APPENDIX D

WCAP-1530, Revision 1 dated April 2001
"Surry Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation"

WCAP-15130, Revision 1

Surry Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves For Normal Operation

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April 2001

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PREFACE

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Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15130 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15130 Rev. 0.

Note that only the heatup curves and associated data point tables have changed. The cooldown curves and data points remain valid and were not changed.

LEGAL NOTICE

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1. INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference temperature adjusted for irradiation effects) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin term to accommodate uncertainties. 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 unirradiated RT_{NDT} (IRT_{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 (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

2. PURPOSE

Virginia Power, as a member of the WOG Reactor Vessel 60-year Minigroup, has contracted Westinghouse to generate new heatup and cooldown curves applicable to Surry Units 1 and 2 for plant life extension. The heatup and cooldown curves are generated with margins of 0°F and 0 psi for instrumentation errors. The curves include a hydrostatic leak test limit curve from 2485 to 2000 psig and pressure-temperature limits for the vessel flange regions per the requirements of 10 CFR Part 50, Appendix G[2].

The purpose of this report is to present the calculations and the development of Surry Units 1 and 2 heatup and cooldown curves for plant life extension. This report documents the calculated adjusted reference temperature (ART) values, following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

3. CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 End of Life Pressure Temperature Limits

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[2] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[3], contains the conservative methods of analysis.

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_{Ia}, for the metal temperature at that time. K_{Ia} is obtained from the reference fracture toughness curve, defined in Appendix G of the ASME Code, Section XI. The K_{Ia} curve is given by the following equation:

$$K_{Ia} + 26.78 + 1.233 * e^{[0.0145(T - RT_{NDT} + 160)]} \tag{3-1}$$

where,

K_{Ia} = 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:

$$C * K_{Im} + K_{It} < K_{Ia} \tag{3-2}$$

where,

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

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

K_{Ia} = reference stress intensity factor as a function of the metal temperature and 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_{Ia} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, 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 1/4T 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 DT (differential temperature) developed during cooldown results in a higher value 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 is 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 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 1/4T 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_{Ia} for the 1/4T crack during heatup is lower than the K_{Ia} for the 1/4T 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_{Ia} 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 1/4T 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 1/4T flaw located at the 1/4T location from the outside surface 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.

10 CFR Part 50, Appendix G 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 (3107 psig), which is 621 psig[4] for the Surry Units 1 and 2 reactor vessel.

The limiting unirradiated RT_{NDT} of 10°F occurs in the vessel flange of the Surry Units 1 and 2 reactor vessel, so the minimum allowable temperature of this region is 130°F at pressures greater than 621 psig with uncertainties of 0°F and 0 psi. This limit is reflected in the heatup and cooldown curves shown in Figures 9-1 to 9-8.

3.2 End of License Renewal Pressure Temperature Limits

For end of license renewal, the Surry Units 1 and 2 pressure temperature limits are developed using the ASME Code Section XI K_{IC} fracture toughness methodology.

K_{Ic} is obtained from the reference fracture toughness curve and is given by the following equation:

$$K_{IC} = 33.2 + 20.734e^{[0.02(T - RT_{NDT})]} \qquad (3\text{-}3)$$

where,

K_{Ic} = 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 as follows:

$$C * K_{Im} + K_{It} < K_{Ic} \tag{3-4}$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress
K_{It} = stress intensity factor caused by the thermal gradients
K_{Ic} = 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

The balance of the discussion for the end of license curves based on K_{Ia} fracture toughness methodology, in Section 3.1 above, remains applicable to end of license renewal pressure temperature limits.

4. CHEMISTRY FACTOR DETERMINATION

4.1 Surveillance Data Evaluation

Table 4-1 contains the available surveillance data for Surry Units 1 and 2. Table 4-2 contains the representative data ultimately used to assess the material properties of the Surry Units 1 and 2 reactor vessel.

If the irradiation environment (i.e., irradiation temperature and NSSS vendor) for one source is more similar to the irradiation environment of the plant being assessed than other sources, the data from the source with the irradiation environment most similar to that of the plant being assessed should be used to assess the integrity of the vessel. Thus, if plant specific data are available, these data are considered the most applicable to the beltline material being evaluated.

<u>SA-1585/SA-1650 Assesssment</u>:
Weld wire 72445 surveillance data is not available from a Surry plant specific surveillance program. However, it is available from the Point Beach Unit 1 (Westinghouse NSSS) surveillance program and from several BWOG integrated surveillance program capsules irradiated in Crystal River Unit 3 (B&W NSSS). The irradiation environment of the Point Beach Unit 1 surveillance capsules more closely approximates the irradiation environment of Surry Units 1 and 2 because (a) Point Beach and Surry are both Westinghouse NSSS plants, and (b) the irradiation temperatures of the Surry and Point Beach surveillance capsules are closer than the irradiation temperatures of Surry and Crystal River. Therefore, the data derived from the Point Beach Unit 1 surveillance program is used to assess the integrity of the Surry Units 1 and 2 beltline material SA-1585.

<u>SA1526 Assessment</u>:
Weld wire 299L44 surveillance data is available from the Surry Unit 1 plant specific surveillance program. In addition, weld wire 299L44 surveillance data is available from BWOG integrated surveillance program surveillance capsules irradiated in Crystal River Unit 3. Among the BWOG integrated surveillance program capsules is a capsule which contains materials previously irradiated in Three Mile Island Unit 2 (i.e., TMI2-LG1, WF-25). The irradiation environment of the Surry Unit 1 capsules most closely approximates the irradiation environment of the Surry Unit 1 reactor vessel beltline materials. Therefore, the data derived from the Surry Unit 1 plant specific surveillance program is used to assess the integrity of the Surry Unit 1 beltline material SA-1526.

