow the application of American Society of Mechanical Engineers (ASME) Code Case N-514, "Low Temperature Overpressure Protection," in determining the acceptable low temperature overpressure protection (LTOP) setpoints for WBN and is similar to that recently granted for the Sequoyah Nuclear Plant.

Enclosed is the subject exemption with a detailed discussion of the application of ASME Code Case N-514 to WBN's LTOP setpoints. NRC review and response are requested to support the current Unit 1 schedule for requiring LTOP operability before reactor coolant system venting.

TVA will revise the Final Safety Analysis Report (FSAR) to address the use of ASME Code Case N-514.

D030

MAR 10 1994

If you should have any questions, contact P. L. Pace at (615)-365-1824.

Very truly yours,

William J. Museler

Enclosure
cc (Enclosure):
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ENCLOSURE 1

Discussion

Pressure/temperature (P/T) limits for low temperature overpressure protection (LTOP) events can be characterized by two parameters: (1) the system enabling temperature and (2) setpoint pressure for the pressure relieving device. According to current regulatory guidelines, the LTOP system must be enabled at temperatures less than or equal to RT_{NDT} (Reference nil-ductility transition temperature + 90°F, where RT_{NDT} is the Regulatory Guide 1.99 adjusted reference temperature, including margin, at the one quarter thickness location. At temperature greater than RT_{NDT} + 90°F, LTOP need not be provided. The maximum LTOP system pressure is determined based on system-specific considerations, but is chosen so that the maximum pressure attained in the vessel will not exceed the P/T limit curve defined by Appendix G to American Society of Mechanical Engineers (ASME) Sections III and XI and Appendix G to 10 CFR Part 50.

Current LTOP limits produce operational constraints by limiting the range available to the operator to heat-up and cool-down the plant. The "operating window" for heat-up or cool-down of the reactor coolant system (RCS) is determined from ASME Section XI, Appendix G and the reactor coolant pump (RCP) seal pressure, adjusted for LTOP system overshoot/undershoot and temperature errors.

Operation of the LTOP system without relief would impose a significant impact by overly restricting plant operation during heat-up and cool-down. In addition, the narrow operating window can have an adverse impact if it increases the possibility of unnecessary actuations of the LTOP system.

The Working Group on Operating Plant Criteria (WGOPC), which has responsibility for Appendix G to Section XI, has considered the burden and safety significance imposed by the current LTOP criteria, and has recommended guidelines in ASME approved Code Case N-514 (expected to soon be approved by the NRC). Adequate margins against failure for the reactor pressure vessel are still provided but, by relaxing this operational restriction, these guidelines result in a reduced potential for unnecessary activation of the PORVs as the plant is heated up or cooled down.

The P/T limits are contained in a Pressure and Temperature Limits Report (PTLR) as required by the Technical Specifications. TVA provided a sample draft PTLR by letter dated August 27, 1992 and committed to provide an issued PTLR before Unit 1 fuel load. The attachments to this enclosure contains the issued RCS PTLRs for Unit 1 and Unit 2 utilizing Code Case N-514 allowables.

Code Case N-514

ASME Code Case N-514 allows setting the LTOP actuation setpoints such that the Appendix G curves are not exceeded by more than 10 percent at coolant temperatures < 200°F or at coolant temperatures corresponding to a Reactor Vessel metal temperature < RT_{NDT} + 50°F which ever is greater. The application of this

code case to Watts Bar Nuclear Plant (WBN) Units 1 and 2 would allow LTOP operation with the newly established Power Operated Relief Valves (PORV) setpoints.

WGOPC has developed code guidelines to define LTOP limits that will avoid certain unnecessary operational restrictions, provide adequate margins against failure, and reduce the potential for unnecessary operational restrictions, and reduce the potential for unnecessary activation of pressure-relieving devices used for LTOP.

The philosophy used by the WGOPC for developing these guidelines is to ensure that the LTOP limits are still below the P/T limits for normal operation, but allow the pressure excursions that may occur while the reactor is operating at low temperatures to slightly exceed the P/T limits, provided acceptable margins are maintained during these events. This philosophy protects the pressure vessel from LTOP events, and still maintains the Technical Specification P/T limits applicable for normal heat-up and cool-down in accordance with Appendix G to 10 CFR Part 50 and Sections III and XI of the ASME Code.

Inherent Margins to 10 CFR Part 50 Appendix G

There are numerous conservatisms in the development of Appendix G pressure and/or temperature curve calculations. Some of these are as follows:

1. The safety factor of 2 on the principal membrane (pressure) stresses.

2. Two standard deviation (2-sigma) margin for uncertainty applied to RT_{NDT} by using Regulatory Guide 1.99, Revision 2.

3. Pressure and thermal stress intensity factors based on a flaw depth 1/4 the vessel wall thickness and a 6 to 1 length to depth ratio.

These conservatisms support the determinations made by the WGOPC in the development of Code Case N-514. The subject code case allows pressure limits to be defined during low temperature operations utilizing a methodology consistent with the guidelines developed by the WGOPC.

Basis for Exemption

The requested exemption to the regulations as authorized, will not present an undue risk to the public health and safety.

TVA believes the requested exemption meets the criteria in 10 CFR 50.12(a)(2) in that special circumstances are present. These special circumstances include:

1. 10 CFR 50.12(a)(2)(ii)

   The application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

Basis - The basis for the LTOP setpoints is to preclude RCS pressure from exceeding the Appendix G curves when there is a potential for brittle failure of reactor vessel material.

ASME Code Case N-514 recognizes the conservatism of the Appendix G curves and allows establishing a setpoint that preserves the acceptable margin of safety while maintaining operational margins for RCP operation at low temperatures and pressures. Setpoints established in accordance with Code Case N-514 will also minimize the unnecessary actuation of protection system pressure-relieving devices. Therefore, establishing LTOP setpoints using Code Case N-514 criteria satisfies the underlying purpose of the ASME Code and NRC's regulations to ensure nuclear power plant systems and components are operated to ensure an acceptable level of safety and environmental impact.

Based on the above, the application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule.

2. 10 CFR 50.12(1)(2)(iii)

Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

Basis - Administrative restrictions of RCP operations while at low RCS temperatures would result in an unnecessary burden in that an excessive delay would be required to ensure minimum RCS temperature before starting RCPs. TVA believes that this burden can be minimized by the application of the code case. The guidelines developed by the WGOPC for LTOP P/T limits provide acceptable margin against crack initiation and failure in reactor vessels. These limits do not significantly change the likelihood of vessel failure associated with normal heat-up and cool-down P/T limits. Moreover, the LTOP guidelines will reduce the potential for unnecessary activation of PORVs. Consequently, the LTOP limits developed by WGOPC provide both economic and operational benefits.

Therefore, compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted.

Specifics

TVA has received from Westinghouse Electric Corporation the PORV setpoints for WBN Unit 1 (WAT-D-9448). These setpoints account for the Reactor Coolant System pressure differences between the wide range pressure transmitter, and the reactor vessel limiting beltline region, as well as the revised heat-up and cool-down limit curves which incorporate ASME Code Case N-514. As the pressure difference depends on the number of Reactor coolants pumps in operation, setpoints have been provided for 1, 2, 3, and 4 pumps in operation.

Setpoints are selected within a range of allowable pressures for nine different temperatures between 70°F and 450°F. The maximum allowable setpoint is bounded both by the Appendix G Limit at that temperature or the 800 psig PORV piping limit, whichever is smaller. The minimum allowable setpoint is by the pump seal limit which includes a 63 psig pressure channel uncertainty. Both the maximum and minimum setpoints include additional conservatism for temperature streaming and instrument uncertainty, as well as thermal effects due to heat transport. Two distinct setpoints have been provided for each temperature to prevent the PORVs from opening simultaneously. The tables below show the effect of Code Case N-514 on the margin between the maximum PORV setpoint and the Appendix G value for three selected temperatures.

WATTS BAR NUCLEAR PLANT - UNIT 1

WATER TEMP	HIGHEST SETPOINT	APPENDIX G-CURVE	EXISTING APP G MARGIN	CC N-514 ADDITION	TOTAL MARGIN
100°F	486	519	33	52	85
150°F	505	587.5	82.5	59	141
200°F	540	730	190	73	263

WATTS BAR NUCLEAR PLANT - UNIT 2

WATER TEMP	HIGHEST SETPOINT	APPENDIX G-CURVE	EXISTING APP G MARGIN	CC N-514 ADDITION	TOTAL MARGIN
100°F	565	711	146	71	217
150°F	575	983	408	98	506
190°F	715	1394	679	139	818

(1) All values not designated are rounded-off with units psig.

