Westinghouse Class 3 (Non-Proprietary)

WCAP-13829
Revision 2

HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION FOR WATTS BAR UNIT 1

J. M. Chicots
M. J. Malone
P. A. Peter
E. Terek

July 1995

Work Performed Under Shop Order WGJP-4001B

Prepared by Westinghouse Electric Corporation
for the Tennessee Valley Authority

Approved by: [signature]
R. D. Rishel, Manager
Metallurgical & NDE Analysis

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

PREFACE

Revision 2

This report was revised to incorporate comments from the NRC. The fourth line on page 2 was revised from "were obtained" to "will be obtained". The third line of page 6 was revised from "the methods discussed" to "the methods of WCAP-14040 discussed".

Revision 1

Per the Tennessee Valley Authority, this report has been revised to include additional heatup and cooldown curves for normal operation, *without* the incorporation of the ASME Code Case N-514[6]. The heatup and cooldown pressure-temperature limit curves to be used for normal operation are presented in Figures 1 and 2 and Table A-1 (in tabular form). Figures 3 and 4 contain this document's original curves, which have the ASME Code Case N-514 10% pressure relaxation incorporated.

NOTE: The intention of ASME Code Case N-514 is to relax the steady-state curve data points only for use in the development of LTOP/COMS setpoints, not for the generation of Appendix G pressure-temperature limit curves used for normal operation of the plant.

TABLE OF CONTENTS

LIST OF ILLUSTRATIONS

LIST OF TABLES

1.0 INTRODUCTION

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Revision 2 (Radiation Embrittlement of Reactor Vessel Materials)[1]. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values at the 1/4-T and 3/4-T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The pressure-temperature limit curves in Figures 1 through 4 of this report do not include margins for instrumentation errors nor pressure differences between the wide-range pressure transmitter and the limiting reactor vessel beltline region. In addition, the pressure-temperature limit curves presented in Figures 3 and 4 contain a 10% relaxation in pressure for temperatures below 231°F (1/4-T RT_{NDT} + 50°F), per ASME Code Case N-514[6]. Application of ASME Code Case N-514 to the steady state pressure-temperature limit curve will increase the region where the low temperature overpressure protection (LTOP) system is active. However, use of Code Case N-514 has not been formally approved by the NRC and therefore it is recommended that the Tennessee Valley Authority interface with the NRC to obtain their position and approval on the use of ASME Code Case N-514.

2.0 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan[2]. The pre-irradiation fracture-toughness properties of the Watts Bar Unit 1 reactor vessel are presented in Table 1. The post-irradiation fracture toughness properties of the reactor vessel beltline material will be obtained directly from the Watts Bar Unit 1 Reactor Vessel Radiation Surveillance Program.

3.0 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies 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_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G of the ASME Code, Section III[3]. The K_{IR} curve is given by the following equation:

$$K_{IR} = 26.78 + 1.223 * e^{[0.0145 (T - RTNDT + 160)]} \tag{1}$$

where K_{IR} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}.

Therefore, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C * K_{IM} + K_{IT} < K_{IR} \tag{2}$$

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

K_{IT} = stress intensity factor caused by the thermal gradients

K_{IR} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G 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 wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

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 1/4-T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4-T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IR} 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 1/4-T location and, therefore, allowable pressures may unknowingly be violated 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 1/4-T defect 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_{IR} for the 1/4-T crack during heatup is lower than the K_{IR} for the 1/4-T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{IR} values do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4-T flaw is considered. 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.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4-T deep outside surface flaw is assumed. 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 the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. 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 possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside, and the pressure limit must at all times be based on analysis of the most critical criterion.

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 when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Watts Bar Unit 1).

Table 1 indicates that the limiting unirradiated RT_{NDT} of -40°F occurs in the vessel flange of Watts Bar Unit 1, so the minimum allowable temperature of this region is 80°F at pressures greater than 621 psig. This limit is shown in Figures 1 through 4 whenever applicable.

4.0 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor pressure vessel have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods of WCAP-14040 discussed in Section 3.0. Since indication of reactor vessel beltline pressure is not available on the plant, the pressure difference between the wide-range pressure transmitter and the limiting beltline region must be accounted for when using the limit curves presented in this report.

In addition, at the request of Tennessee Valley Authority, the current heatup and cooldown curves presented in Figures 3 and 4 were adjusted to incorporate the 10% relaxation in pressure for temperatures below 231°F (1/4-T RT_{NDT} + 50°F), per ASME Code Case N-514. This enables LTOP systems to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G of Section XI, Article G-2215. In addition, Code Case N-514 requires LTOP systems to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than 1/4-T RT_{NDT} + 50°F, whichever is greater. This ASME Code Case N-514, however, has not been formally accepted by the NRC.

Figure 1 presents the heatup curves using heatup rates of 60°F/hr and 100°F/hr applicable for the first 7 EFPY. Figure 2 presents the cooldown curves using cooldown rates up to 100°F/hr applicable for the first 7 EFPY. No margins for possible instrumentation errors are included in the development of heatup and cooldown curves.