4.2 Chemistry Factor Methodology:

The calculation of chemistry factor (CF) values for the Surry Units 1 and 2 reactor vessel beltline materials is performed in accordance with Regulatory Guide 1.99, Revision 2 [1] as follows:

The CF is based on the Cu and Ni weight % of the material or it is based on the results of surveillance capsule test data. When the weight percent of copper and nickel is used to determine the CF, the CF is obtained from either Table 1 or Table 2 of Regulatory Guide 1.99, Revision 2 [1]. The results of this method are listed in Table 4-1.

When surveillance capsule data is used to determine the CF, the CF is determined as follows:

$$CF = \frac{\sum_{i=1}^{n}[A_i \times f_i^{(0.28-0.1\log f_i)}]}{\sum_{i=1}^{n}[f_i^{(0.28-0.1\log f_i)}]^2} \qquad (4\text{-}1)$$

Where:

	n	=	The Number of Surveillance Data Points
	A_i	=	The Measured Value of ΔRT_{NDT}
	f_i	=	Fluence for each Surveillance Data Point

The results of the CF determination for the Surry Units 1 and 2 surveillance data are listed in Table 4-3.

4.2.1 Application of the Regulatory Guide 1.99, Revision 2 Ratio Procedure for Beltline Material CF Determination

When credible surveillance data are used in the determination of the beltline material CF, correction for differences between the chemical compositions of surveillance weld specimens and the beltline material being evaluated is accomplished with the Regulatory Guide 1.99, Revision 2 [1], Position 2.1 ratio procedure. The ratio procedure is not applicable to the plate material. It is not necessary to correct for differences between the chemical compositions of surveillance specimens and the beltline material being evaluated when the chemical compositions are essentially equal. When there are significant differences between the surveillance and beltline material chemical composition, the Regulatory Guide 1.99 Revision 2 [1] Position 2.1 ratio procedure is applied in the determination of the beltline material chemistry factor.

Per Table 4-2, for the Unit 1 Intermediate to Lower Shell Circumferential Weld and the Unit 2 Lower Shell Longitudinal Weld, the copper and nickel concentrations of the surveillance weld metal is not the same as the corresponding beltline material. Therefore, the ratio procedure of Regulatory Guide 1.99, Revision 2[1], Position 2.1 is applied to this surveillance weld metal.

Similarly, for the Unit 1 Lower Shell Longitudinal Weld, the copper and nickel concentrations of the surveillance weld metal is not the same as the Unit 1 Lower Shell Longitudinal Weld SA-1526. Therefore the ratio procedure is applied to this surveillance weld metal, as well.

The copper and nickel weight concentrations of the Surry Unit 2 surveillance weld metal is the same for the Unit 2 Intermediate to Lower Shell Circumferential Weld R3008. Therefore, the ratio procedure of Regulatory Guide 1.99, Revision 2[1], Position 2.1 will not be applied to this surveillance weld metal (ie. Ratio = 1.0 for Unit 2 Intermediate to Lower Shell Circumferential Weld R3008).

4.2.2 Irradiation Temperature Effects on Surveillance Data for Determination of the Beltline Material CF

Application of the selected surveillance data to the beltline material requires normalization of measured transition temperature shift values to the mean irradiation temperature of the beltline material. A correction of 1.0°F is applied to measured values of transition temperature shift for each degree of irradiation temperature difference between surveillance specimens and beltline materials. This correction factor is documented in EPRI report NP-6114 [10] and ASTM report STP-1046 [11] and is cited in the November 12, 1997 NRC/industry meeting minutes [7].

When applying the selected surveillance data to the beltline material, the temperature correction is applied to each measured value of transition temperature shift for which the surveillance capsule irradiation temperature is higher than the beltline material irradiation temperature. The temperature correction is not performed when the surveillance capsule irradiation temperature is lower than the beltline material irradiation temperature. Correction for irradiation temperature is not necessary when the surveillance specimens are irradiated in the plant which is being evaluated.

Table 4-1: Available Surveillance Data for the Surry Units 1 and 2 Credibility Determination						
Materials	Capsule	Capsule Material	Fluence	ΔRT_{NDT}[a]	wt% Cu	wt% Ni
Plate Materials						
Lower Shell Forging (Unit 1) C4415-1	Surry Unit 1 Capsule T	Forging C4415-1	0.281	50	0.110	0.500
	Surry Unit 1 Capsule V	Forging C4415-1	1.940	113	0.110	0.500
	Surry Unit 1 Capsule X	Forging C4415-1	1.599	86	0.110	0.500
Intermediate Shell Forging C4339-1 (Unit 2)	Surry Unit 1 Capsule X	Forging C4339-1	.302	55	0.104	0.520
	Surry Unit 1 Capsule V	Forging C4339-1	1.88	75	0.104	0.520
Weld Materials						
Intermediate to Lower Shell Circumferential Weld SA-1585 (Unit 1) and Lower Shell Longitudinal Weld (Unit 2)	Pt Beach Unit 1 Capsule T	SA-1263	2.42	181	0.230	0.615
	Pt Beach Unit 1 Capsule R	SA-1263	2.38	181	0.230	0.615
	Pt Beach Unit 1 Capsule S	SA-1263	0.829	165	0.230	0.615
	Pt Beach Unit 1 Capsule V	SA-1263	0.502	107	0.230	0.615
	BWOG Capsule CR3-LG2	SA-1585	1.67	156	0.220	0.590
	BWOG Capsule CR3-LG1	SA-1585	0.51	141	0.220	0.590
Lower Shell Longitudinal Weld SA-1526 (Unit 1)	BWOG Capsule TMI2-LG1	SA-1526	0.83	216	0.370	0.700
	BWOG Capsule CR3-LG1	WF-25	0.799	205	0.380	0.700
	BWOG Capsule TMI2-LG1	WF-25	0.968	226	0.330	0.670
	TMI Unit 1 Capsule C	WF-25	0.866	166	0.330	0.670
	TMI Unit 1 Capsule E	WF-25	0.107	74	0.330	0.670
	Surry Unit 1 Capsule T	SA-1526	0.281	171	0.230	0.640
	Surry Unit 1 Capsule V	SA-1526	1.94	250	0.230	0.640
	Surry Unit 1 Capsule X	SA-1526	1.599	234	0.230	0.640
Intermediate to Lower Shell Circ. Weld R3008/0227 (Unit 2)	Surry Unit 2 Capsule X	R3008	.302	95	0.190	0.550
	Surry Unit 2 Capsule V	R3008	1.88	145	0.190	0.550

