H.B.ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

PAGE CHANGE INSTRUCTIONS

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TECHNICAL SPECIFICATION PAGES

LIST OF FIGURES

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1~~a~~ and Figure 3.1-2~~a~~ ~~(for vessel exposure up to 12.5 EFPY) or Figure 3.1-1b and Figure 3.1-2b (~~for vessel exposure up to ~~15~~ 24 EFPY). ~~The 15 EFPY curves may be used for operation prior to the end of 12.5 EFPY.~~ These limitations are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2~~a or 3.1-2b (as appropriate)~~. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2~~a or Figure 3.1-2b~~ may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1~~a or Figure 3.1-1b (as appropriate)~~ is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:

1. Cooldown and depressurize the RCS or

2. Heatup the RCS to above 350°F.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 120°F.

3.1.2.3 The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

3.1.2.4 Figures 3.1-1~~b~~ and 3.1-2~~b~~ shall be updated periodically in accordance with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposure for which the figures apply.

a. At least 60 days before the end of the integrated power period for which Figures 3.1-1~~b~~ and 3.1-2~~b~~ apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and in accordance with applicable appendices of 10CFR50. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

b. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility and impact toughness under certain conditions of irradiation. Accompanying a decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperature 60°F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of Section III of the ASME Boiler and Pressure Vessel ~~Code~~. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The Certified Material Test Reports (CMTRs) for the original steam generators provided records of Charpy V-notch tests performed at +10°F. Acceptable Charpy V-notch tests of +10°F indicate RT_{NDT} is at or below this temperature. The steam generator lower assemblies were replaced in 1984 and the material tests results indicate the highest RT_{NDT} is 60°F or below. The ASME code recommends that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60F°, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure.

V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial $+\Delta RT_{NDT}$) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperture, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of

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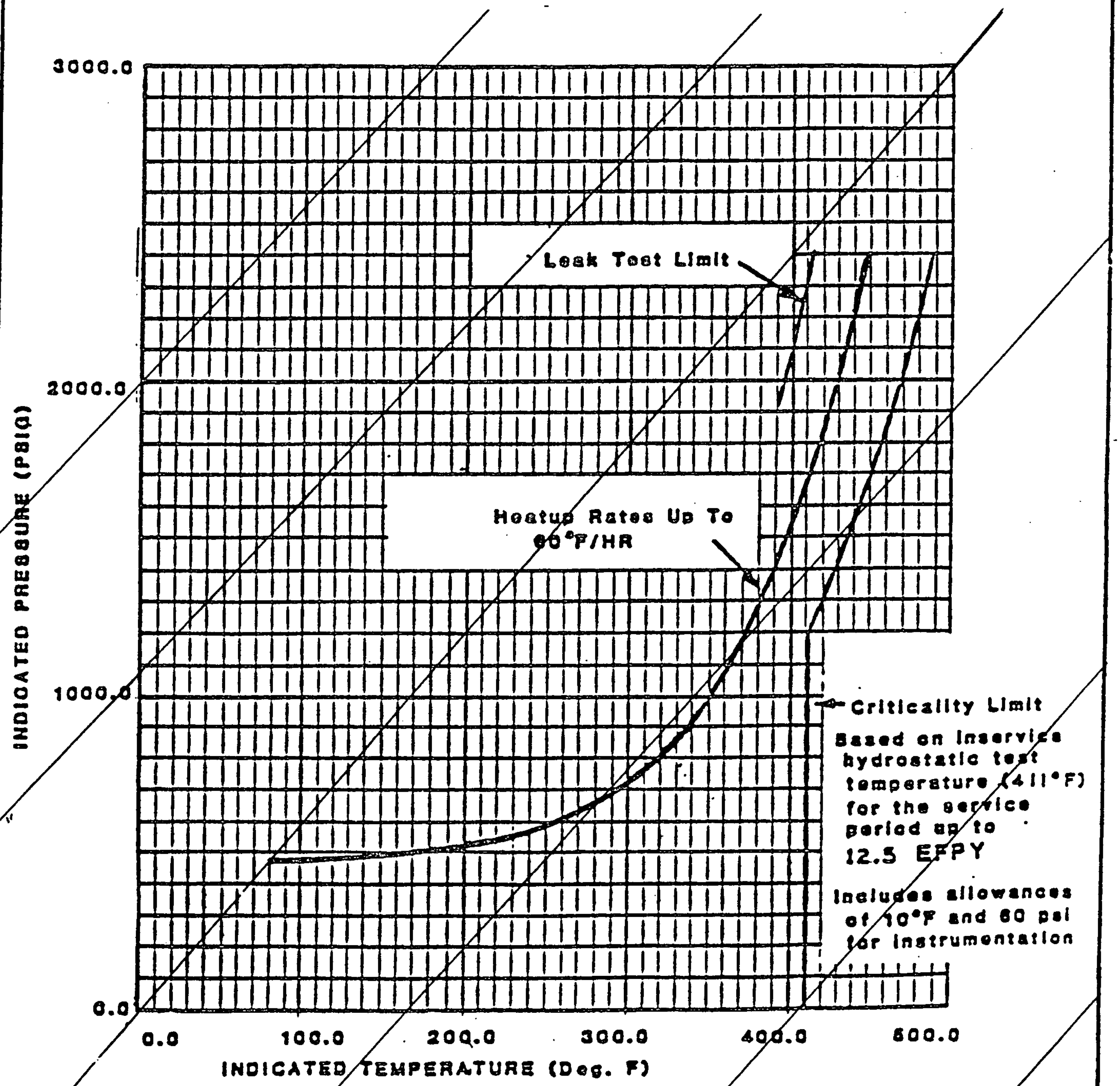