Figure 3 presents the heatup curves using heatup rates of 60°F/hr and 100°F/hr applicable for the first 7 EFPY. Figure 4 presents the cooldown curves using cooldown rates up to 100°F/hr applicable for the first 7 EFPY. No margins for possible instrumentation errors are included in the development of heatup and cooldown curves. However, the heatup and cooldown curves presented in Figures 3 and 4 include a 10% relaxation in pressure for temperatures below 231°F, per the ASME Code Case N-514.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 1 through 4. This is in addition to other criteria which must be met before the reactor is made critical.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 and 3. The straight line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10CFR Part 50. The governing equation for the hydrostatic test is defined in Appendix G to Section III of the ASME Code as follows:

$$1.5\ K_{IM} < K_{IR},$$

where K_{IM} is the stress intensity factor covered by membrane (pressure) stress,

$$K_{IR} = 26.78 + 1.233\ e^{[0.0145\ (T - RTNDT + 160)]},$$

where T is the minimum permissible metal temperature and RTNDT is the metal reference nil-ductility temperature.

The curved portion of the criticality limit is shifted 40°F to the right of and parallel to the heatup curve as required by Appendix G to 10 CFR Part 50. It should be noted that there are other criteria which must be met before the reactor is made critical. For example, the reactor must not be made critical until a steam bubble is formed in the pressurizer. The leak test limit curve shown on the heatup curves in Figures 1 and 3 represents minimum temperature requirements at leak test pressures ranging from 2000 psig to 2485 psig. The leak test limit curve was determined by the same method used to compute the inservice hydrostatic test temperature. This method used a 1.5 safety factor on the pressure stress intensity factor as explained previously.

The leak limit curve shown in Figure 1 represents minimum temperature requirements at the leak test pressure specified by applicable codes[2,3]. The leak test limit curve was determined by methods of References 2 and 4.

Figures 1 through 4 define limits for ensuring prevention of nonductile failure for the Watts Bar Unit 1 reactor vessel.

The data points used to develop the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 4 are presented in Tables A-1 and A-2 in Appendix A.

5.0 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE (ART)

From Regulatory Guide 1.99, Revision 2[1], the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (3)$$

Initial RT_{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. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may 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 adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \quad (4)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4-T or 3/4-T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth\ x)} = f_{surface} * e^{(-.24x)} \quad (5)$$

where 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 (4) to calculate ΔRT_{NDT} at the specific depth. The calculated surface fluence at 7 EFPY for Watts Bar Unit 1 is 6.96×10^{18} n/cm^2, based on projections using the design basis fluence[5].

The chemistry factor (CF, °F) was determined from Tables 1 and 2 of Reference 1, using the mean values of the copper and nickel content as reported in Table 1. If plant-specific surveillance data has been deemed credible per Regulatory Guide 1.99, Revision 2, it may be considered in the calculation of the chemistry factor.

All materials in the beltline region of Watts Bar Unit 1 reactor vessel were considered in determining the limiting material. The resulting ART values at the 1/4-T and 3/4-T locations are summarized in

Table 2. From Table 2, it can be seen that the limiting material is the Intermediate Shell Forging 05 for heatup and cooldown curves applicable to 7 EFPY. Sample calculations to determine the ART values for the intermediate shell forging at 7 EFPY are shown in Table 3.

TABLE 1

WATTS BAR UNIT 1 REACTOR VESSEL TOUGHNESS TABLE (UNIRRADIATED)

Material Description	Cu (%)*	Ni (%)*	IRT_{NDT} (°F) [(a)]
Closure Head Flange	0.13	0.75	-42 [(b)]
Vessel Flange	--	0.92	-40 [(b)]
Intermediate Shell Forging 05	0.17	0.80	47
Lower Shell Forging 04	0.08	0.83	5
Intermediate to Lower Shell Circumferential Weld Seam W05	0.05	0.70	-43

(a) Initial RT_{NDT} values were estimated per U.S. NRC Standard Review Plan[2]. The initial RT_{NDT} values for the plates and welds are measured values.

(b) These values are used for considering flange requirements[4] for the heatup/cooldown curves.

* Material property values were obtained from material test certifications from the original fabrication.

TABLE 2

SUMMARY OF ADJUSTED REFERENCE TEMPERATURE (ART) VALUES AT THE 1/4-T AND 3/4-T LOCATIONS FOR 7 EFPY

Material	1/4-T ART (°F)	3/4-T ART (°F)
Inter. Shell Forging 05	181.11*	147.70*
Lower Shell Forging 04	77.68	56.54
Circumferential Weld Seam	60.14	25.72

* These ART numbers were used to generate the heatup and cooldown curves presented in this report.

TABLE 3

CALCULATION OF ADJUSTED REFERENCE TEMPERATURES AT 7 EFPY FOR THE LIMITING WATTS BAR UNIT 1 REACTOR VESSEL MATERIAL INTERMEDIATE SHELL FORGING 05

Parameter	Regulatory Guide 1.99 -Revision 2 7 EFPY 1/4-T	3/4-T
Chemistry Factor, CF (°F)	132	132
Fluence, f (10^{19} n/cm^2)[(a)]	0.4188	0.1517
Fluence Factor, ff	0.758	0.505
********		
ΔRT_{NDT} = CF x ff (°F)	100.1	66.7
Initial RT_{NDT}, I (°F)	47	47
Margin, M (°F) [(b)]	34	34
********		
Revision 2 to Regulatory Guide 1.99 Adjusted Reference Temperature, ART (°F) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin	181.1	147.7

****

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1 MeV) = 0.696 at 7 EFPY. The Watts Bar Unit 1 reactor vessel wall thickness is 8.465 inches at the beltline region.