NOTES:
(a) Fluence values are in units of 10^{19} n/cm^2, E > 1.0 MeV.

Table 4-2: Reactor Vessel Beltline Material Copper and Nickel Content			
Surry Unit 1			
Material Description	wt. % Cu	wt. % Ni	CF [a]
Nozzle Shell Forging 122V109VA1	0.11	0.74	76.1
Intermediate Shell C4326-1	0.11	0.55	73.5
Intermediate Shell C4326-2	0.11	0.55	73.5
Lower Shell 4415-1	0.11	0.50	73.0
Lower Shell 4415-2	0.11	0.50	73.0
Nozzle to Intermediate Shell Circumferential Weld J726/25017	0.33	0.10	152.0
Intermediate to Lower Shell Circumferential Weld (ID 40%) SA-1585/72445	0.22	0.54	158.0
Intermediate to Lower Shell Circumferential Weld (ID 60%) SA-1650/72445	0.22	0.54	158.0
Intermediate Shell Longitudinal Welds L3 & L4 SA-1494/8T1554	0.16	0.57	143.9
Lower Shell Longitudinal Weld L1 SA-1494/8T1554	0.16	0.57	143.9
Lower Shell Longitudinal Weld L2 SA-1526/299L44	0.34	0.68	220.6
Surveillance Weld SA-1263 Pt. Beach Unit 1	0.23	0.615	171.6
Surveillance Weld SA-1526 Surry Unit 1	0.23	0.64	175.8

Notes:

(a) The chemistry factors shown in Table 4-2 are based on Position 1.1 using the best estimate chemistry for the beltline or surveillance material

Table 4-2 (cont'd): Reactor Vessel Material Copper and Nickel Content			
Surry Unit 2			
Material Description	wt. % Cu	wt. % Ni	CF [(a)]
Nozzle Shell Forging 123V303VA1	0.11	0.72	75.8
Intermediate Shell C4331-2	0.12	0.60	83.0
Intermediate Shell C4339-2	0.11	0.54	73.4
Lower Shell 4208-2	0.15	0.55	107.3
Lower Shell 4339-1	0.11	0.54	73.4
Nozzle to Intermediate Shell Circumferential Weld L737/4275	0.35	0.10	160.5
Intermediate to Lower Shell Circumferential Weld R3008/0227	0.19	0.55	149.3
Intermediate to Lower Shell Circumferential Weld L4 (ID 50%) WF-4/8T1762	0.19	0.57	152.4
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%) SA-1585/72445	0.22	0.54	158.0
Lower Shell Longitudinal Weld L2 (ID63%) and L1 (100%) WF-4/8T1762	0.19	0.57	152.4
Lower Shell Longitudinal Weld L2 (OD 37%) WF-8/8T1762	0.19	0.57	152.4
Surveillance Weld SA-1263 Pt. Beach Unit 1	0.23	0.615	171.6
Surveillance Weld R3008 Surry Unit 2	0.19	0.550	149.3

Notes:

(a) The chemistry factors shown in Table 4-2 are based on Position 1.1 using the best estimate chemistry for the beltline or surveillance material.

Table 4-3: Calculation of Chemistry Factors using Surry Units 1 and 2 Surveillance Capsule Data						
Materials	Capsule	Fluence(a)	FF	ΔRT_{NDT}(b)	FF * ΔRT_{NDT}	FF^2
Lower Shell Forging (Unit 1) C4415-1	Surry Unit 1 Capsule T	0.281	.6535	50	32.7	0.4271
	Surry Unit 1 Capsule V	1.940	1.1811	113	133.5	1.3951
	Surry Unit 1 Capsule X	1.599	1.1296	86	97.1	1.2760
				SUM:	263.3	3.10
	Chemistry Factor = 85.0(c)					
Intermediate Shell Forging C4339-1 (Unit 1)	Surry Unit 1 Capsule X	.302	.6720	55	37.0	0.4516
	Surry Unit 1 Capsule V	1.88	1.1729	75	88.0	1.3756
				SUM:	125.0	1.83
	Chemistry Factor = 68.4(c)					
Weld Materials						
Intermediate to Lower Shell Circumferential Weld SA-1585 (Unit 1) and Lower Shell Longitudinal Weld (Unit 2)	Pt Beach Unit 1 Capsule T	2.42	1.2380	181	224.1	1.5328
	Pt Beach Unit 1 Capsule R	2.38	1.2338	181	223.3	1.5224
	Pt Beach Unit 1 Capsule S	0.829	0.9474	165	156.3	0.8976
	Pt Beach Unit 1 Capsule V	0.502	0.8077	107	86.4	0.6523
				SUM:	690.2	4.61
	Chemistry Factor = 149.9(c)					
Lower Shell Longitudinal Weld SA-1526 (Unit 1)	Surry Unit 1 Capsule T	0.281	0.6535	171	111.7	0.4271
	Surry Unit 1 Capsule V	1.94	1.1811	250	295.3	1.3951
	Surry Unit 1 Capsule X	1.599	1.1296	234	264.3	1.2760
				SUM	671.4	3.10
	Chemistry Factor = 216.7(c)					
Intermediate to Lower Shell Circ. Weld R3008/0227 (Unit 2)	Surry Unit 2 Capsule X	.302	0.6720	95	63.8	0.4516
	Surry Unit 2 Capsule V	1.88	1.1729	145	170.1	1.3756
				SUM:	233.9	1.83
	Chemistry Factor = 128.0(c)					