H.B. ROBINSON Unit #2 Carolina Power & Light Company Technical Specifications	Reactor Coolant System Heatup Limitations - Applicable Up To 12.5 EFPY	FIGURE 3.1-1.a

Amendment No. ~~113~~

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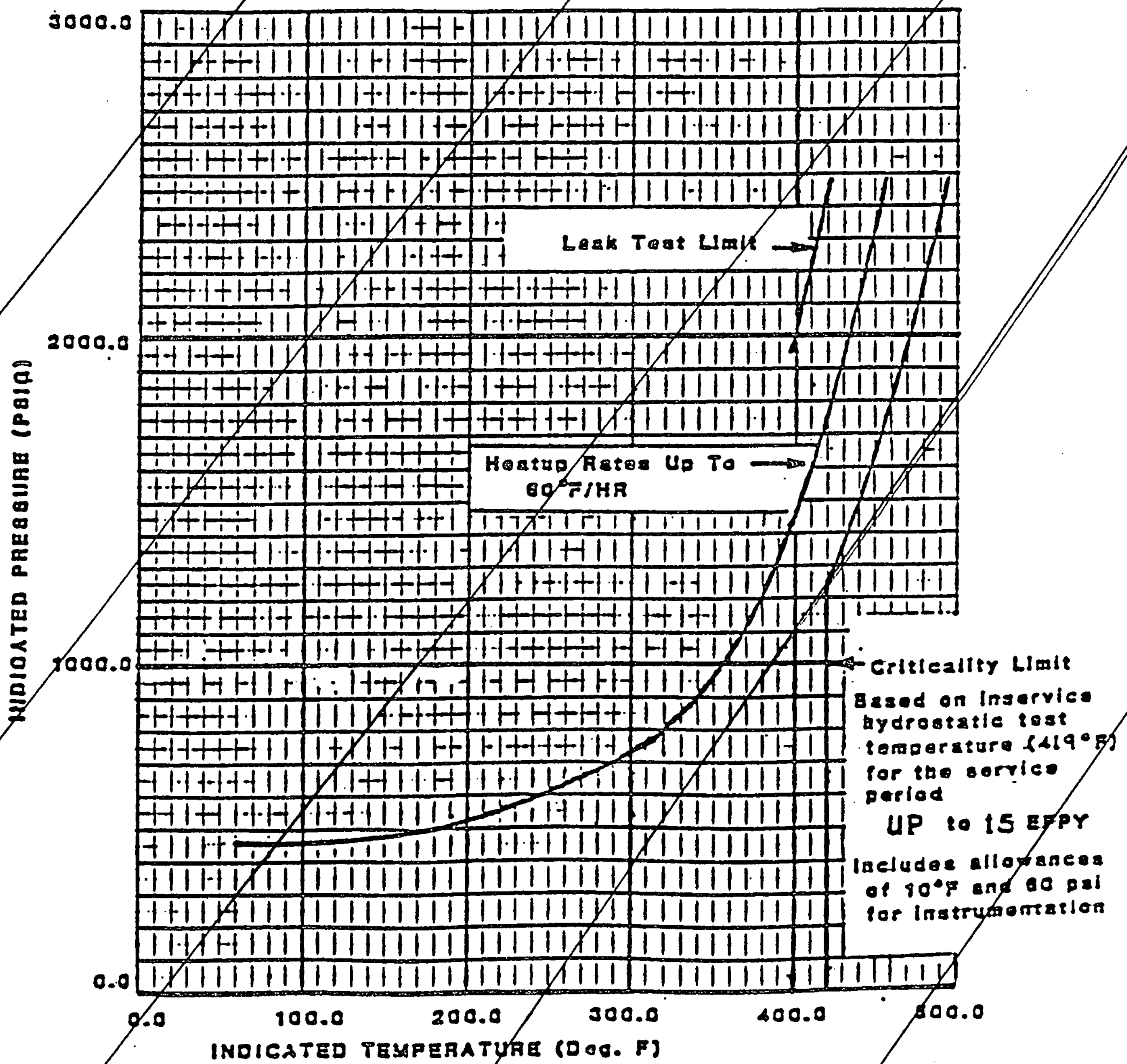