(b) Margin is calculated as, $M = 2\,[\,\sigma_i^2 + \sigma_\Delta^2]^{0.5}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is assumed to be 0°F since the initial RT_{NDT} is a measured value. The standard deviation for ΔRT_{NDT} 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}.

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

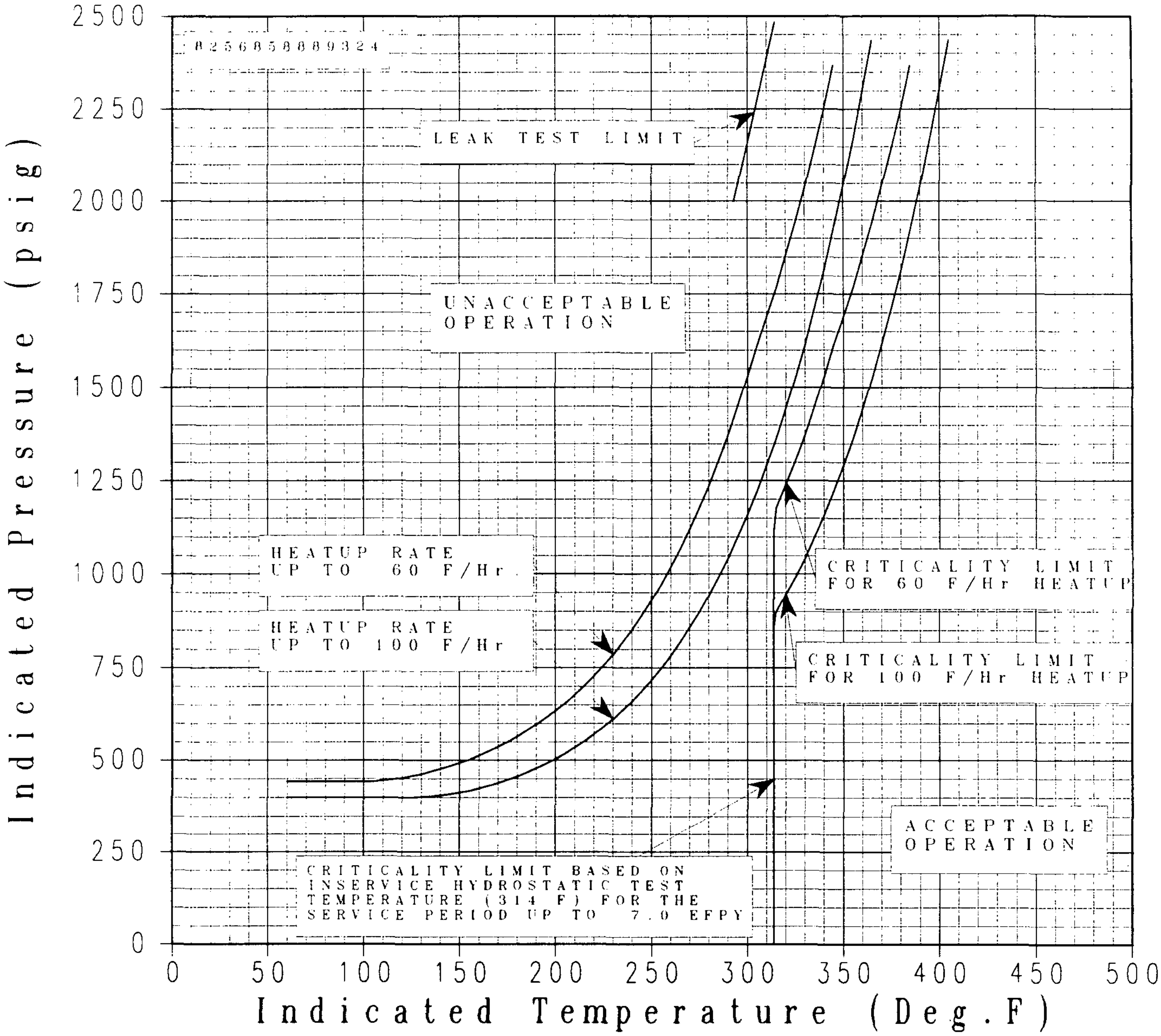

Figure 1 Watts Bar Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 60°F/hr and 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)