Notes:

(a) Fluence values are in units of 10^{19} n/cm^2, E > 1.0 MeV.
(b) ΔRT_{NDT} are measured values.
(c) Chemistry Factor before application of the Regulatory Guide 1.99, Rev. 2[1] Position 2.1 ratio procedure.

4.3 Surveillance program credibility evaluation:

4.2.1 Credibility Criterion 1:

Criterion 1: *The materials in the surveillance capsules must be those which are controlling materials with regard to radiation embrittlement.*

The Surry Units 1 and 2 reactor vessels consists of the following beltline region materials:

a) Nozzle Shell Forging,
b) Intermediate Shell Forgings,
c) Lower Shell Forgings,
d) Intermediate to Lower Shell Circumferential Weld
e) Intermediate Shell Longitudinal Welds
f) Lower Shell Longitudinal Welds

Criterion 1 was applied at the time that the Surry surveillance program was designed and licensed. Therefore, the materials selected for use in the Surry Units 1 and 2 surveillance program are those judged to be most likely controlling with regard to radiation embrittlement according to the accepted methodology at the time the surveillance program was developed. Based on engineering judgment, the Surry Units 1 and 2 surveillance program meets the intent of this criteria.

The Surry Units 1 and 2 reactor vessel materials, from which the controlling materials are selected, are presented below.

<u>Weld Metal</u>:

The weld metal for Unit 1 includes the following:

Material	Heat Number	Flux Type	Flux Lot #
Nozzle to Intermediate Shell Circ. Weld	J726/25017	SAF 89	1197
Intermediate to Lower Shell Circ. Weld (ID 40%)	SA-1585/72445	Linde 80	8597
Intermediate to Lower Shell Circ. Weld (ID 60%)	SA-1650/72445	Linde 80	8632
Intermediate Shell Longitudinal Welds L3 & L4	SA-1494/8T1554	Linde 80	8579
Lower Shell Longitudinal Weld L1	SA-1494/8T1554	Linde 80	8579
Lower Shell Longitudinal Weld L2	SA-1526/299L44	Linde 80	8596
Pt. Beach Unit 1 Surveillance Weld	SA-1263/72445	Linde 80	8504
Surry Unit 1 Surveillance Weld	SA-1526/299L44	Linde 80	8596

The weld metal for Unit 2 includes the following:

Material	Heat Number	Flux Type	Flux Lot #
Nozzle to Intermediate Shell Circ. Weld	L737/4275	SAF 89	02275
Intermediate to Lower Shell Circ. Weld	R3008/0227	Grau Lo	LW320
Intermediate to Lower Shell Circ.l Weld L4 (ID 50%)	WF-4/8T1762	Linde 80	8597
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%)	SA-1585/72445	Linde 80	8597
Lower Shell Longitudinal Weld L2 (ID63%) and L1 (100%)	WF-4/8T1762	Linde 80	8632
Lower Shell Longitudinal Weld L2 (OD 37%)	WF-8/8T1762	Linde 80	8632
Pt. Beach Unit 1 Surveillance Weld	SA-1263/72445	Linde 80	8504
Surry Unit 2 Surveillance Weld	R3008/0227	Grau Lo	LW320

Forgings:

The forgings for Unit 1 include the following:

Material	Heat #	Forging Type
Nozzle Shell Forging	122V109VA1	SA508, Class 2
Intermediate Shell	C4326-1	SA533, Grade B1
Intermediate Shell	C4326-2	SA533, Grade B1
Lower Shell	4415-1	SA533, Grade B1
Lower Shell	4415-2	SA533, Grade B1

The forgings for Unit 2 include the following:

Material	Heat #	Forging Type
Nozzle Shell Forging	123V303VA1	SA508, Class 2
Intermediate Shell	C4331-2	SA533, Grade B1
Intermediate Shell	C4339-2	SA533, Grade B1
Lower Shell	4208-2	SA533, Grade B1
Lower Shell	4339-1	SA533, Grade B1

4.2.2 Credibility Criterion 2;

Criterion 2: *Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions must be small enough to permit the determination of the 30 ft-lb temperature unambiguously.*

Virginia Power reviewed the surveillance capsule analysis reports which support the chemistry factor calculations and determined that this criterion is met.

4.2.3 Credibility Criterion 3:

Criterion 3: *Where there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values must be less than 28°F for welds and 17°F for base metal. Even if the range in the capsule fluences is large (two or more orders of magnitude), the scatter may not exceed twice those values.*

The least squares method, as described in Regulatory Guide 1.99 Position 2.1, Revision 2 [1], as clarified in the November 12, 1997 meeting minutes [7], is utilized in determining a best-fit line for this data to determine if this criteria is met.