H.B. ROBINSON Unit #2 Carolina Power & Light Company Technical Specifications	Reactor Coolant System Heatup Limitations - Applicable Up To 15 EFPY	FIGURE 3.1-1.b

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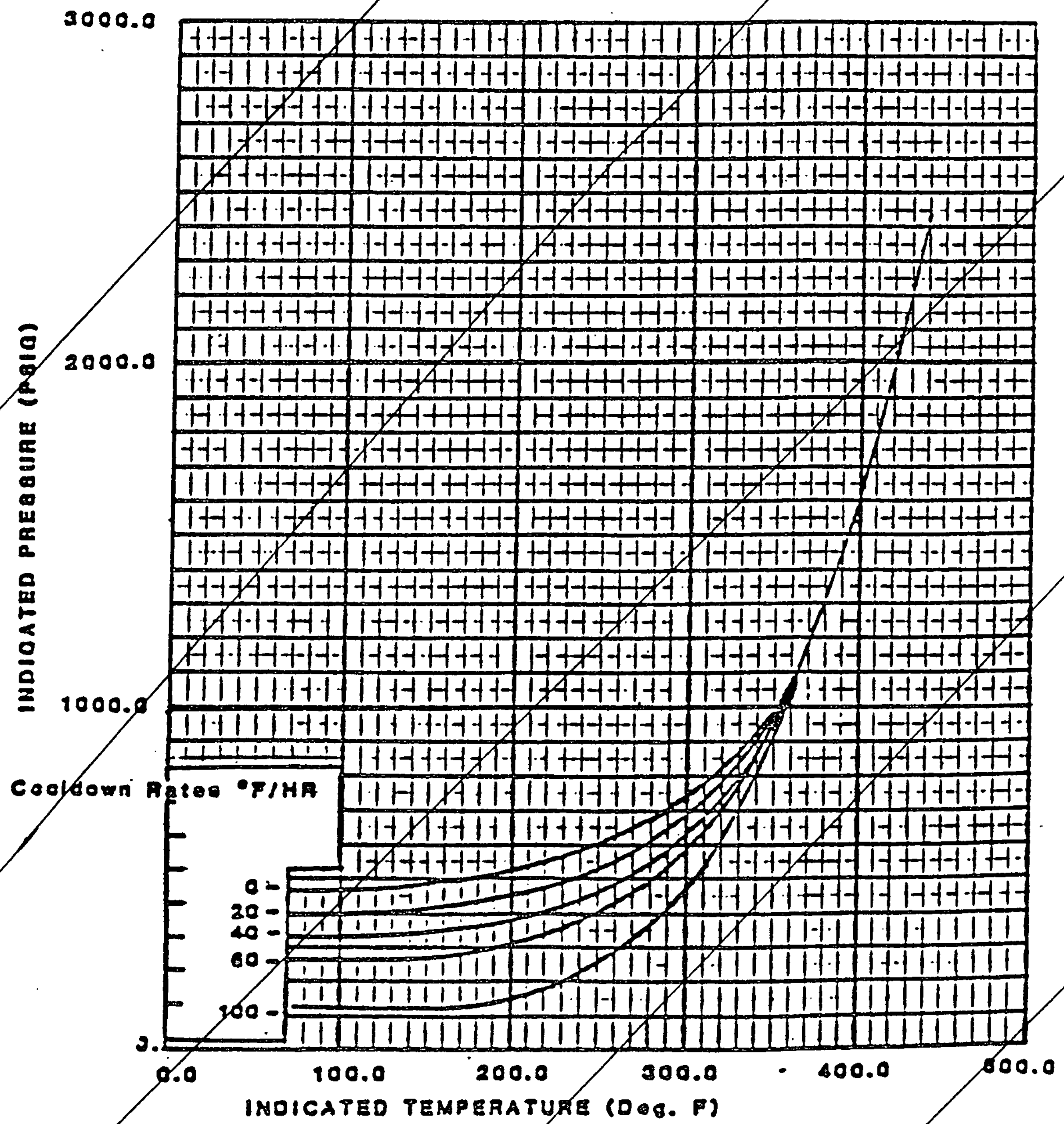
MATERIALS PROPERTIES BASIS