(2) The highest setpoint is based on PORV-334 (Westinghouse number 456) with one RCS Pump operating.

(3) The value listed under Appendix G-Curve is based on steady-state Appendix G Limits since most overpressure events occur under isothermal conditions.

(4) Current Appendix G curves based on 7 EFPY for Unit 1 and 32 EFPY for Unit 2.

Conclusion

ASME Code Case N-514 allows setting the LTOP setpoints such that the Appendix G curves are not exceeded by more than 10 percent. The ASME Code Committee has concluded the LTOP guidelines provide acceptable margin against crack initiation and failure in reactor vessels, and will reduce the potential or unnecessary activation of protection system pressure-relieving devices. Consequently, the current LTOP setpoints provide operational (economic) benefits with no adverse safety or environmental impact. TVA believes that use of Code Case N-514 provides an acceptable level of quality and safety and does not represent an undue risk to public health.

The inability to utilize Code Case N-514 in establishing P/T limits would result in an operating hardship to TVA, without a compensating increase in the level of quality or safety.

APPENDIX "A"

TO RCS SYSTEM DESCRIPTION N3-68-4001

WATTS BAR UNIT 1

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

REVISION 2

Prepared by : E D Cooper 2/11/94

Checked by: M D Bugge 2/12/94

Approved by : Thomas A Kooman 2-24-94

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 5.9.1.7. 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 5.9.1.7.

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 $\leq$ 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 5.9.1.7.

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.

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".

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

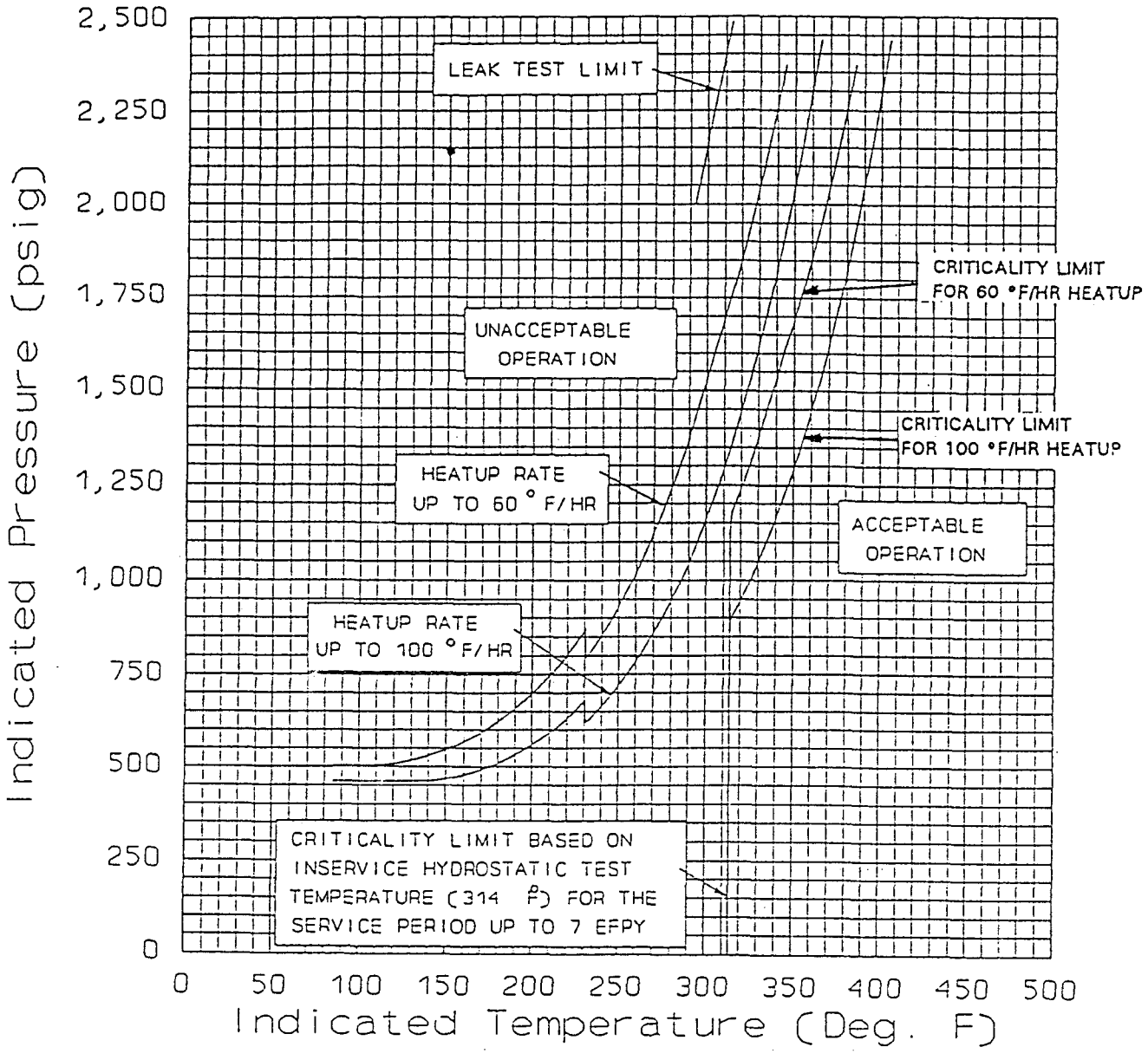

Figure 2.1-1
Watts Bar Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates of 60 and 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors) Including 10% Relaxation in Pressure for Temperature < 231 °F per ASME Code Case N-514

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

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data (Ref. 1) plotted on Fig 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)
85	506	462			
90	506	462			
95	506	462			
100	506	462			
105	506	462			
110	506	462			
115	506	462			
120	509	462			
125	512	462			
130	517	462			
135	523	462			
140	530	463			
145	538	465			
150	547	469			
155	557	473			
160	568	478			
165	580	484			
170	593	492			
175	607	500			
180	623	510			
185	640	521			
190	658	532			
195	678	545			
200	699	560			
205	722	575			

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data (Ref. 1) plotted on Fig 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)
210	746	592			
215	773	611			
220	801	631			
225	832	653			
230	865	676			
231	872	682			
231	792	620			
235	818	638			
240	853	663			
245	890	690			
250	930	719			
255	972	750			
260	1018	784			
265	1068	820			
270	1121	858			
275	1177	900			
280	1238	945			
285	1303	993			
290	1373	1044			
293			2000		
295	1448	1100	2046		
300	1528	1159	2162		
305	1612	1223	2277		
310	1686	1291	2393		
314			2485	0.0 to 1121	0.0 to 858

Table 2.1-1
Watts Bar Unit 1 Heatup Limits
(Data (Ref. 1) plotted on Fig 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)
315	1765	1364		1177	900
320	1850	1442		1238	945
325	1940	1525		1303	993
330	2037	1615		1373	1044
335	2141	1711		1448	1100
340	2251	1813		1528	1159
345	2369	1922		1612	1223
350		2039		1686	1291
355		2163		1765	1364
360		2296		1850	1442
365		2437		1940	1525
370				2037	1615
375				2141	1711
380				2251	1813
385				2369	1922
390					2039
395					2163
400					2296
405					2437