Table 4-4: Surry Units 1 and 2 Surveillance Capsule Data Scatter about the Best Fit Line for the Shell Forging Material				
Material	FF[b]	Measured ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT}[a] (°F)	Scatter of ΔRT_{NDT} (°F)
Lower Shell Forging	.6535	50	55.54	-6
C-4415-1 (Unit 1)	1.1811	113	100.38	13
CF = 85.0°F	1.1296	86	96.00	-10
Intermediate Shell	.6720	55	45.94	9
C-4339-1 (Unit 2)	1.1729	75	80.19	-5
CF = 68.4°F				

NOTES:

(a) The Chemistry Factor used for the best fit ΔRT_{NDT} is calculated in accordance with Regulatory Guide 1.99, Rev. 2, Position 2.1.

(b) Inputs to Fluence Factor calculations are in units of 10^{19} n/cm^2, E > 1.0 MeV.

(c) The scatter of ΔRT_{NDT} values about a best-fit line drawn, as described in Regulatory Guide 1.99, Rev. 2, Position 2.1, should be less than 17°F for base metal. As shown above, the scatter of all data points is less than 17°F. Therefore, credibility criterion 3 is met. Since this surveillance data is credible, a σ_{Δ} margin of 8.5°F is used when predicting the Surry Units 1 and 2 beltline plate material properties for these materials.

(d) For forgings, the credibility determination requires normalization of measured transition temperature shift values to the mean irradiation temperature of surveillance specimens. A correction of 1.0°F is applied to measured values of transition temperature shift for each degree of irradiation temperature difference between surveillance specimens and beltline materials. This correction factor is documented in EPRI report NP-6114 [10] and ASTM report STP-1046 [11] and is cited in the November 12, 1997 NRC/Industry meeting minutes[7]. A temperature correction is not applied to measured values of transition temperature shift for the credibility determination if applicable surveillance data are irradiated in a single reactor (i.e., are irradiated at a similar temperature). For the credibility determination, a chemistry correction is not applied to measured values of transition temperature shift for the credibility determination if applicable surveillance data are obtained from a single source (i.e., are machined from the same block of material).

Table 4-5: Surry Units 1 and 2 Surveillance Capsule Data Scatter about the Best Fit Line for the Weld Material				
Material	FF[(b)]	Measured ΔRT_{NDT} (30 ft-lb) (°F)	Best Fit ΔRT_{NDT}[(a)] (°F)	Scatter of ΔRT_{NDT} (°F)
Interm. To Lower Shell	1.2380	181	185.55	-5
Circumferential Weld	1.2338	181	184.92	-4
SA-1585/1650 (Units 1&2)	0.9474	165	141.99	23
CF = 149.9°F	0.8077	107	121.05	-14
Intermediate to Lower Shell	0.6720	95	86.02	9
Circumferential Weld R3008	1.1729	145	150.14	-5
(Unit 2) CF = 128.0 °F				
Lower Shell	0.6535	171	141.61	29
Longitudinal Weld	1.1811	250	255.95	-6
SA-1526	1.1296	234	244.78	-11
(Unit 1) CF = 216.7 °F				

NOTES:

(a) The Chemistry Factor used for the best fit ΔRT_{NDT} is calculated in accordance with Regulatory Guide 1.99, Rev. 2, Position 2.1, using measured values of transition temperature shift that are normalized to the mean chemical composition of the surveillance materials by application of the Position 2.1 ratio procedure.

(b) Inputs to Fluence Factor calculations are in units of 10^{19} n/cm^2, E > 1.0 MeV.

(c) The scatter of ΔRT_{NDT} values about a best-fit line drawn, as described in Regulatory Guide 1.99, Rev. 2, Position 2.1, is less than 28°F for the Intermediate to Lower Shell Circumferential Welds SA-1585 and R3008. Therefore, credibility criterion 3 is met for these two surveillance weld materials. Since this surveillance weld data is credible, a σ_Δ margin of 14°F is used when predicting the Surry Units 1 and 2 beltline weld material properties for these materials. The scatter of ΔRT_{NDT} values about a best-fit line for the Unit 1 Lower Shell Longitudinal Weld SA-1526 exceeds 28 °F. Other combinations of surveillance data applicable to SA-1526 on Table 4-1 have also been determined to be non-credible. Therefore, for the Unit 1 Lower Shell Longitudinal Weld SA-1526, a σ_Δ margin of 28°F is used when predicting the weld material properties of this material. The Position 1.1 Chemistry Factor, based on the beltline material chemical composition, is determined to be conservative in accordance with the guidelines presented in the November 12, 1997 meeting minutes [7].

(d) The credibility determination requires normalization of measured transition temperature shift values to the mean irradiation temperature of surveillance specimens. A correction of 1.0°F is applied to measured values of transition temperature shift for each degree of irradiation temperature difference between surveillance specimens and beltline materials. This correction factor is documented in EPRI report NP-6114 [10] and ASTM report STP-1046 [11] and is cited in the November 12, 1997 NRC/industry meeting minutes[7]. A temperature correction is not applied to measured values of transition temperature shift for the credibility determination if applicable surveillance data are irradiated in a single reactor (i.e., are irradiated at a similar temperature). For the credibility determination, a chemistry correction is not applied to measured values of transition temperature shift if applicable surveillance data are obtained from a single source (i.e., machined from the same block of material).

4.2.4 Credibility Criterion 4:

Criterion 4: *The irradiation temperature of the Charpy specimens in the capsule must equal the vessel wall temperature at the cladding/base metal interface within +/- 25°F.*

The surveillance capsule analysis reports which support the chemistry factor calculations demonstrate that this criterion is met.