Controlling Material	:	Weld Metal
Copper Content	:	0.35 wt.%
Phosphorus Content	:	0.012 wt.%
RT_{NDT} Initial	:	0°F
RT_{NDT} After 12.5 EFPY	:	1/4 T, 282°F 3/4 T, 139°F

Curves applicable for cooldown rates up to 100°F/HR for the service period up to 12.5 EFPY.

Includes 10° and 60 psi allowance for instrumentation.



H.B. ROBINSON Unit #2 Carolina Power & Light Company Technical Specifications	Reactor Coolant System Cooldown Limitations Applicable Up To 12.5 EFPY	FIGURE 3.1-2.a

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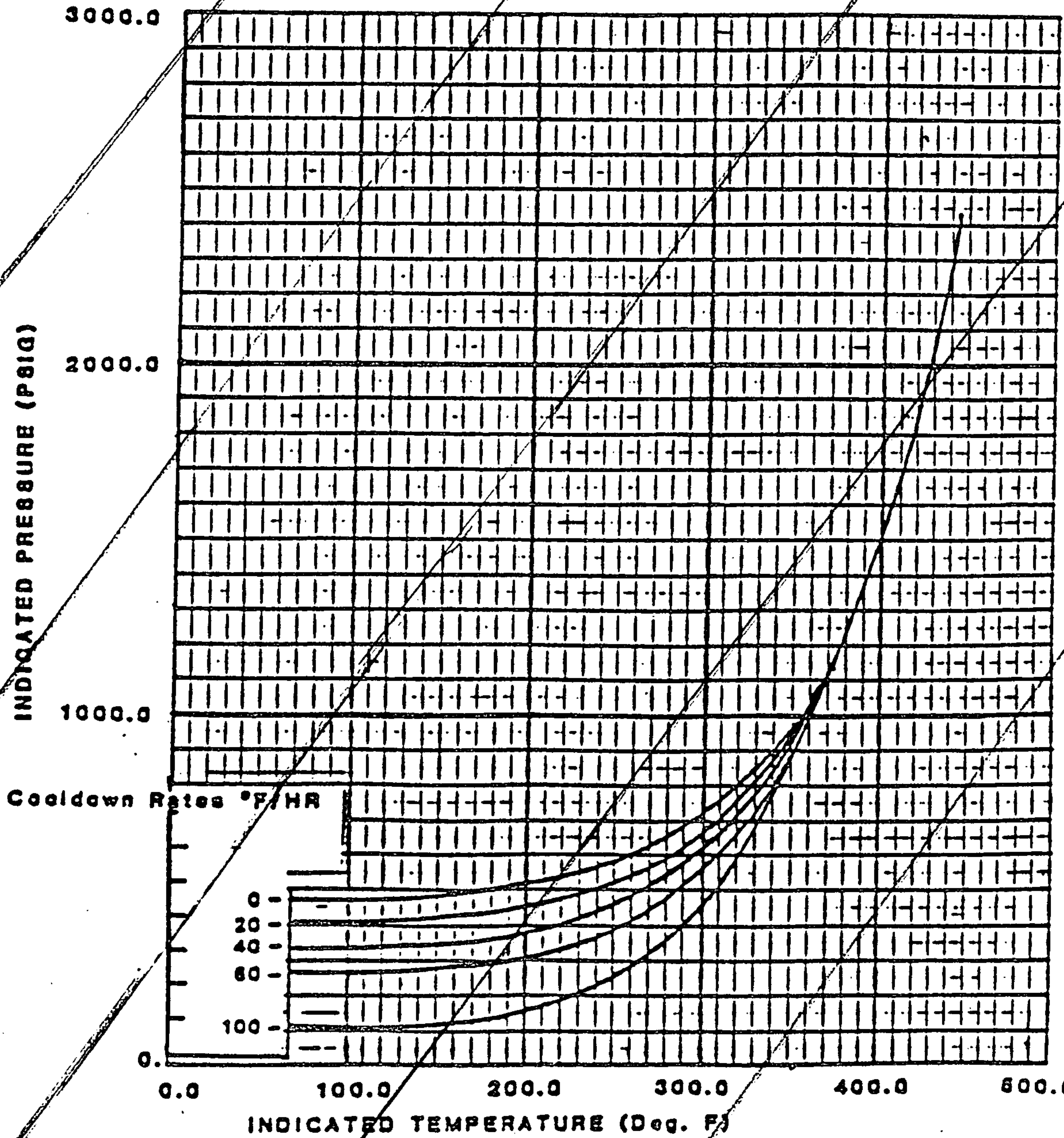
MATERIALS PROPERTIES BASIS