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

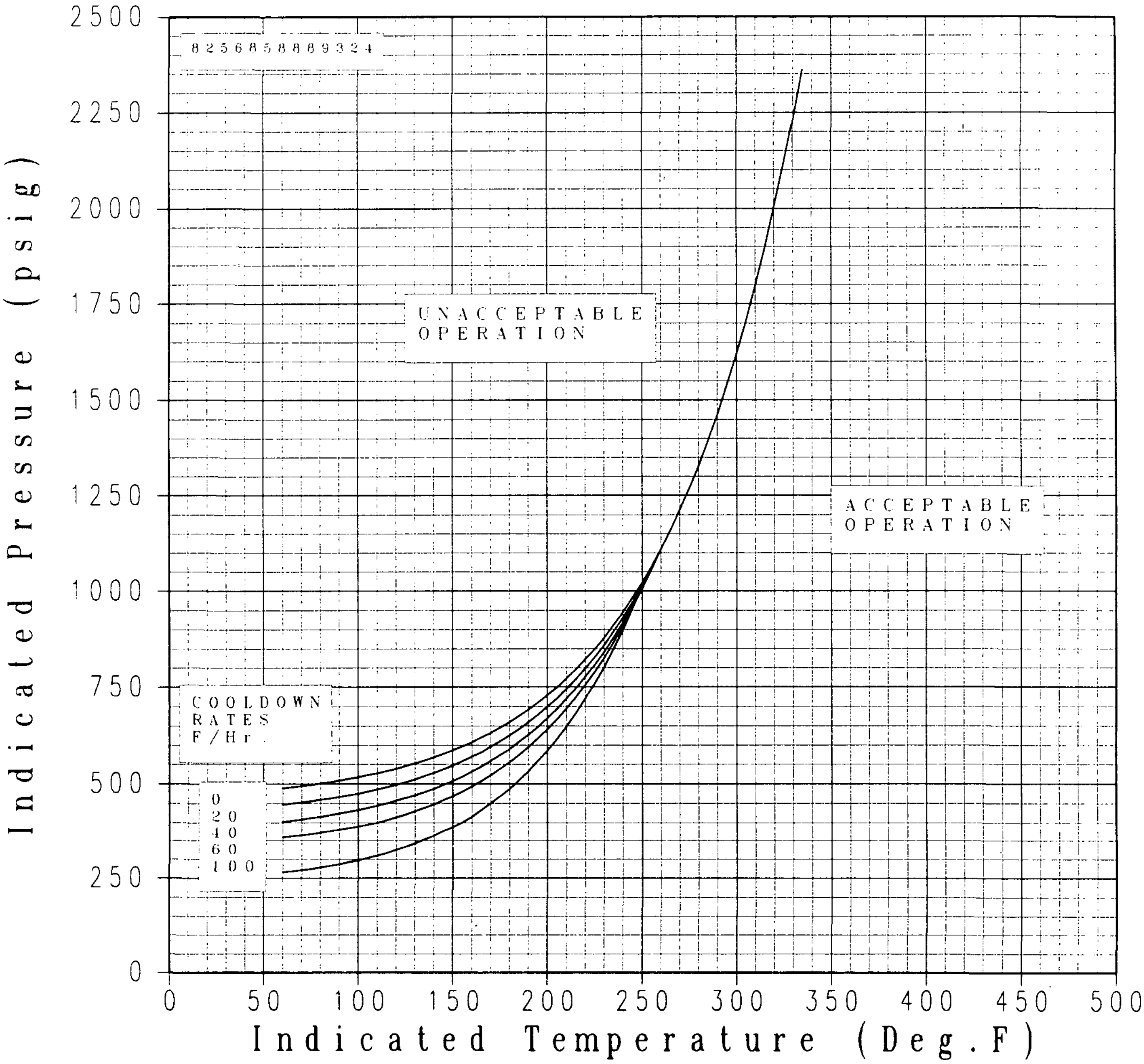

Figure 2 Watts Bar Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)