4.2.5 Credibility Criterion 5:

Criterion 5: *The surveillance data for the correlation monitor material in the capsule, if present, must fall within the scatter band of the data base for the material.*

The surveillance capsule analysis reports which support the chemistry factor calculations demonstrate that this criterion is met.

4.2.6 Results and Conclusions of the Credibility Evaluation

Results of the credibility evaluation indicate that, except for the Surry Unit 1 weld material SA-1526, the surveillance materials meet all of the credibility criteria discussed above. For weld material SA-1526, the scatter of measured ΔRT_{NDT} values about the best fit ΔRT_{NDT} trend line exceeds 28°F. Thus, the SA-1526 surveillance data are not credible. The Regulatory Guide 1.99, Revision 2 [1], Position 1.1 CF for SA-1526 based on the beltline material chemistry composition is evaluated and determined to be conservative in accordance with the guidance presented in the November 12, 1997 meeting minutes [7]. Therefore, the Position 1.1 CF is used to evaluate the material condition of the SA-1526 beltline material.

5. UNIRRADIATED PROPERTIES

5.1 *Initial RT_{NDT} of Beltline Materials*

Table 5-1 below contains a description of the beltline materials and their initial RT_{NDT} values.

5.2 *Determination of σ_I:*

For initial RT_{NDT} values not measured, the standard deviations for initial RT_{NDT} values for Surry Units 1 and 2 beltline materials are determined by statistical assessment of available measured values (i.e., set equal to the standard deviation of the estimate of the unirradiated RT_{NDT}). A σ_I value of 0°F is applied to measured values of initial RT_{NDT}.

5.3 *Bolt-up Temperature:*

The reactor vessel may be bolted up at temperatures greater than the initial RT_{NDT} of the material stressed by the boltup (i.e., the vessel flange). The most limiting initial RT_{NDT} value is 10°F on the vessel flange. However, a minimum RCS temperature limit of 60°F is imposed to ensure that the RCS temperatures are sufficiently high to prevent damage to the closure head/ vessel flange during the removal or installation of the reactor vessel head bolts.

Table 5-1: Reactor Vessel Beltline Material Initial RT_{NDT} Values		
Material Description	Heat #	Initial RT_{NDT}[(a)]
Surry Unit 1		
Nozzle Shell Forging	122V109VA1	40
Intermediate Shell	C4326-1	10
Intermediate Shell	C4326-2	0
Lower Shell	4415-1	20
Lower Shell	4415-2	0
Nozzle to Intermediate Shell Circumferential Weld	J726/25017	0
Intermediate to Lower Shell Circumferential Weld (ID 40%)	SA-1585/72445	-5
Intermediate to Lower Shell Circumferential Weld (ID 60%)	SA-1650/72445	-5
Intermediate Shell Longitudinal Welds L3 & L4	SA-1494/8T1554	-5
Lower Shell Longitudinal Weld L1	SA-1494/8T1554	-5
Lower Shell Longitudinal Weld L2	SA-1526/299L44	-7
Closure Head Flange	FV-1894	10
Vessel Flange	FV-1870	10

Table 5-1 (cont'd): Reactor Vessel Beltline Material Description		
Material Description	**Heat #**	**Initial RT_{NDT}[a]**
Surry Unit 2		
Nozzle Shell Forging	123V303VA1	30
Intermediate Shell	C4331-2	-10
Intermediate Shell	C4339-2	-20
Lower Shell	4208-2	-30
Lower Shell	4339-1	-10
Nozzle to Intermediate Shell Circumferential Weld	L737/4275	0
Intermediate to Lower Shell Circumferential Weld	R3008/0227	0
Intermediate to Lower Shell Circumferential Weld L4 (ID 50%)	WF-4/8T1762	-5
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%)	SA-1585/72445	-5
Lower Shell Longitudinal Weld L2 (ID63%) and L1 (100%)	WF-4/8T1762	-5
Lower Shell Longitudinal Weld L2 (OD 37%)	WF-8/8T1762	-5
Closure Head Flange	ZV-3475	<10
Vessel Flange	ZV-3476	-65

6. REACTOR VESSEL GEOMETRIC & SYSTEM PARAMETERS

The applicable reactor vessel physical dimensions and operating conditions, along with other system parameters, are shown in Table 6-1.

Table 6-1: RV Physical Dimensions and Operating Conditions

Parameter	Value
Vessel Beltline Thickness	8.08 inches
Vessel Inner Radius to Clad	78.95 inches
Vessel Clad Thickness	0.16 inches
Pre-service System Hydrostatic Pressure	3107 psig
Capacity Factor (future cycles)	90%
System and Component Operating Conditions/Dimensions	Design Press. - 2485 psig
	Oper. Press. - 2235 psig

7. FLUENCE FACTOR DETERMINATION

7.1 Peak Clad Base Metal Interface Fluence for each Beltline Material:

Tables 7-8 through 7-15 present the best estimate peak fluences for the various materials in the Surry Units 1 and 2 reactor vessels. The cumulative core burnup values (EFPY) at which the pressure/temperature limit curves are calculated are:

Unit 1:

Current EOL = 29.6 EFPY
Renewal EOL = 47.6 EFPY

Unit 2:

Current EOL = 30.1 EFPY
Renewal EOL = 48.1 EFPY

Table 7-1: Best-Estimate Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 1 Reactor Vessel Intermediate and Lower Shell Plate Materials	
Unit 1 EFPY	Unit 1 Fluence
1.1	.183
1.6	.290
2.3	.408
3.4	.583
4.5	.757
5.9	.950
6.8	1.16
8.0	1.32
9.3	1.44
10.6	1.56
11.7	1.68
13.3	1.83
14.6	1.96
29.6	3.53
47.6	5.40