Controlling Material : Weld Metal
Copper Content : 0.35 wt.%
Phosphorus Content : 0.012 wt.%
RT_{NDT} Initial : 0°F
RT_{NDT} After 15 EFPY: 1/4 T, 290°F
3/4 T, 149°F

Curves applicable for cooldown rates up to 100°F/HR for the service period up to 15 EFPY.

Includes 10° and 60 psi allowance for instrumentation.



H.B. ROBINSON Unit #2 Carolina Power & Light Company Technical Specifications	Reactor Coolant System Cooldown Limitations Applicable Up To 15 EFPY	FIGURE 3.1-2.b

LIST OF FIGURES

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1 and Figure 3.1-2 (for vessel exposure up to 24 EFPY). These limitations are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2. This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1 is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:

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V-notch 30 ft-lb temperature ($\Delta\ RT_{NDT}$) due to irradiation is added to the original ΔRT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

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As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of

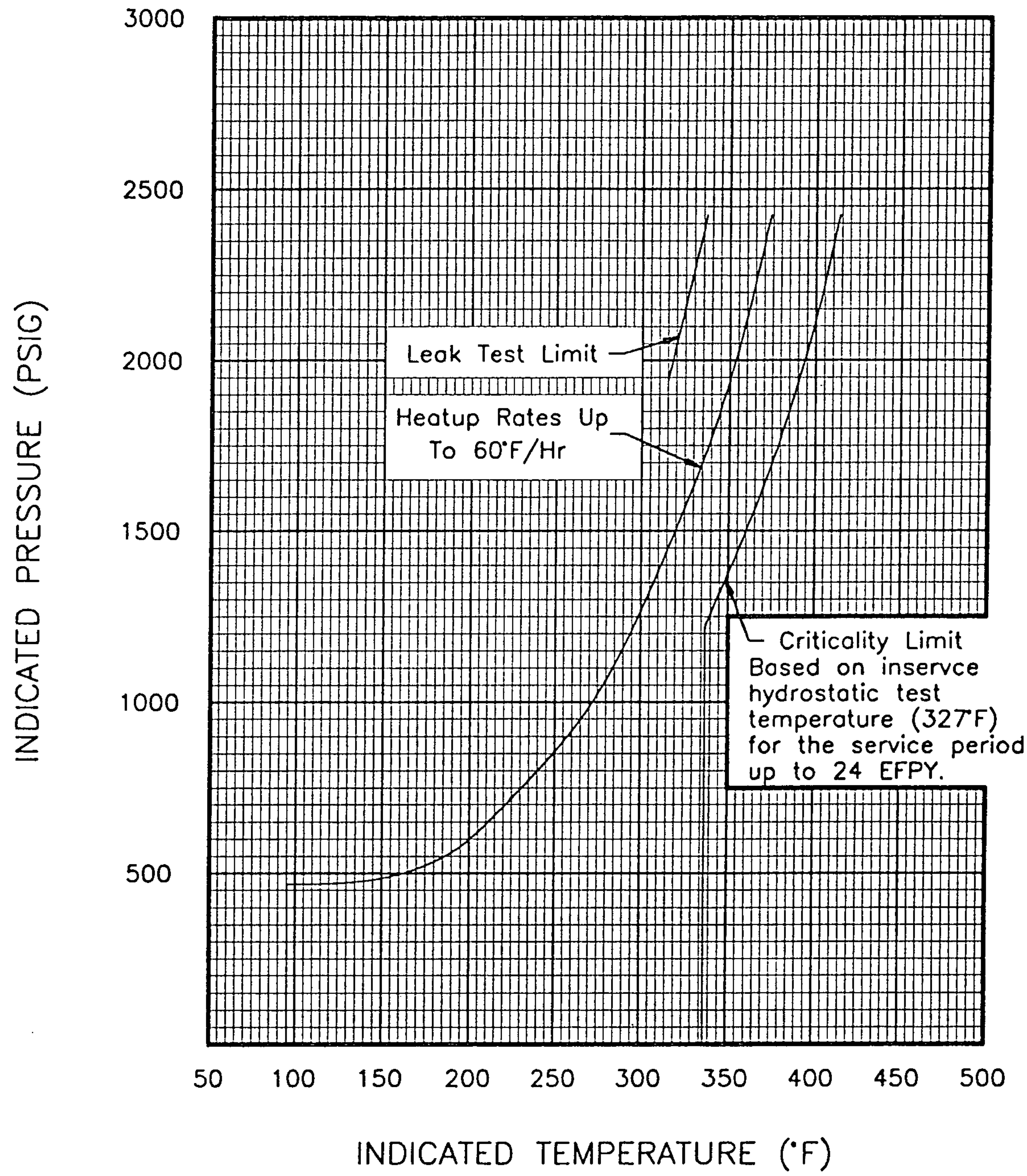
MATERIALS PROPERTIES BASE

Controlling Material : Lower Circumferential Weld
Copper Content : 0.20 wt.%
Nickel Content : 1.06 wt.%
RT_{NDT} Initial : −80°F

RT_{NDT} After 24 EFPY : 1/4 T. 207.83°F
3/4 T. 137.15°F

Curves applicable for heatup rates up to 60°F/Hr for the service period up to 24 EFPY.

Includes +10°F and −60 PSIG allowance for instrumentation error.



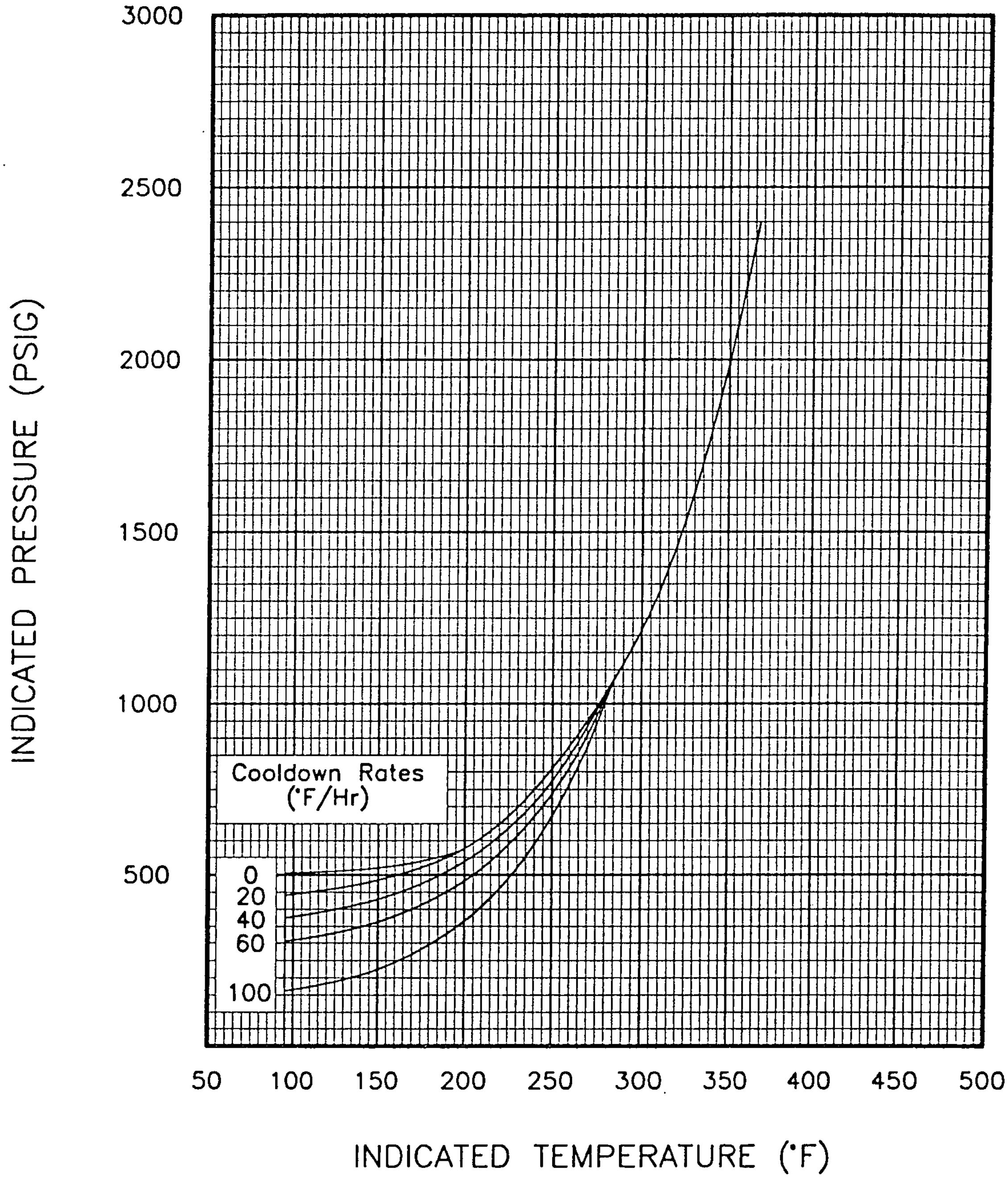
H.B. Robinson Unit #2 CP&L CAROLINA POWER & LIGHT COMPANY Technical Specifications	Reactor Coolant System Heatup Limitations Applicable Up To 24 EFPY	FIGURE 3.1-1