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

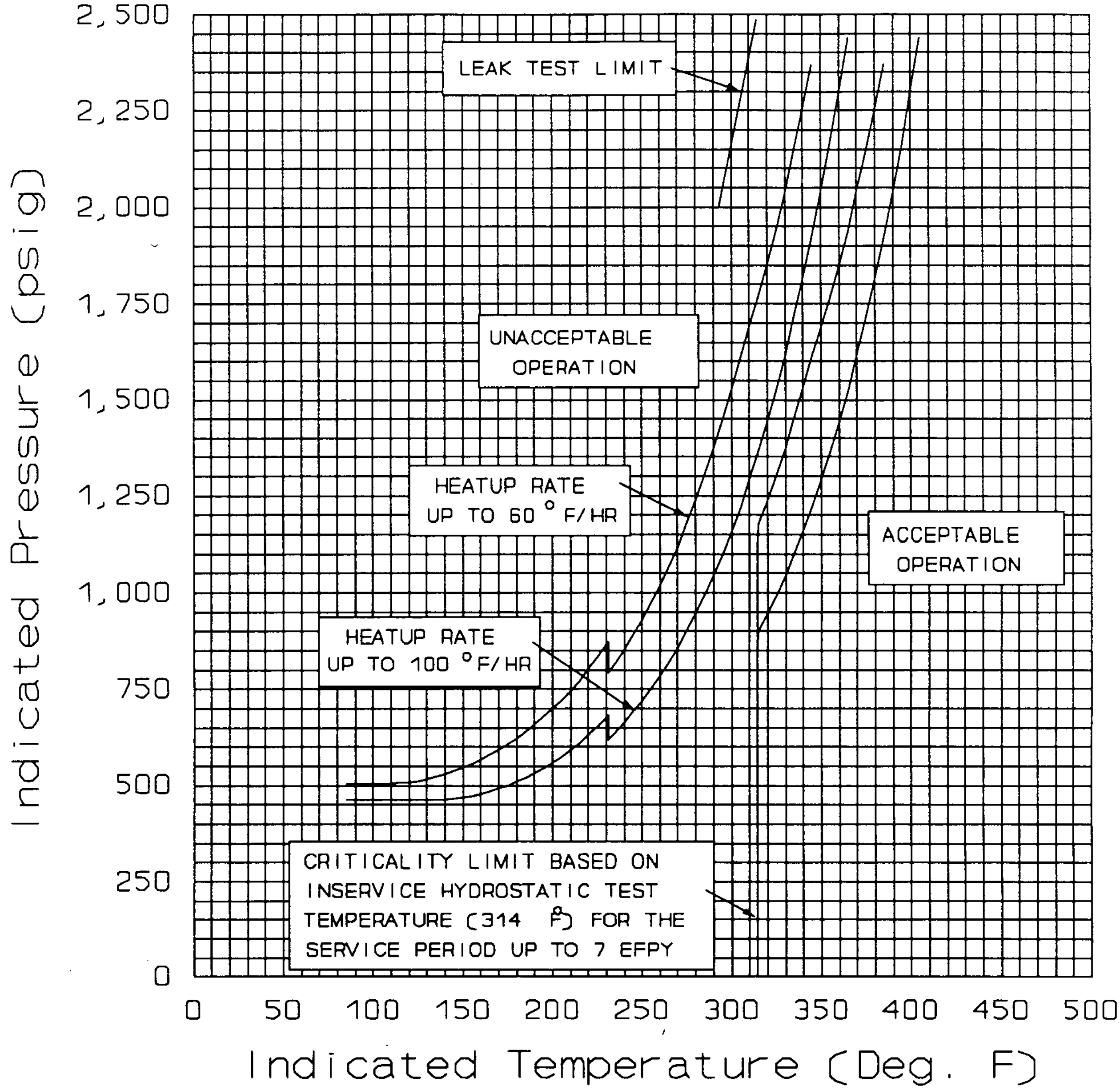

Figure 3 Watts Bar Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 60°F/hr and 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors)
Including 10% Relaxation in Pressure for Temperatures <231°F per ASME Code Case N-514

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

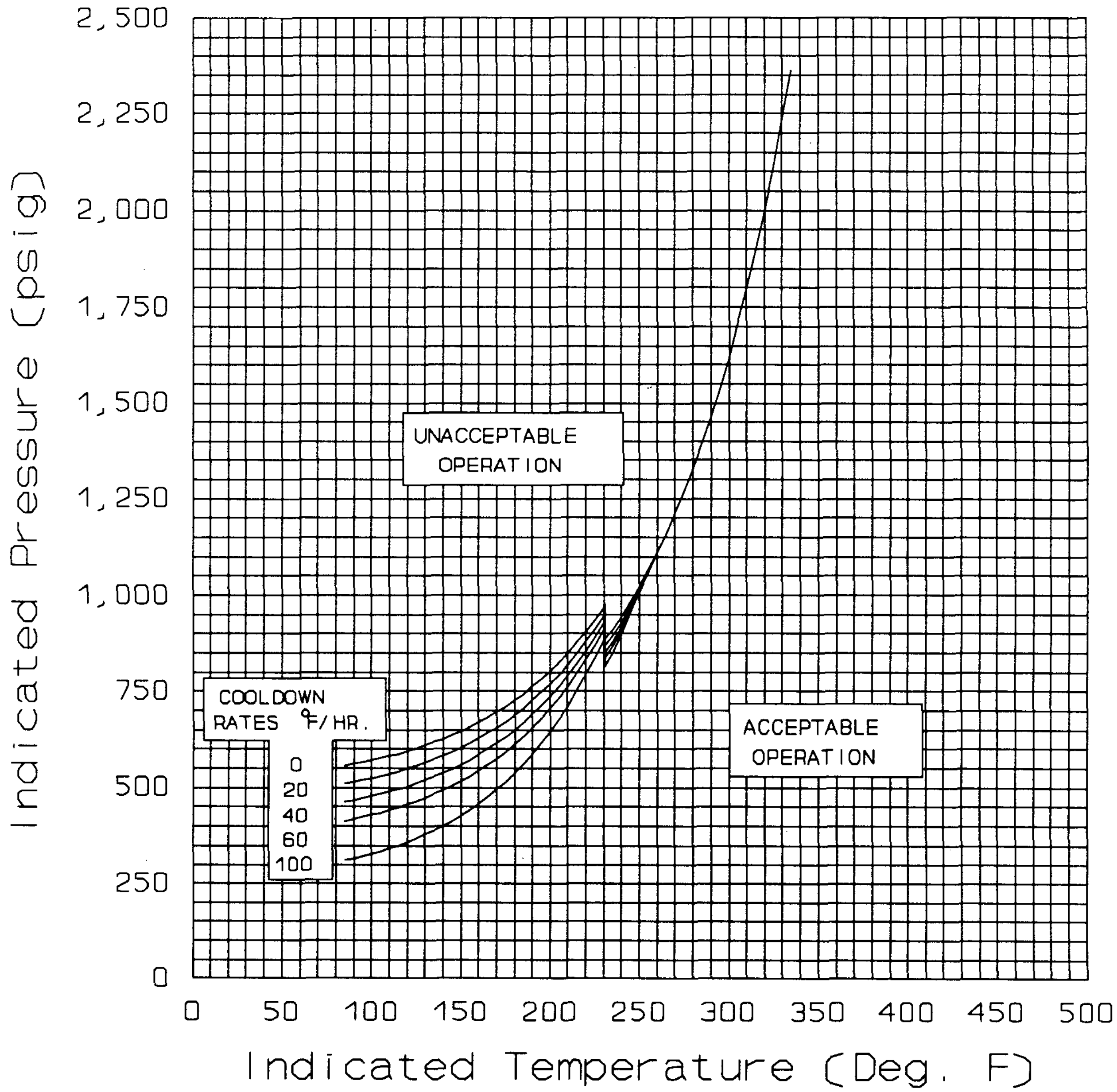

Figure 4 Watts Bar Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 7 EFPY (Without Margins for Instrumentation Errors) Including 10% Relaxation in Pressure for Temperatures <231°F per ASME Code Case N-514

6.0 REFERENCES

[1] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U.S. Nuclear Regulatory Commission, May, 1988.