Table 7-2: Best Estimate Peak Fluence (10^{19} n/cm2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 1 Nozzle to Intermediate Shell Circumferential Weld	
Unit 1 EFPY	Unit 1 Fluence
1.1	.00999
1.6	.0163
2.3	.0252
3.4	.0347
4.5	.0462
5.9	.0601
6.8	.0746
8.0	.0868
9.3	.0976
10.6	.108
11.7	.120
13.3	.135
14.6	.149
29.6	.307
47.6	.496

Table 7-3: Best-Estimate Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 1 Upper Shell Longitudinal Welds	
Unit 1 EFPY	Unit 1 Fluence
1.1	.0308
1.6	.0495
2.3	.0697
3.4	.100
4.5	.126
5.9	.156
6.8	.176
8.0	.201
9.3	.229
10.6	.256
11.7	.280
13.3	.315
14.6	.337
29.6	.599
47.6	.913

Table 7-4: Best-Estimate Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 1 Reactor Vessel Beltine Intermediate to Lower Shell Circumferential Weld	
Unit 1 EFPY	Unit 1 Fluence
1.1	.182
1.6	.294
2.3	.412
3.4	.590
4.5	.766
5.9	.959
6.8	1.18
8.0	1.33
9.3	1.45
10.6	1.57
11.7	1.68
13.3	1.83
14.6	1.94
29.6	3.20
47.6	4.70

Table 7-5: Best-Estimate Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 1 Reactor Vessel Lower Shell Longitudinal Welds	
Unit 1 EFPY	Unit 1 Fluence
1.1	.00307
1.6	.00502
2.3	.00703
3.4	.101
4.5	.128
5.9	.158
6.8	.178
8.0	.204
9.3	.230
10.6	.257
11.7	.281
13.3	.314
14.6	.332
29.6	.540
47.6	.790

Table 7-6: Best-Estimate Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 2 Reactor Vessel Plate and Intermediate to Lower Shell Circumferential Weld Material	
Unit 2 EFPY	**Unit 2 Fluence**
1.2	.202
1.9	.329
2.6	.450
3.8	.656
4.9	.828
6.2	1.02
7.4	1.23
8.4	1.34
9.7	1.47
10.9	1.60
12.4	1.74
13.9	1.89
30.1	3.52
48.1	5.34

Table 7-7: Best Estimate Peak Fluence (10^{19} n/cm2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 2 Nozzle to Intermediate Shell Circumferential Weld	
Unit 2 EFPY	Unit 2 Fluence
1.2	.0121
1.9	.0197
2.6	.0289
3.8	.0416
4.9	.0539
6.2	.0677
7.4	.0824
8.4	.0909
9.7	.104
10.9	.115
12.4	.128
13.9	.142
30.1	.298
48.1	.471

Table 7-8: Best-Estimate Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) at the Pressure Vessel Base Metal/Clad Interface of the Surry Unit 2 Upper and Lower Shell Longitudinal Welds

Unit 2 EFPY	Unit 2 Fluence
1.2	.0339
1.9	.0572
2.6	.0821
3.8	.116
4.9	.146
6.2	.180
7.4	.211
8.4	.238
9.7	.266
10.9	.292
12.4	.323
13.9	.354
30.1	.697
48.1	1.08

7.2 Fluence Methodology Used:

The methodology used for determining the reactor vesesl neutron fluences is that which is documented in the Virginia Power Topical Report VEP-NAF-3, "Reactor Vessel Fluence Analysis Methodology" [8]. This topical report has been submitted to the NRC for review [9], but has not been approved as of this writing.

7.3 Uncertainty in Fluence Evaluation:

The fluence estimates have been demonstrated to be within 20% of the true value.

7.4 Fluence Values for Beltline Materials at Locations Other than the Inner Surface

The neutron fluence at any depth in the vessel wall is calculated as follows:

$$f = f_{surf} \cdot e^{-0.24\,(x)}\text{, } 10^{19}\ \text{n/cm}^2\ (E > 1.0\ \text{MeV}) \qquad (7\text{-}1)$$

where: f_{surf} = Vessel inner wall surface fluence, 10^{19} n/cm^2 (E > 1.0 MeV)
x = is the depth into the vessel wall from the inner surface, inches

7.5 Fluence Factor and how it is Determined:

Fluence factors at the 1/4T and 3/4T locations are determined using the method described in Regulatory Guide 1.99, Revision 2[1], as follows:

$$FF = \text{fluence factor} = f^{(0.28 - 0.1 \log (f))} \qquad (7\text{-}2)$$

where: f = Vessel inner wall surface fluence, 1/4T fluence or 3/4T fluence, 10^{19} n/cm^2 (E > 1.0 MeV)

8. CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

8.1 Methodology:

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:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \tag{8-1}$$

8.1.1 Initial RT_{NDT}

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[4]. 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. Initial RT_{NDT} values are documented in Table 5-1.

8.1.2 ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as the product of the chemistry factor, CF, and the fluence factor determined per Equation 7-2, as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10\log f)} \tag{8-2}$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the attenuated fluence at the specific depth must be determined based on Equation 7-1. The resultant fluence is then placed in Equation 8-2 in order to calculate the ΔRT_{NDT} at the specific depth.

8.1.3 Margin used in Adjusted Reference Temperature Calculation:

The margin term used in determining the adjusted RT_{NDT} is calculated using the margin term equation from Regulatory Guide 1.99, Revision 2[1], Position 2.1 as follows:

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{8-3}$$

where each of the terms is described below.