MATERIALS PROPERTIES BASE

Controlling Material	:	Lower Circumferential Weld
Copper Content	:	0.20 wt.%
Nickel Content	:	1.06 wt.%
RT_{NDT} Initial	:	−80°F
RT_{NDT} After 24 EFPY	:	1/4 T. 207.83°F 3/4 T. 137.15°F

Curves applicable for cooldown rates up to 100°F/Hr for the service period up to 24 EFPY.

Includes +10°F and −60 PSIG allowance for instrumentation error.



H.B. Robinson Unit #2 CP&L CAROLINA POWER & LIGHT COMPANY Technical Specifications	Reactor Coolant System Cooldown Limitations Applicable Up To 24 EFPY	FIGURE 3.1-2

Pressurized Thermal Shock

In accordance with the revised Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61, effective June 14, 1991, the reactor vessel beltline materials were evaluated to determine the impact of the revised rule. A summary of the pertinent data required by revised Section 50.61(b)(1) for both the current date and the HBR2 Operating License expiration date is included in the table of this Enclosure 6.

The PTS rule requires calculation of RT_{PTS} values for all beltline materials for both (1) End-Of-License (EOL) and (2) the time of submittal. August of 1993 was chosen as the time of submittal to coincide with the planned Technical Specification submittal of updated Pressure - Temperature limit curves.

The calculation results show the upper circumferential weld joint (Weld 10-273) to be the most limiting beltline material for PTS considerations. However, as shown in the attached table, none of the calculated RT_{PTS} values exceed the 10 CFR 50.61 screening criterion. Therefore, as concluded in 1991, a response to the NRC by December 16, 1991, in accordance to the PTS rule, was not required.

The methods for calculation of RT_{PTS} do not currently permit the use of plant specific materials surveillance program data, as allowed by the methods for calculating RT_{NDT} . Because the upper weld (10-273) has been demonstrated as applicable to the surveillance program, plant specific data was used for RT_{NDT} calculations. This information makes the lower circumferential weld (11-273) most limiting for RT_{NDT} purposes and the development of Pressure-Temperature curves, even though the upper circumferential weld (10-273) is calculated to be most limiting for PTS considerations.

PREDICTED REFERENCE PTS TEMPERATURES FOR H. B. ROBINSON UNIT 2 BELTLINE MATERIALS (BASED ON 10 CFR 50.61, AS ISSUED MAY 1991)