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

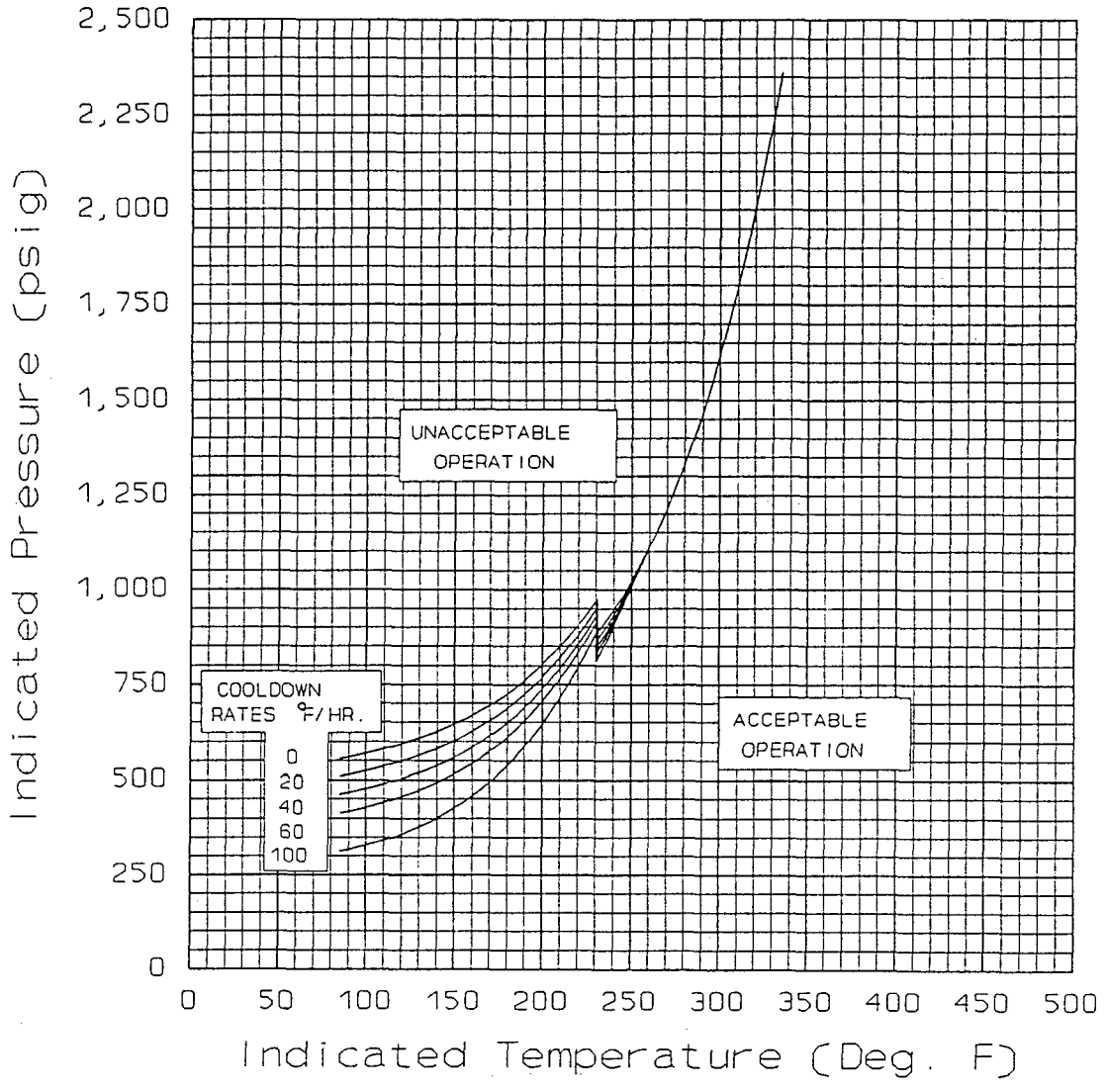

Figure 2.1-2
Watts Bar Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors) Including 10% Relaxation in Pressure for Temperatures < 231 °F per ASME Code Case N-514

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

Table 2.1-2
Watts Bar Unit 1 Cooldown Limits
(Data (Ref. 1) plotted on Fig 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
85	312	413	462	510	557
90	317	418	467	514	561
95	323	423	472	519	566
100	329	429	477	524	570
105	335	435	483	530	576
110	343	441	489	536	582
115	350	448	496	542	588
120	359	456	503	549	594
125	368	464	511	557	602
130	378	473	519	565	609
135	389	482	528	573	618
140	401	493	538	583	627
145	413	504	549	593	636
150	427	516	560	603	646
155	442	529	573	615	657
160	458	544	586	628	669
165	476	559	600	641	682
170	495	575	616	656	696
175	515	593	632	672	711
180	538	612	650	688	726
185	562	633	670	707	744
190	588	656	691	726	762
195	615	680	713	747	782
200	645	706	737	770	803
205	678	734	763	794	826
210	713	764	791	820	850
215	751	796	822	848	876
220	792	831	854	879	904

Table 2.1-2
Watts Bar Unit 1 Cooldown Limits
(Data (Ref. 1) plotted on Fig 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
225	836	869	889	911	935
230	883	909	927	946	968
231	894	918	935	954	975
231	812	835	850	867	886
235	850	867	879	894	912
240	900	909	919	931	946
245	954	955	962	971	983
250	1011	1005	1007	1013	1022
255	1064	1058	1057	1059	1064
260	1110	1110	1109	1108	1110
265	1159	1159	1159	1159	1159
270	1212	1212	1212	1212	1212
275	1268	1268	1268	1268	1268
280	1329	1329	1329	1329	1329
285	1393	1393	1393	1393	1393
290	1463	1463	1463	1463	1463
295	1538	1538	1538	1538	1538
300	1618	1618	1618	1618	1618
305	1704	1704	1704	1704	1704
310	1796	1796	1796	1796	1796
315	1894	1894	1894	1894	1894
320	2000	2000	2000	2000	2000
325	2112	2112	2112	2112	2112
330	2233	2233	2233	2233	2233
335	2361	2361	2361	2361	2361