[2] "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

[3] ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendixes, "Rules for Construction of Nuclear Power Plant Components, Appendix G, Protection Against Nonductile Failure", pp. 558-563, 1986 Edition, American Society of Mechanical Engineers, New York, 1986.

[4] Code of Federal Regulations, 10CFR50, Appendix G, "Fracture Toughness Requirements", U.S. Nuclear Regulatory Commission, Washington, D.C.,Federal Register, Vol. 48 No. 104, May 27, 1983.

[5] WCAP-13300, Revision 1, "Evaluation of Pressurized Thermal Shock for Watts Bar Unit 1", J. M. Chicots and P. L. Strauch, December 1992.

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

APPENDIX A

DATA POINTS FOR HEATUP AND COOLDOWN CURVES

TABLE A-1

HEATUP AND COOLDOWN CURVE DATA POINTS AT 7 EFPY (WITHOUT MARGINS)

Cooldown Curves Steady State		20 DEG CD		40 DEG CD		60 DEG CD		100 DEG CD	
T	P	T	P	T	P	T	P	T	P
60	490.13	60	447.48	60	404.02	60	359.39	60	266.59
65	492.77	65	450.24	65	406.70	65	362.11	65	269.45
70	495.72	70	453.18	70	409.67	70	365.11	70	272.56
75	498.88	75	456.38	75	412.91	75	368.41	75	276.04
80	502.28	80	459.81	80	416.40	80	371.97	80	279.82
85	505.94	85	463.54	85	420.21	85	375.89	85	283.98
90	509.87	90	467.55	90	424.32	90	380.12	90	288.53
95	514.09	95	471.89	95	428.79	95	384.74	95	293.54
100	518.64	100	476.55	100	433.61	100	389.73	100	298.99
105	523.52	105	481.60	105	438.77	105	395.18	105	304.97
110	528.77	110	487.03	110	444.40	110	401.06	110	311.45
115	534.42	115	492.81	115	450.52	115	407.40	115	318.55
120	540.39	120	499.12	120	457.10	120	414.30	120	326.20
125	546.91	125	505.93	125	464.24	125	421.80	125	334.59
130	553.93	130	513.26	130	471.92	130	429.89	130	343.66
135	561.47	135	521.17	135	480.25	135	438.60	135	353.56
140	569.58	140	529.67	140	489.20	140	448.06	140	364.21
145	578.31	145	538.75	145	498.80	145	458.32	145	375.84
150	587.56	150	548.62	150	509.23	150	469.38	150	388.41
155	597.63	155	559.26	155	520.50	155	481.35	155	402.06
160	608.47	160	570.71	160	532.63	160	494.17	160	416.74
165	620.12	165	582.92	165	545.63	165	508.12	165	432.73
170	632.50	170	596.19	170	559.73	170	523.16	170	449.91
175	645.97	175	610.49	175	574.96	175	539.30	175	468.61
180	660.44	180	625.85	180	591.21	180	556.81	180	488.79
185	675.98	185	642.29	185	608.89	185	575.72	185	510.54
190	692.57	190	660.09	190	627.89	190	595.96	190	534.09
195	710.56	195	679.12	195	648.27	195	617.94	195	559.46
200	729.70	200	699.74	200	670.32	200	641.47	200	586.80
205	750.49	205	721.92	205	693.97	205	667.02	205	616.49
210	772.78	210	745.65	210	719.52	210	694.35	210	648.35
215	796.64	215	771.30	215	746.96	215	723.96	215	682.74
220	822.23	220	798.79	220	776.41	220	755.76	220	719.95
225	849.99	225	828.31	225	808.37	225	789.97	225	760.01
230	879.55	230	860.26	230	842.54	230	826.78	230	803.10
235	911.51	235	894.47	235	879.31	235	866.62	235	849.59
240	945.77	240	931.19	240	918.85	240	909.35	240	899.62
245	982.53	245	970.69	245	961.60	245	955.36	245	953.53
250	1022.00	250	1013.15	250	1007.39	250	1004.81	250	1011.36
255	1064.40	255	1058.81	255	1056.69	255	1057.86		
260	1109.98	260	1107.90	260	1109.45				
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								