PLATE OR WELD	%Cu	%Ni	CHEMISTRY FACTOR (CF)	INITIAL REFERENCE TEMP. (I)	MARGIN OF UNCERTAINTY (M)	ID SURFACE FLUENCE (f) ($X10^{19} n/cm^2$) (NOTE 1)	SHIFT IN REFERENCE TEMP. (ΔRT_{PTS}) (NOTES 1,2)	ADJUSTED REFERENCE TEMP. (RT_{PTS}) (NOTES 1,2)	PTS RULE SCREENING CRITERION TEMP.
Weld 10-273	0.34	0.66	217.7	-56 °F	66 °F	1.83 (.951)	254 °F (215 °F)	264 °F (225 °F)	300 °F
Weld 11-273	0.17	0.92	197.8	-80 °F	56 °F	2.01 (1.61)	235 °F (224 °F)	211 °F (200 °F)	300 °F
Weld 1-273A	0.22	.054	101.05	-56 °F	66 °F	.63 (.327)	88 °F (70 °F)	98 °F (80 °F)	270 °F
Weld 1-273B	0.22	.054	101.05	-56 °F	66 °F	1.34 (.696)	109 °F (91 °F)	119 °F (101 °F)	270 °F
Weld 1-273C	0.22	.054	101.05	-56 °F	66 °F	.285 (.148)	66 °F (51 °F)	76 °F (61 °F)	270 °F
Weld 2-273A	0.22	.054	101.05	-56 °F	66 °F	3.93 (.373)	137 °F (74 °F)	147 °F (84 °F)	270 °F
Weld 2-273B	0.22	.054	101.05	-56 °F	66 °F	.752 (.413)	93 °F (76 °F)	103 °F (86 °F)	270 °F
Weld 2-273C	0.22	.054	101.05	-56 °F	66 °F	1.66 (.912)	115 °F (98 °F)	125 °F (108 °F)	270 °F
Weld 3-273A	0.22	.054	101.05	-56 °F	66 °F	2.01 (1.61)	120 °F (114 °F)	130 °F (124 °F)	270 °F
Weld 3-273B	0.22	.054	101.05	-56 °F	66 °F	.44 (.353)	78 °F (72 °F)	88 °F (82 °F)	270 °F
Weld 3-273C	0.22	.054	101.05	-56 °F	66 °F	.44 (.353)	78 °F (72 °F)	88 °F (82 °F)	270 °F

PREDICTED REFERENCE PTS TEMPERATURES FOR H. B. ROBINSON UNIT 2 BELTLINE MATERIALS (BASED ON 10 CFR 50.61, AS ISSUED MAY 1991)

PLATE OR WELD	%Cu	%Ni	CHEMISTRY FACTOR (CF)	INITIAL REFERENCE TEMP. (I)	MARGIN OF UNCERTAINTY (M)	ID SURFACE FLUENCE (f) ($X10^{19} n/cm^2$) (NOTE 1)	SHIFT IN REFERENCE TEMP. (ΔRT_{PTS}) (NOTES 1,2)	ADJUSTED REFERENCE TEMP. (RT_{PTS}) (NOTES 1,2)	PTS RULE SCREENING CRITERION TEMP.
Plate W10201-1	0.13	0.11	63	69 °F	34 °F	1.83 (.951)	73 °F (62 °F)	176 °F (165 °F)	270 °F
Plate W10201-2	0.15	0.25	84.75	30 °F	34 °F	1.83 (.951)	99 °F (84 °F)	163 °F (148 °F)	270 °F
Plate W10201-3	0.11	0.08	51.8	36 °F	34 °F	1.83 (.951)	60 °F (51 °F)	130 °F (121 °F)	270 °F
Plate W10201-4	0.12	0.09	57.1	20 °F	34 °F	4.82 (2.65)	80 °F (72 °F)	134 °F (126 °F)	270 °F
Plate W10201-5	0.10	0.12	51.2	20 °F	34 °F	4.82 (2.65)	71 °F (65 °F)	125 °F (119 °F)	270 °F
Plate W10201-6	0.09	0.09	44.2	45 °F	34 °F	4.82 (2.65)	62 °F (56 °F)	141 °F (135 °F)	270 °F
Plate W9807-3	0.12	0.10	58	50 °F	34 °F	2.01 (1.61)	69 °F (66 °F)	153 °F (150 °F)	270 °F
Plate W9807-5	0.15	0.10	70.5	33 °F	34 °F	2.01 (1.61)	84 °F (80 °F)	151 °F (147 °F)	270 °F
Plate W9807-9	0.14	0.15	70.5	9 °F	34 °F	2.01 (1.61)	84 °F (80 °F)	127 °F (123 °F)	270 °F

NOTES:

(1) Values are specified for both End-of-License and August 1993 (in parenthesis). End-of-License values are based on maximum anticipated 29 effective full power years of operation through July 10, 2010.

(2) Predicted "Shift in Reference Temperature (ΔRT_{PTS})" and "Adjusted Reference Temperature (RT_{PTS})" were calculated using Equations 1 and 2 in 10 CFR 50.61 (shown below). Equation variables are as noted in the table headings, above.

$$\Delta RT_{PTS} = (CF) f^{(0.28 - 0.10 \log f)}$$

$$RT_{PTS} = I + M + \Delta RT_{PTS}$$