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

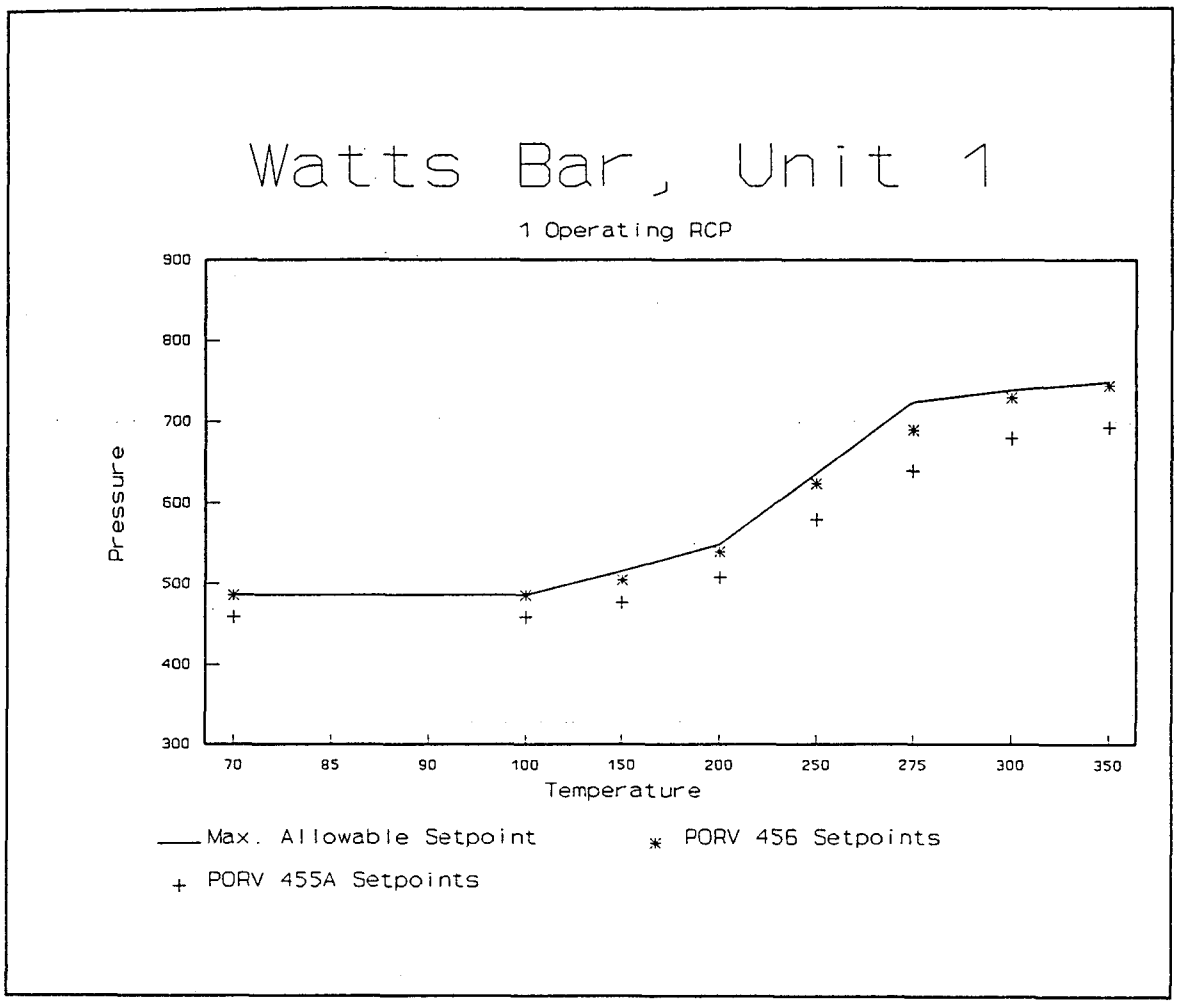

Figure 3.1-1
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

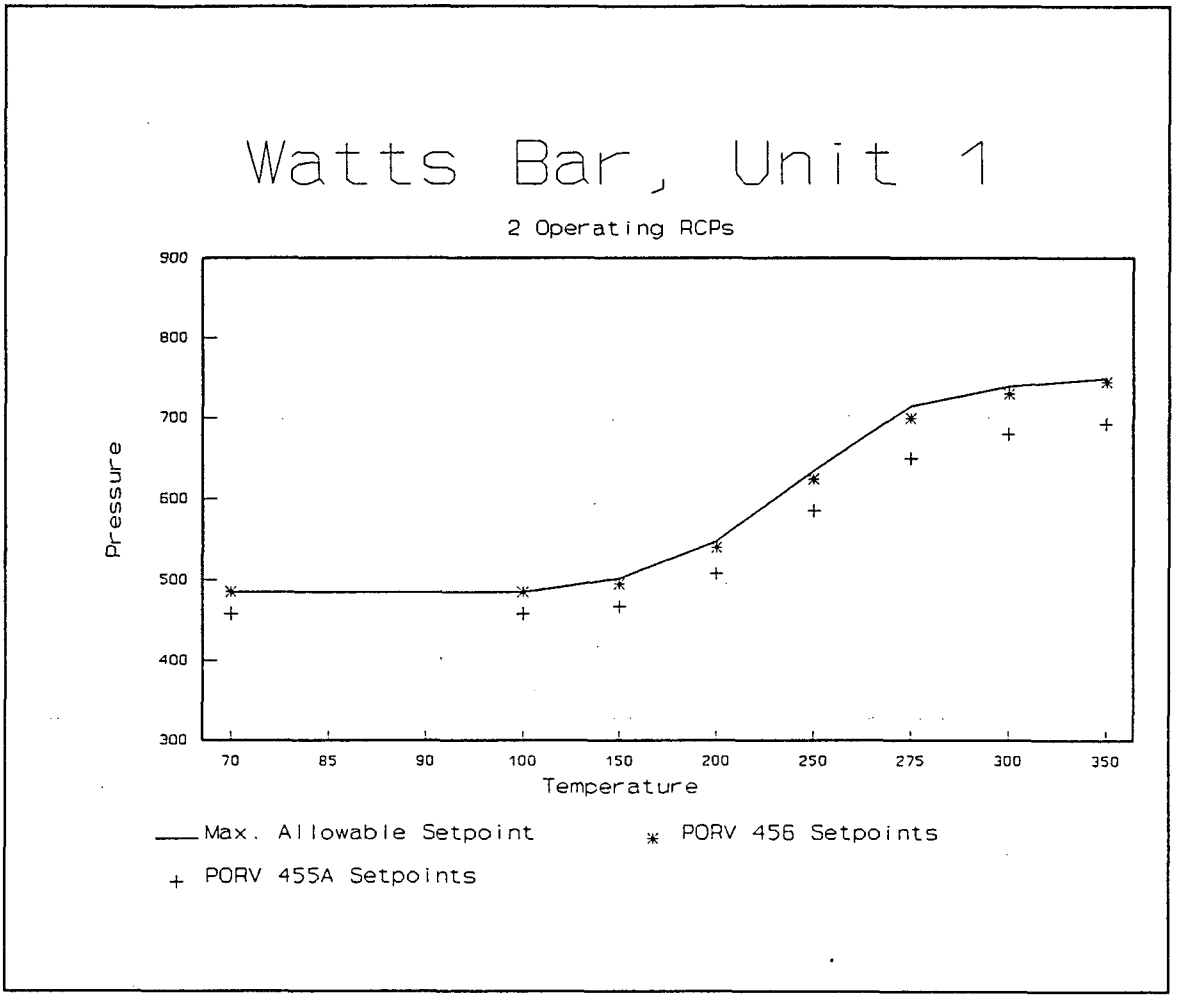

Figure 3.1-2
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

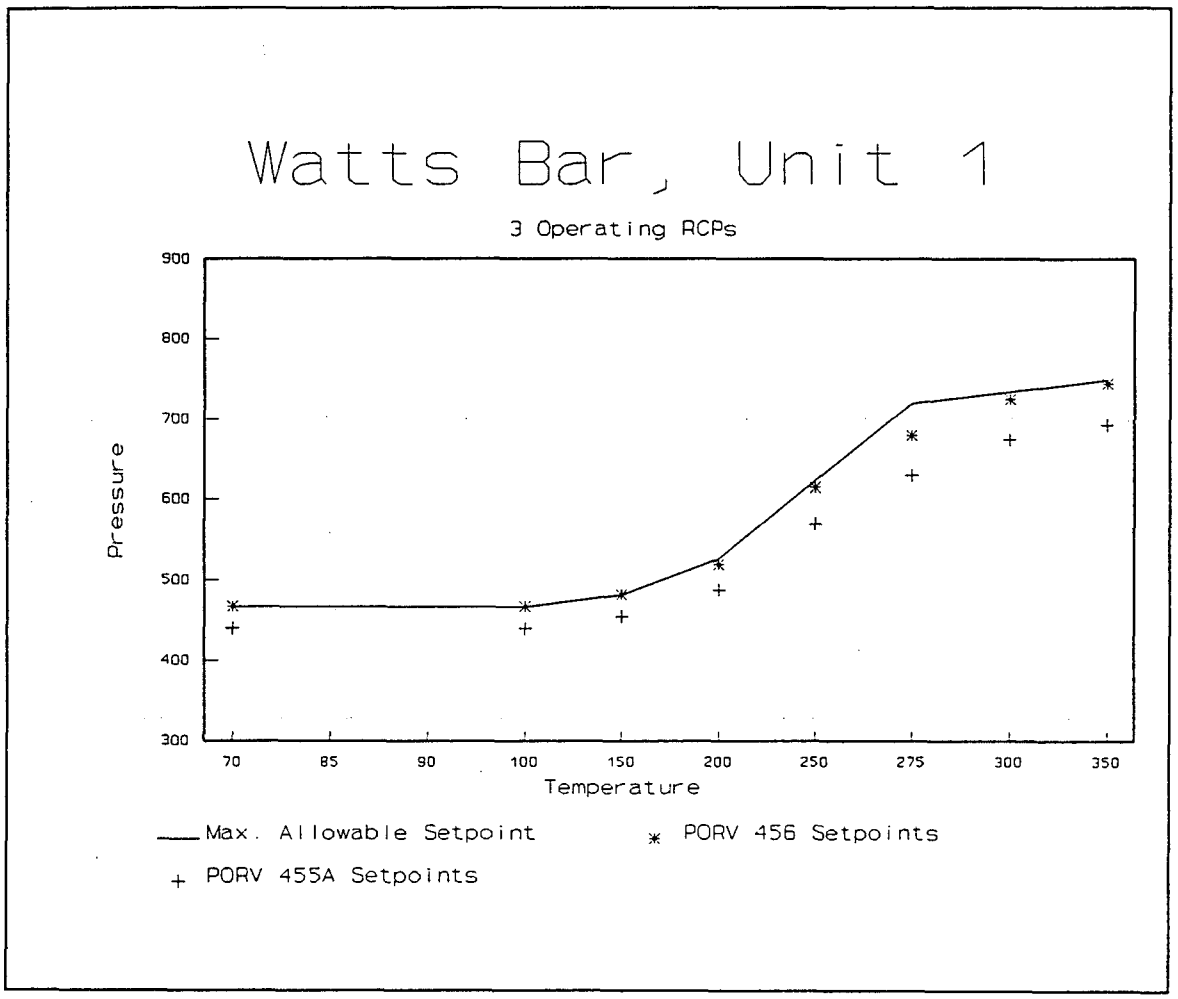

Figure 3.1-3
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

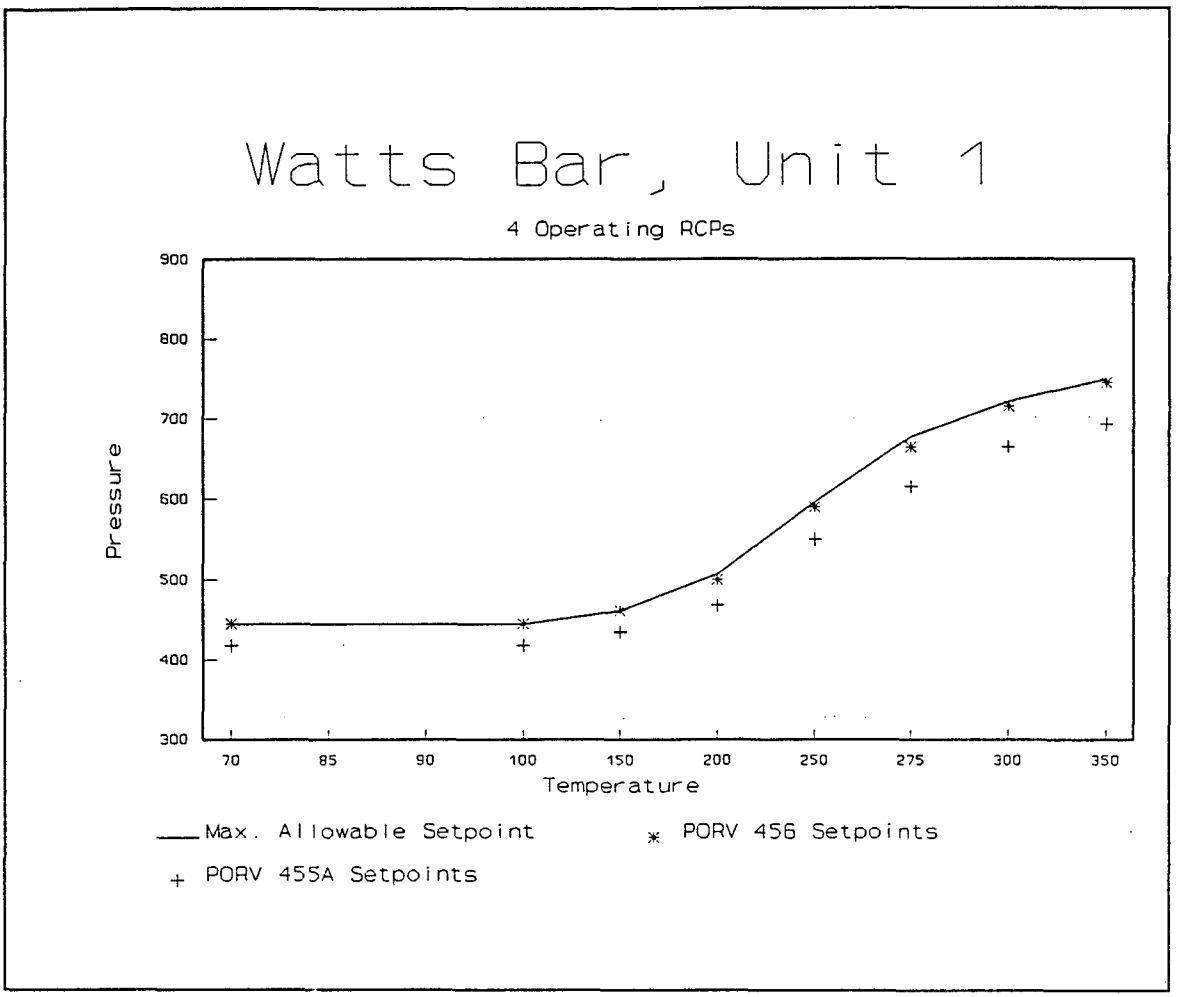

Figure 3.1-4
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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°	3.0	Stand-By	----
Y	238.5°	3.0	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]

SOURCE NOTES

1. NCO820285003
2. NCO820285004

APPENDIX "B"

TO RCS SYSTEM DESCRIPTION N3-68-4001

WATTS BAR UNIT 2

RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

REVISION 2

Prepared by: Ed Cooper 2/11/94

Checked by: Robert D Briggs 2-12-94

Approved by: Frank A Koontz 2-24-94

RCS PRESSURE AND TEMPERATURE LIMITS REPORT FOR WATTS BAR UNIT 2

1.0 RCS Pressure and Temperature Limits Report (PTLR)

This PTLR for Watts Bar Unit 2 has been prepared in accordance with the requirements of Technical Specification 5.9.1.7. 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 5.9.1.7.