Heatup Curves 60 DEG		Crit Limit		100 DEG		Crit Limit		Leak Test Data	
T	P	T	P	T	P	T	P	T	P
60	443.05	314	0.00	60	401.48	314	0.00	293	2000
65	443.05	314	481.68	65	401.48	314	484.09	314	2485
70	443.05	314	468.96	70	401.48	314	468.61		
75	443.05	314	459.21	75	401.48	314	454.54		
80	443.05	314	452.27	80	401.48	314	442.62		
85	443.05	314	447.40	85	401.48	314	432.48		
90	443.05	314	444.50	90	401.48	314	424.13		
95	443.05	314	443.06	95	401.48	314	417.18		
100	443.05	314	443.05	100	401.48	314	411.73		
105	444.11	314	444.11	105	401.48	314	407.49		
110	446.29	314	446.29	110	401.48	314	404.46		
115	449.34	314	449.34	115	401.48	314	402.51		
120	453.33	314	453.33	120	401.48	314	401.54		
125	458.10	314	458.10	125	401.48	314	401.48		
130	463.72	314	463.72	130	402.34	314	402.34		
135	470.07	314	470.07	135	404.05	314	404.05		
140	477.24	314	477.24	140	406.56	314	406.56		
145	485.17	314	485.17	145	409.93	314	409.93		
150	493.84	314	493.84	150	414.14	314	414.14		
155	503.41	314	503.41	155	419.16	314	419.16		
160	513.86	314	513.86	160	425.01	314	425.01		
165	525.20	314	525.20	165	431.68	314	431.68		
170	537.41	314	537.41	170	439.16	314	439.16		
175	550.72	314	550.72	175	447.59	314	447.59		
180	565.12	314	565.12	180	456.94	314	456.94		
185	580.63	314	580.63	185	467.24	314	467.24		
190	597.27	314	597.27	190	478.55	314	478.55		
195	615.30	314	615.30	195	490.90	314	490.90		
200	634.60	314	634.60	200	504.29	314	504.29		
205	655.51	314	655.51	205	518.93	314	518.93		
210	678.00	314	678.00	210	534.82	314	534.82		
215	702.06	314	702.06	215	551.92	314	551.92		
220	727.92	314	727.92	220	570.53	314	570.53		
225	755.91	314	755.91	225	590.51	314	590.51		
230	785.79	314	785.79	230	612.21	314	612.21		
235	818.05	314	818.05	235	635.46	314	635.46		
240	852.61	314	852.61	240	660.68	314	660.68		
245	889.70	314	889.70	245	687.66	314	687.66		
250	929.52	314	929.52	250	716.88	314	716.88		
255	972.27	314	972.27	255	748.15	314	748.15		
260	1018.19	314	1018.19	260	781.76	314	781.76		
265	1067.49	314	1067.49	265	818.06	314	818.06		
270	1120.42	314	1120.42	270	856.92	314	856.92		
275	1177.15	315	1177.15	275	898.66	315	898.66		
280	1237.98	320	1237.98	280	943.47	320	943.47		
285	1303.13	325	1303.13	285	991.58	325	991.58		
290	1373.09	330	1373.09	290	1043.20	330	1043.20		
295	1447.99	335	1447.99	295	1098.55	335	1098.55		
300	1527.97	340	1527.97	300	1157.76	340	1157.76		
305	1612.31	345	1612.31	305	1221.51	345	1221.51		
310	1686.16	350	1686.16	310	1289.68	350	1289.68		
315	1765.15	355	1765.15	315	1362.69	355	1362.69		
320	1849.89	360	1849.89	320	1440.81	360	1440.81		
325	1940.35	365	1940.35	325	1524.60	365	1524.60		
330	2037.07	370	2037.07	330	1614.19	370	1614.19		
335	2140.54	375	2140.54	335	1709.92	375	1709.92		
340	2250.90	380	2250.90	340	1812.16	380	1812.16		
345	2368.62	385	2368.62	345	1921.54	385	1921.54		
				350	2038.25	390	2038.25		
				355	2162.68	395	2162.68		
				360	2295.42	400	2295.42		
				365	2436.49	405	2436.49		

TABLE A-2

HEATUP AND COOLDOWN CURVE DATA POINTS AT 7 EFPY(WITHOUT MARGINS)

Including 10% Relaxation in Pressure for Temperatures <231°F per ASME Code Case N-514