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 $\leq$ 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 5.9.1.7.

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 a 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.

REFERENCES

1. WCAP-13830, "Heatup and Cooldown Limit Curves for Normal Operation for Watts Bar Unit 2", August 1993.

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

3. WCAP-9455, Revision 1, "Watts Bar Unit 2 Reactor Vessel Radiation Surveillance Program", April, 1993.

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

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD (1/4T)
INTERMEDIATE SHELL FORGING 05 (3/4-T)

INITIAL RT_{NDT} 14°F

LIMITING ART AT 32 EFPY: 1/4-T, 86.1 °F
3/4-T, 69.6 °F

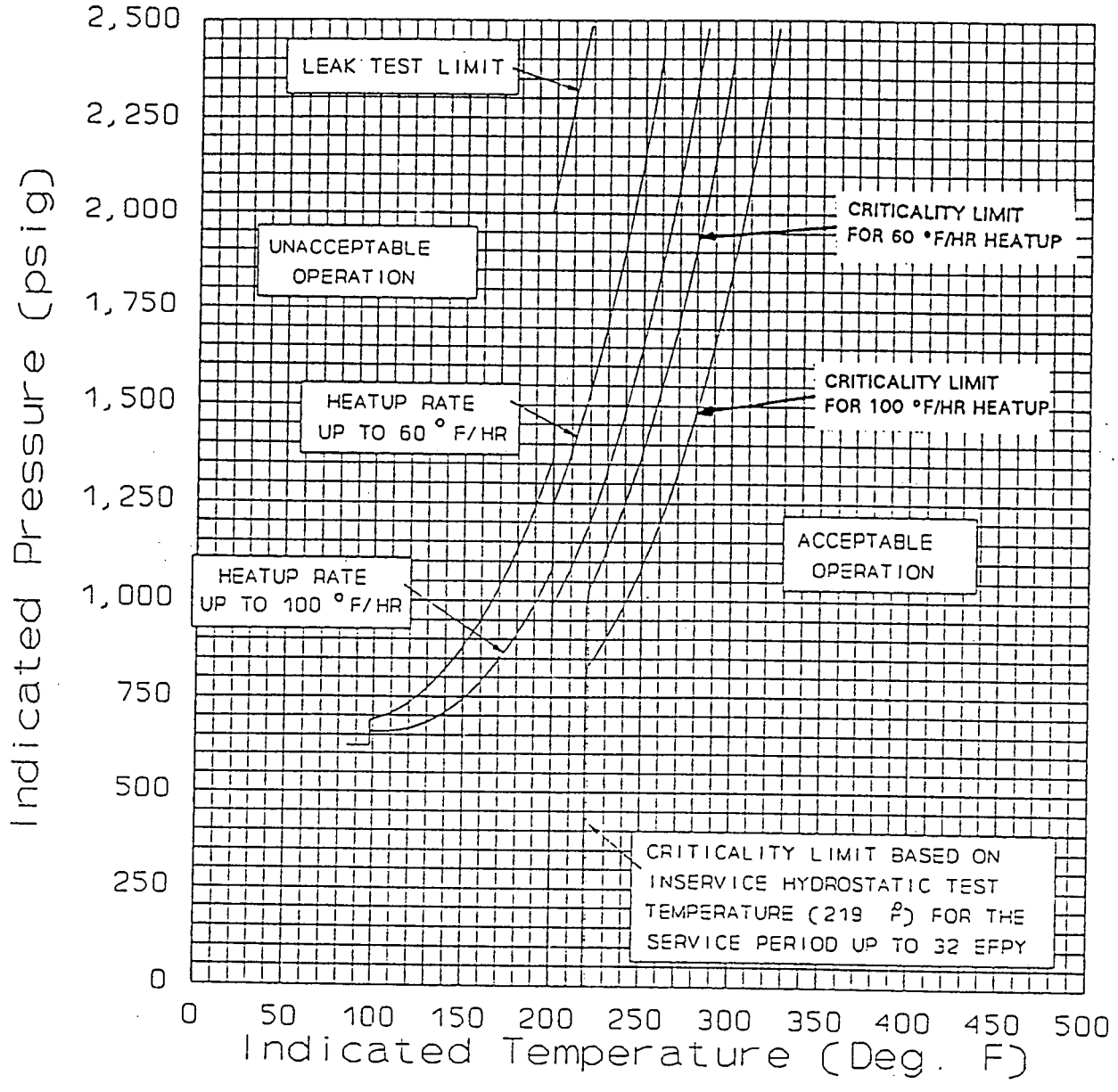

Figure 2.1-1
Watts Bar Unit 2 Reactor Coolant System Heatup Limitations (Heatup rates up to 60 and 100°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors) Including 10% Relaxation in Pressure for Temperatures < 200 °F per ASME Code Case N-514
(Plotted data (Ref. 1) provided on Table 2.1-1)

Table 2.1-1
Watts Bar Unit 2 Heatup Limits
(Data (Ref. 1) plotted on Fig 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMIT (60 °F/HR)	CRITICALITY LIMIT (100 °F/HR)
85	621	621			
90	621	621			
95	621	621			
98	621	621			
98	690	660			
100	691	660			
105	698	660			
110	708	660			
115	720	662			
120	736	667			
125	754	675			
130	774	684			
135	797	696			
140	823	710			
145	851	727			
150	882	746			
155	915	767			
160	952	791			
165	991	817			
170	1034	846			
175	1080	878			
180	1129	913			
185	1182	951			
190	1240	991			
195	1301	1036			

Table 2.1-1
Watts Bar Unit 2 Heatup Limits
(Data (Ref. 1) plotted on Fig 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMIT (60 °F/HR)	CRITICALITY LIMIT (100 °F/HR)
198			2000		
200	1368	1084	2046		
200	1243	985	2046		
205	1308	1032	2162		
210	1377	1083	2277		
215	1452	1137	2393		
219			2485	0.0 to 981	0.0 to 798
220	1531	1196		1026	830
225	1617	1259		1075	864
230	1708	1327		1127	901
235	1806	1400		1183	942
240	1910	1478		1243	985
245	2022	1562		1308	1032
250	2141	1651		1377	1083
255	2269	1747		1452	1137
260	2404	1850		1531	1196
265		1960		1617	1259
270		2077		1708	1327
275		2202		1806	1400
280		2336		1910	1478
285		2478		2022	1562
290				2141	1651
295				2269	1747
300				2404	1850
305					1960

Table 2.1-1
Watts Bar Unit 2 Heatup Limits
(Data (Ref. 1) plotted on Fig 2.1-1)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	HEATUP RATE (60 °F/HR)	HEATUP RATE (100 °F/HR)	LEAK TEST LIMITS	CRITICALITY LIMIT (60 °F/HR)	CRITICALITY LIMIT (100 °F/HR)
310					2077
315					2202
320					2336
325					2478

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	CIRCUMFERENTIAL WELD (1/4-T) INTERMEDIATE SHELL FORGING 05 (3/4-T)
LIMITING ART AT 32 EFPY:	1/4-T, 86.1 °F 3/4-T, 69.6 °F

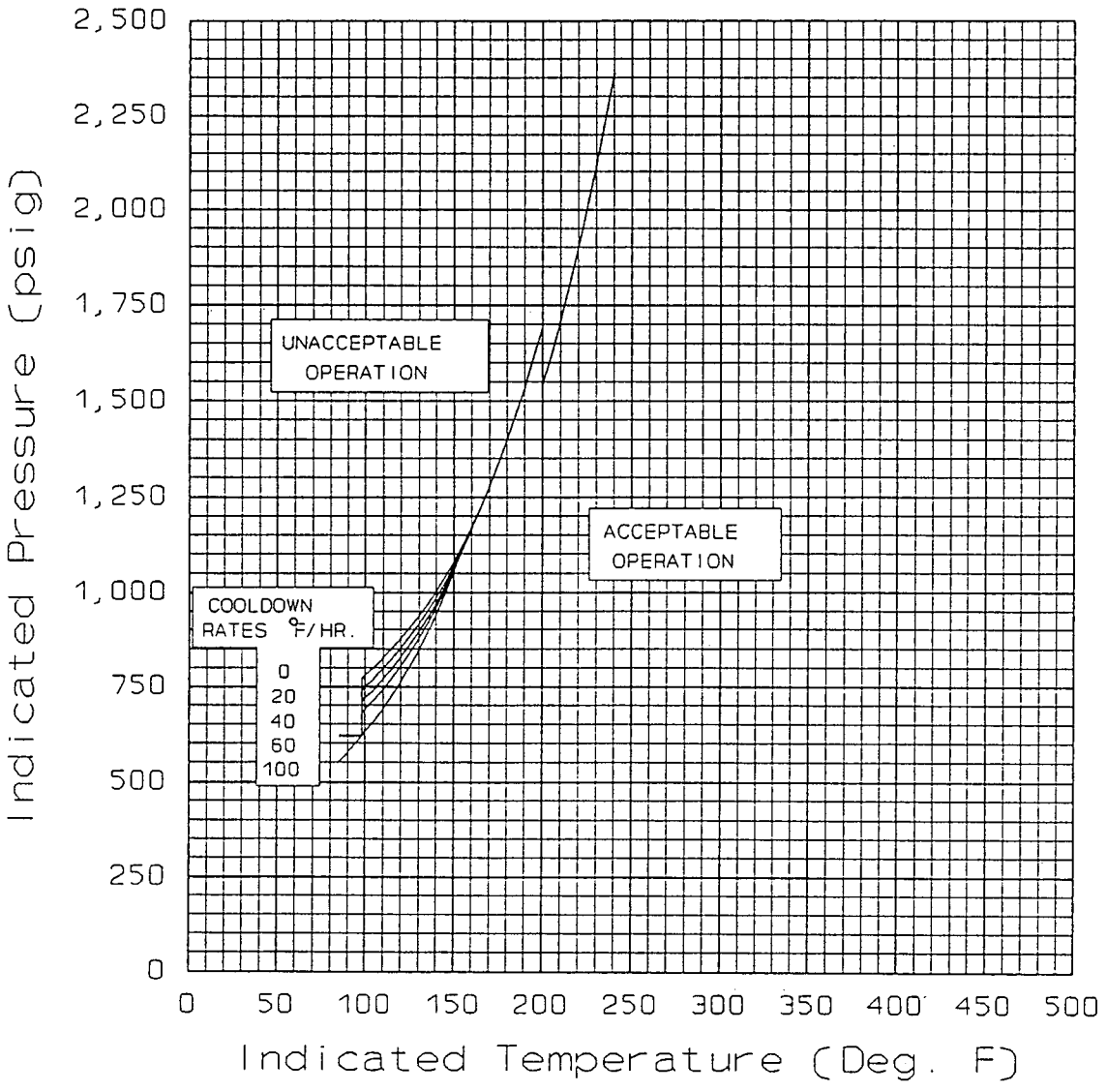

Figure 2.1-2
Watts Bar Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown rates up to 100°F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors) Including 10% Relaxation in Pressure for Temperatures < 200 ° F per ASME Code Case N-514
(Plotted data (Ref. 1) provided on Table 2.1-2)

Table 2.1-2
Watts Bar Unit 2 Cooldown Limits
(Data (Ref. 1) plotted on Fig 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
85	553	621	621	621	621
90	577	621	621	621	621
95	602	621	621	621	621
98		621	621	621	621
98		679	710	742	774
100	629	689	719	750	782
105	659	714	743	773	803
110	691	742	769	797	826
115	725	772	797	823	850
120	762	804	827	851	877
125	802	838	859	881	905
130	845	875	894	914	935
135	891	915	931	949	968
140	941	958	971	986	1003
145	995	1005	1014	1026	1041
150	1053	1055	1061	1070	1081
155	1115	1109	1111	1116	1125
160	1171	1166	1164	1166	1171
165	1221	1221	1221	1220	1221
170	1275	1275	1275	1275	1275
175	1333	1333	1333	1333	1333
180	1395	1395	1395	1395	1395
185	1462	1462	1462	1462	1462
190	1533	1533	1533	1533	1533
195	1610	1610	1610	1610	1610
200	1692	1692	1692	1692	1692

Table 2.1-2
Watts Bar Unit 2 Cooldown Limits
(Data (Ref. 1) plotted on Fig 2.1-2)

RCS TEMPERATURE (°F)	INDICATED PRESSURE (PSIG)				
	100 °F/HR	60 °F/HR	40 °F/HR	20 °F/HR	0 °F/HR
200	1539	1539	1539	1539	1539
205	1619	1619	1619	1619	1619
210	1705	1705	1705	1705	1705
215	1797	1797	1797	1797	1797
220	1895	1895	1895	1895	1895
225	2001	2001	2001	2001	2001
230	2113	2113	2113	2113	2113
235	2234	2234	2234	2234	2234
240	2363	2363	2363	2363	2363