Cooldown Curves Steady State		20 DEG CD		40 DEG CD		60 DEG CD		100 DEG CD		Heatup Curves 60 DEG		Crit Limit		100 DEG		Crit Limit		Leak Test Data	
T	P	T	P	T	P	T	P	T	P	T	P	T	P	T	P	T	P	T	P
85	556.53	85	509.89	85	462.23	85	413.48	85	312.38	85	505.54	314	0 00	85	462 14	314	0 00	293	2000
90	560.86	90	514.31	90	466.75	90	418.13	90	317.38	90	505 54	314	459 68	90	462 14	314	420 13	314	2485
95	565.50	95	519.08	95	471.67	95	423.21	95	322.89	95	505.54	314	459 68	95	462.14	314	420 13		
100	570.50	100	524.21	100	476.97	100	428 70	100	328.89	100	505 54	314	459.68	100	462.14	314	420 13		
105	575.87	105	529.76	105	482.65	105	434.70	105	335.47	105	505.54	314	459 68	105	462 14	314	420.13		
110	581.65	110	535.73	110	488.84	110	441.17	110	342.60	110	505 54	314	459.68	110	462.14	314	420 13		
115	587.86	115	542.09	115	495.57	115	448.14	115	350.41	115	506.46	314	460.42	115	462 14	314	420 13		
120	594.43	120	549.03	120	502.81	120	455.73	120	358.82	120	508 70	314	462.45	120	462.14	314	420 13		
125	601.60	125	556.52	125	510.66	125	463.98	125	368.05	125	512.30	314	465 73	125	462 14	314	420 13		
130	609.32	130	564.59	130	519.11	130	472.88	130	378.03	130	517.01	314	470 01	130	462.14	314	420 13		
135	617.62	135	573.29	135	528.28	135	482.46	135	388.92	135	522.91	314	475.37	135	462 14	314	420 13		
140	626.54	140	582.64	140	538.12	140	492.87	140	400.63	140	529.78	314	481 62	140	463.16	314	421 05		
145	636.14	145	592.63	145	548.68	145	504 15	145	413.42	145	537.76	314	488 87	145	465.31	314	423 01		
150	646.32	150	603.48	150	560.15	150	516.32	150	427.25	150	546.60	314	496 91	150	468.50	314	425.91		
155	657.39	155	615.19	155	572.55	155	529.49	155	442.27	155	556.61	314	506.01	155	472.79	314	429 81		
160	669.32	160	627.78	160	585.89	160	543.59	160	458.41	160	567.63	314	516.03	160	478.08	314	434.62		
165	682.13	165	641.21	165	600.19	165	558.93	165	476.00	165	579.76	314	527.05	165	484.37	314	440 34		
170	695.75	170	655.81	170	615.70	170	575.48	170	494.90	170	592.85	314	538.95	170	491.77	314	447 06		
175	710.57	175	671.54	175	632.46	175	593 23	175	515.47	175	607.24	314	552.04	175	500.26	314	454.78		
180	726.48	180	688.44	180	650.33	180	612.49	180	537.67	180	622.84	314	566.22	180	509.82	314	463 47		
185	743.58	185	706.52	185	669.78	185	633.29	185	561.59	185	639.74	314	581 58	185	520.52	314	473 20		
190	761.83	190	726.10	190	690.68	190	655 56	190	587 50	190	657.87	314	598.06	190	532 38	314	483 98		
195	781.62	195	747.03	195	713.10	195	679 73	195	615.41	195	677.60	314	616.00	195	545 38	314	495 80		
200	802.67	200	769.71	200	737.35	200	705.62	200	645.48	200	698 70	314	635 18	200	559 72	314	508.84		
205	825.54	205	794.11	205	763.37	205	733.72	205	678 14	205	721.62	314	656.02	205	575.42	314	523 11		
210	850.06	210	820.22	210	791.47	210	763.79	210	713.19	210	746.26	314	678.42	210	592.41	314	538 55		
215	876.30	215	848.43	215	821.66	215	796.36	215	751.01	215	772.68	314	702.44	215	611 00	314	555 45		
220	904.45	220	878.67	220	854.05	220	831.34	220	791.95	220	801.05	314	728.23	220	631.16	314	573 78		
225	934.99	225	911.14	225	889.21	225	868.97	225	836.01	225	831.80	314	756.18	225	652.87	314	593 52		
230	967.51	230	946.29	230	926.79	230	909.46	230	883.41	230	864.62	314	786.02	230	676.49	314	614 99		
231	974.53	231	953.81	231	934.88	231	918.23	231	893.64	231	871 72	314	792.47	231	681.56	314	619.60		
231	885.94	231	867.10	231	849.89	231	834 75	231	812.40	231	792.47	314	792.47	231	619 60	314	619.60		
235	911.51	235	894.47	235	879.31	235	866.62	235	849.59	235	818.26	314	818.26	235	638.04	314	638.04		
240	945.77	240	931.19	240	918.85	240	909.35	240	899.62	240	852.78	314	852.78	240	663.06	314	663.06		
245	982.53	245	970.69	245	961.60	245	955.36	245	953.53	245	889.86	314	889.86	245	689.89	314	689.89		
250	1022.00	250	1013.15	250	1007.39	250	1004.81	250	1011.36	250	929.65	314	929.65	250	718.94	314	718.94		
255	1064.40	255	1058.81	255	1056.69	255	1057.86			255	972.39	314	972.39	255	750 08	314	750.08		
260	1109.98	260	1107.90	260	1109.45					260	1018.29	314	1018.29	260	783.55	314	783.55		
265	1158.95									265	1067 58	314	1067.58	265	819.71	314	819 71		
270	1211.57									270	1120.50	314	1120.50	270	858.48	314	858.48		
275	1268.05									275	1177.22	315	1177.22	275	900.14	315	900 14		
280	1328.61									280	1238.04	320	1238.04	280	944.85	320	944 85		
285	1393.44									285	1303.18	325	1303 18	285	992.88	325	992 88		
290	1463.21									290	1373.13	330	1373.13	290	1044.38	330	1044 38		
295	1537.95									295	1448.03	335	1448.03	295	1099.66	335	1099.66		
300	1618.11									300	1528.00	340	1528 00	300	1158.84	340	1158 84		
305	1703.92									305	1612.29	345	1612.29	305	1222.54	345	1222 54		
310	1795.88									310	1686.18	350	1686 18	310	1290.65	350	1290.65		
315	1894.39									315	1765 13	355	1765.13	315	1363.61	355	1363 61		
320	1999.72									320	1849 92	360	1849 92	320	1441.68	360	1441.68		
325	2112.46									325	1940.33	365	1940.33	325	1525.42	365	1525.42		
330	2232.88									330	2037 10	370	2037 10	330	1614 97	370	1614.97		
335	2361.41									335	2140 52	375	2140.52	335	1710.66	375	1710 66		
										340	2250 92	380	2250.92	340	1812 86	380	1812 86		
										345	2368 60	385	2368.60	345	1922 22	385	1922 22		
														350	2038 39	390	2038 39		
														355	2163.29	395	2163 29		
														360	2295 99	400	2295 99		
														365	2437 04	405	2437 04		

Enclosure 4

Watts Bar Unit 1

Pressure and Temperature Limits Report

Revision 4