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

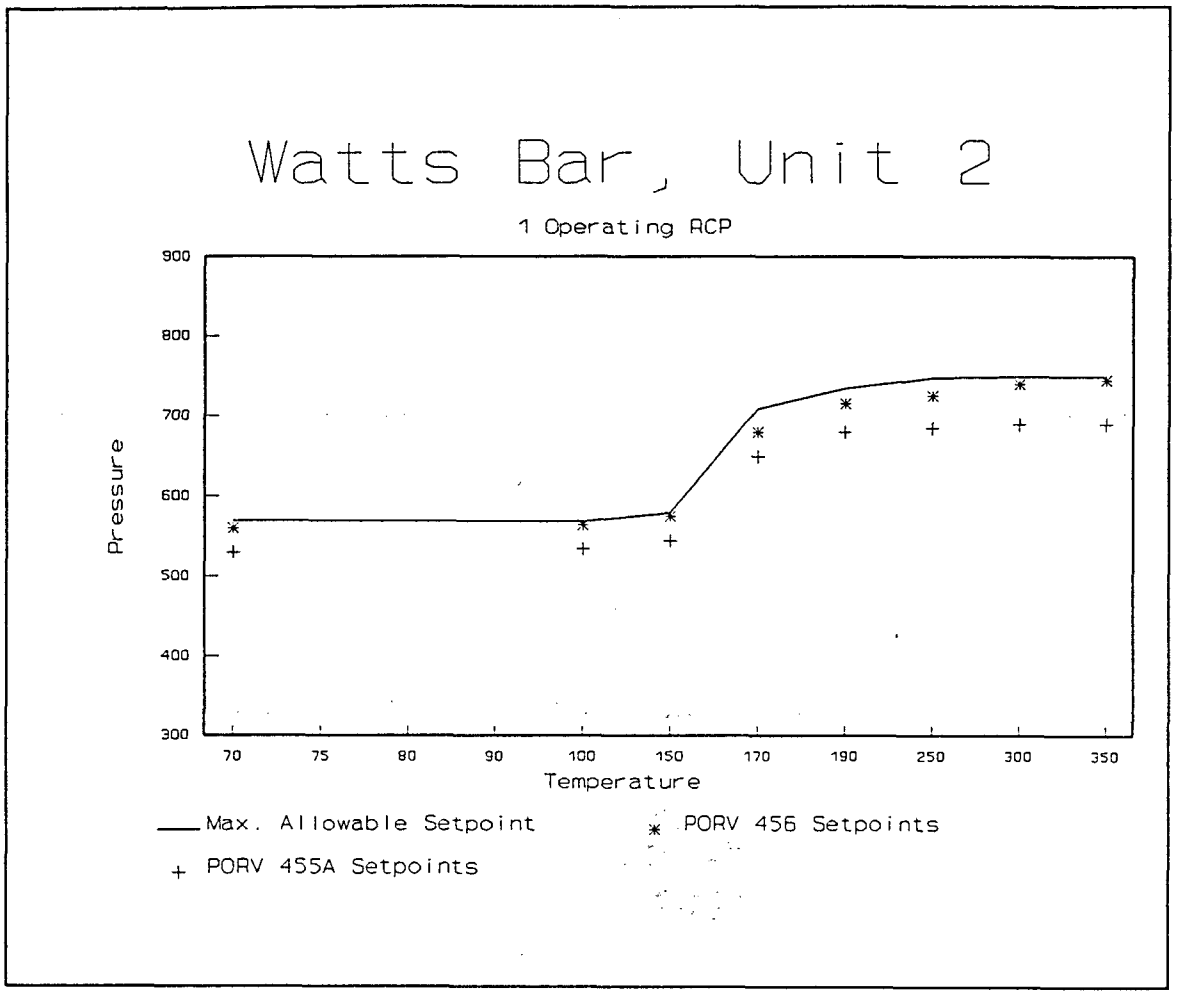

Figure 3.1-1
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

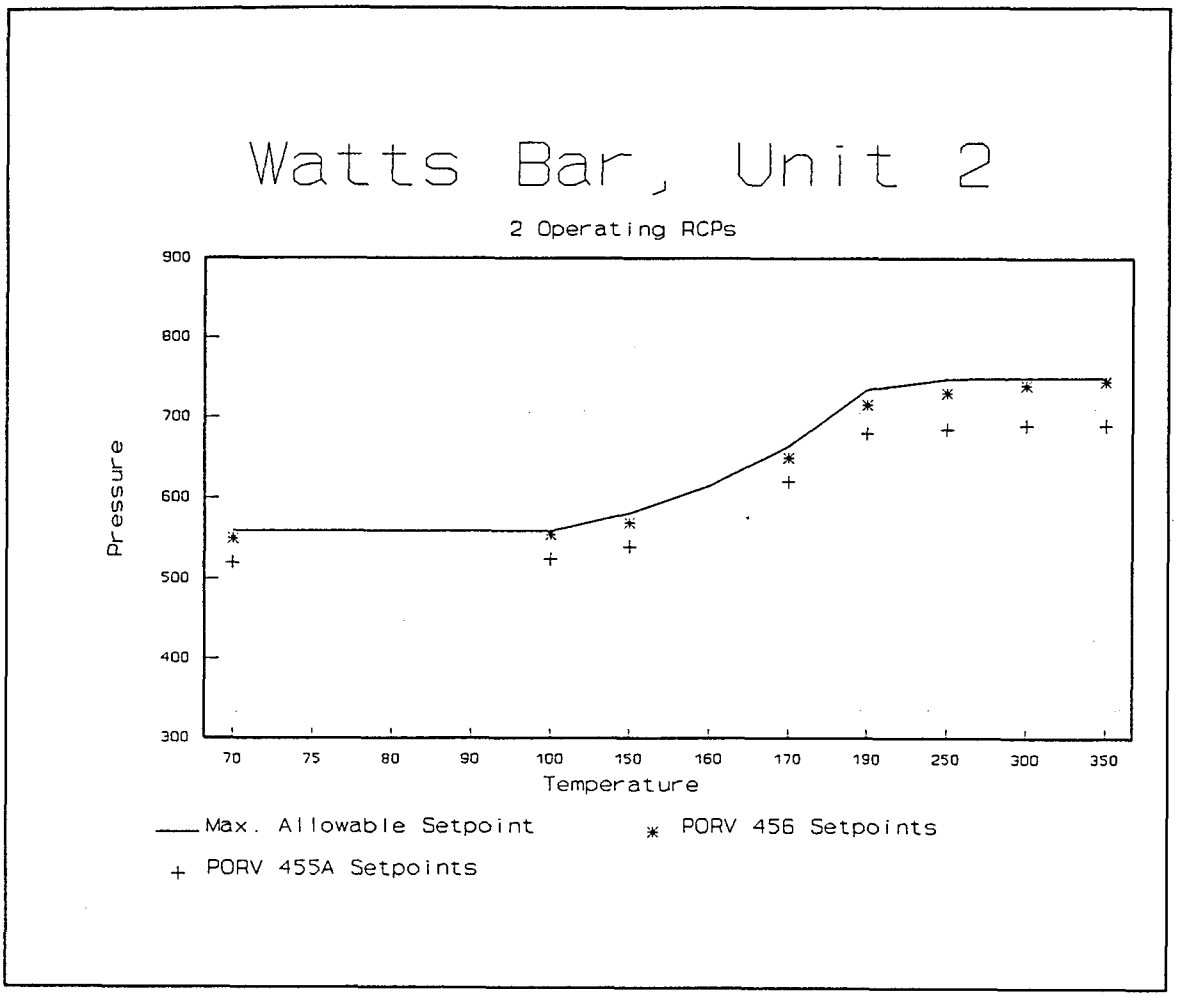

Figure 3.1-2
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

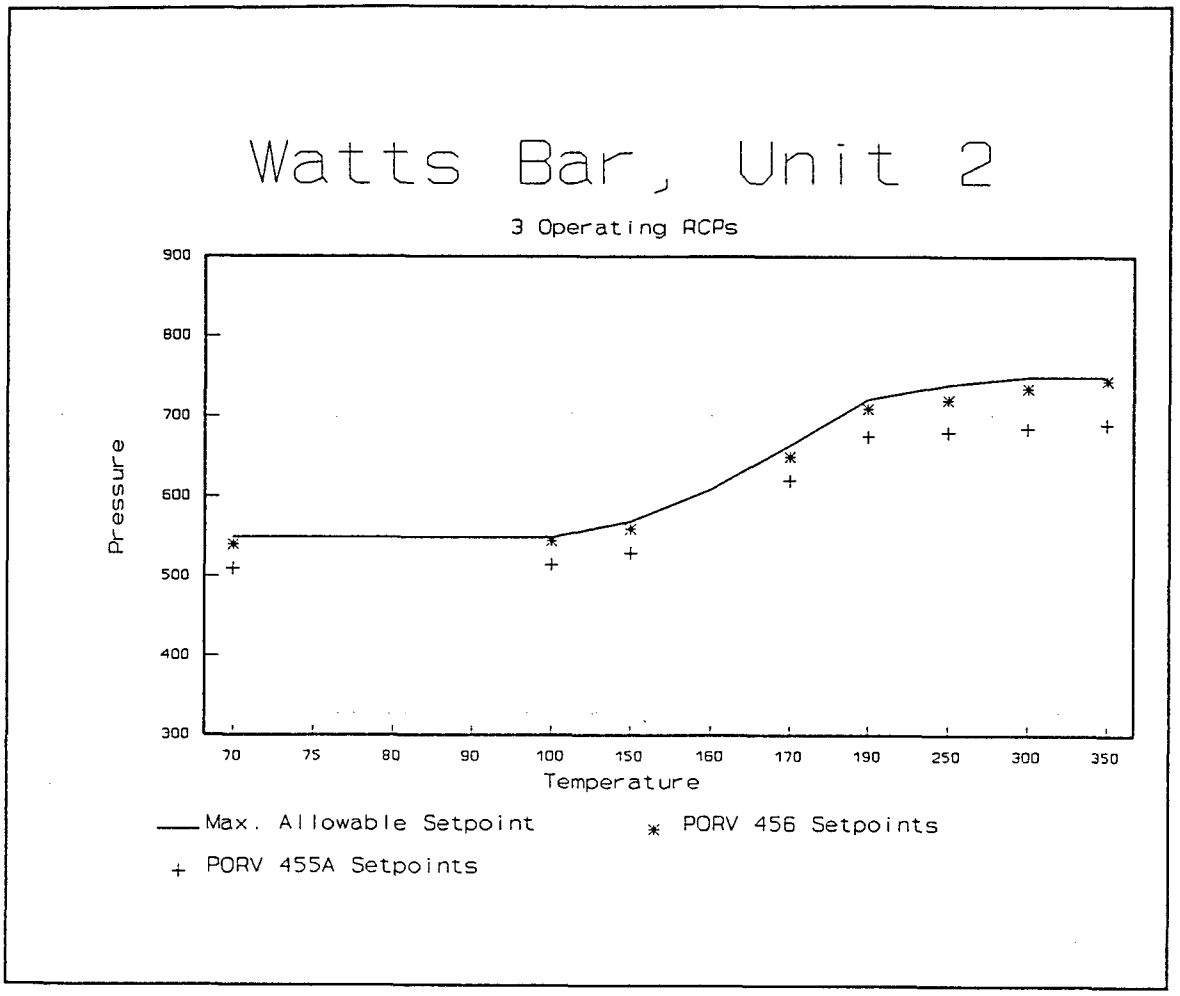

Figure 3.1-3
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

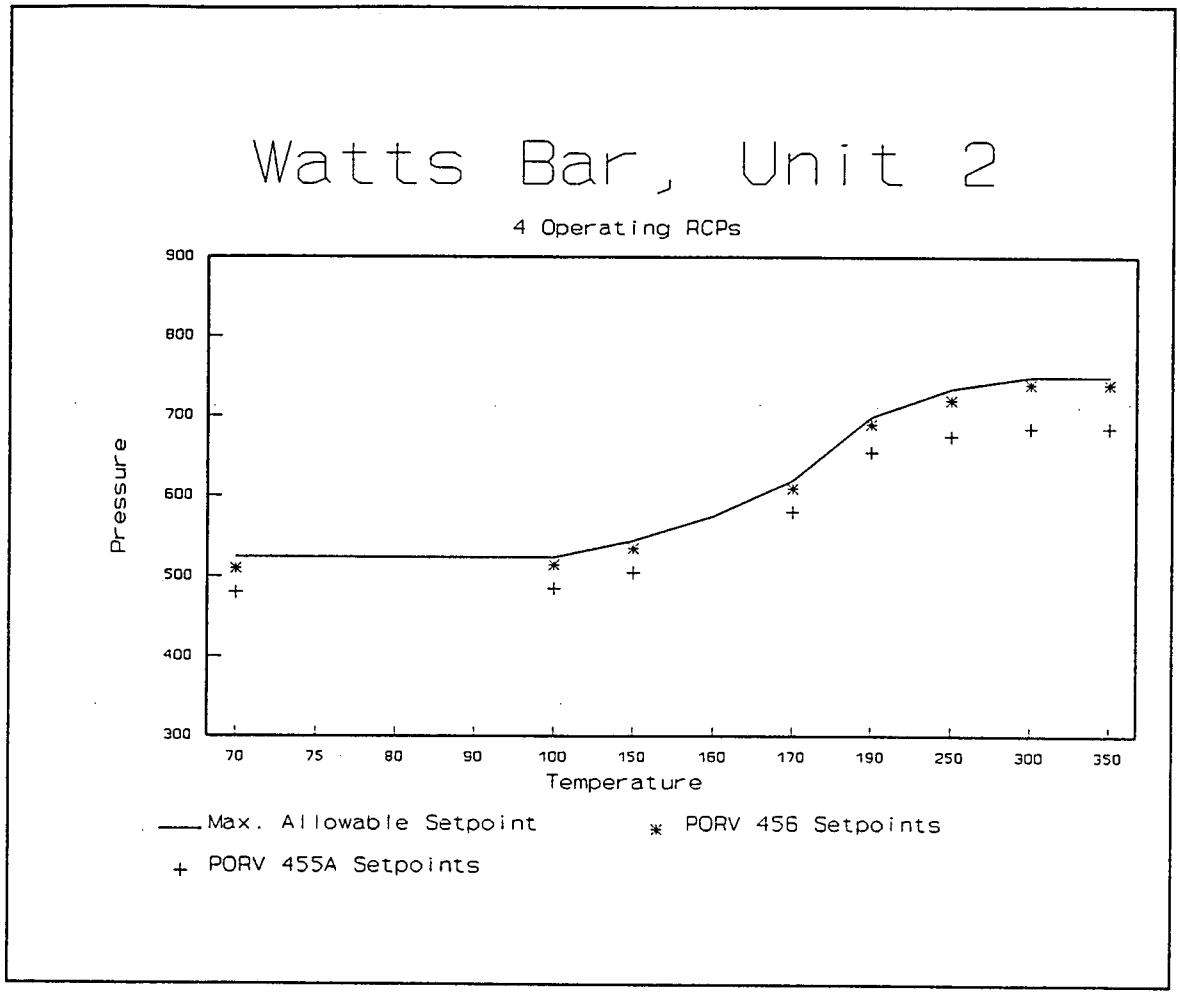

Figure 3.1-4
PORV Setpoint vs RCS Temperature
(Plotted data (Ref. 3) provided on Table 3.1-1)

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

Table 3.1-1
Watts Bar Unit 2 PORV Setpoints vs Temperature
(Data (Ref. 3) Plotted on Figure 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	560	530	550	520	540	510	510	480
100	565	535	555	525	545	515	515	485
150	575	545	570	540	560	530	535	505
170	680	650	650	620	650	620	600	570
190	715	680	715	680	710	675	690	655
250	725	685	730	685	720	680	720	675
300	740	690	740	690	735	685	740	685
350	745	690	745	690	745	690	740	685
450	2350	2350	2350	2350	2350	2350	2350	2350

Table 4.0-1
Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Time[a] [b]	Estimated Capsule Fluence (n/cm^2)[c]
U	56.0°	4.86	1st Refueling Outage	4.80×10^{18}
W	124.0°	4.86	6.6	3.18×10^{19}
X	236.0°	4.86	13.2	6.36×10^{19}
Z	304.0°	4.86	Stand-By	----
V	58.5°	4.05	Stand-By	----
Y	238.5°	4.05	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 ($E > 1MeV$).