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ:

σ_Δ is the standard deviation of the estimate of the shift in the initial RT_{NDT} determined in accordance with Regulatory Guide 1.99, Revision 2 [1]. Specific values of σ_Δ are as follows:

For plates and forgings:

σ_Δ = 17 when surveillance capsule data is not used
σ_Δ = 8.5 when credible surveillance capsule data is used

For welds:

σ_Δ = 28 when surveillance capsule data is not used
σ_Δ = 14 when credible surveillance capsule data is used

NOTE: σ_Δ need not exceed $0.5^*\Delta RT_{NDT}$ per Regulatory Guide 1.99, Revision 2.

Standard Deviation for Initial RT_{NDT} Margin Term, σ_i:

When a heat-specific measured value of the initial RT_{NDT} is not available, then σ_I is the standard deviation of the initial RT_{NDT} determined in accordance with Regulatory Guide 1.99, Revision 2[1]. When a heat-specific measured value of the initial RT_{NDT} is available, σ_I is assumed to be zero. When σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} is available, the total margin term, based on Equation 4 of Regulatory Guide 1.99 Rev. 2, is as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

8.1.2 Summary of the Margin Terms and Adjusted Reference Temperature Calculations:

Using the methodology described above, the initial RT_{NDT}, ΔRT_{NDT} and margins used in the ART calculations for each of the Surry Units 1 and 2 reactor vessel materials are shown in Tables 8-1 through 8-4. Tables 8-1 and 8-2 are applicable to the end of license. Tables 8-3 and 8-4 are applicable to the end of license renewal period.

Table 8-1: Calculation of the ART Values for the 1/4T Location at End of License							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 1 (29.6 EFPY)							
Nozzle Shell Forging 122V109VA1	Position 1.1	76.1	0.187	42.1	34.0	40.0	116.1
Intermediate Shell C4326-1	Position 1.1	73.5	2.154	88.8	34.0	10.0	132.8
Intermediate Shell C4326-2	Position 1.1	73.5	2154	88.8	34.0	0	122.8
Lower Shell 4415-1	Position 2.1	85.0	2154	102.7	17.0	20.0	139.7
Lower Shell 4415-2	Position 1.1	73.0	2.154	88.2	34.0	0	122.2
Nozzle to Intermediate Shell Circumferential Weld J726/25017	Position 1.1	152.0	0.187	84.2	68.8	0	153.0
Intermediate to Lower Shell Circumferential Weld (ID 40%) SA-1585/72445	Position 2.1	138.0	1.952	163.3	48.3	-5	206.6
Intermediate to Lower Shell Circumferential Weld (ID 60%) SA-1650/72445	Position 2.1	138.0	1.952	163.3	48.3	-5	206.6
Intermediate Shell Longitudinal Welds L3 & L4 SA-1494/8T1554	Position 1.1	143.9	0.366	103.9	68.5	-5	167.4
Lower Shell Longitudinal Weld L1 SA-1494/8T1554	Position 1.1	143.9	0.329	100.0	68.5	-5	163.4
Lower Shell Longitudinal Weld L2 SA-1526/299L44	Position 1.1	220.6	0.329	153.2	69.5	-7	215.7

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-1 (cont'd): Calculation of the ART Values for the 1/4T Location at End of License							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 2 (30.1 EFPY)							
Nozzle Shell Forging 123V303VA1	Position 1.1	75.8	0.182	41.5	34.0	30	105.5
Intermediate Shell C4331-2	Position 1.1	83.0	2.147	100.2	34.0	-10	124.2
Intermediate Shell C4339-2	Position 1.1	73.4	2.147	88.6	34.0	-20	102.6
Lower Shell 4208-2	Position 1.1	107.3	2.147	129.5	34.0	-30	133.5
Lower Shell 4339-1	Position 2.1	68.4	2.147	82.6	17.0	-10	89.6
Nozzle to Intermediate Shell Circumferential Weld L737/4275	Position 1.1	160.5	0.182	87.8	68.8	0	156.6
Intermediate to Lower Shell Circumferential Weld R3008/0227	Position 2.1	128.0	2.147	154.6	48.8	0	203.4
Intermediate to Lower Shell Circumferential Weld L4 (ID 50%) WF-4/8T1762	Position 1.1	152.4	0.425	116.2	68.5	-5	179.6
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%) SA-1585/72445	Position 2.1	138.0	0.425	105.2	48.3	-5	148.6
Lower Shell Longitudinal Weld L2 (ID63%) and L1 (100%) WF-4/8T1762	Position 1.1	152.4	0.425	116.2	68.5	-5	179.6
Lower Shell Longitudinal Weld L2 (OD 37%) WF-8/8T1762	Position 1.1	152.4	0.425	116.2	68.5	-5	179.6

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-2: Calculation of the ART Values for the 3/4T Location at End of License							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 1 (29.6 EFPY)							
Nozzle Shell Forging 122V109VA1	Position 1.1	76.1	0.070	26.5	34.0	40.0	100.5
Intermediate Shell C4326-1	Position 1.1	73.5	0.802	68.9	34.0	10.0	112.9
Intermediate Shell C4326-2	Position 1.1	73.5	0.802	68.9	34.0	0	102.9
Lower Shell 4415-1	Position 2.1	85.0	0.802	79.7	17.0	20.0	116.7
Lower Shell 4415-2	Position 1.1	73.0	0.802	68.5	34.0	0	102.5
Nozzle to Intermediate Shell Circumferential Weld J726/25017	Position 1.1	152.0	0.070	53.0	68.8	0	121.8
Intermediate to Lower Shell Circumferential Weld (ID 40%) SA-1585/72445	Position 2.1	138.0	0.727	125.7	48.3	-5	169.0
Intermediate to Lower Shell Circumferential Weld (ID 60%) SA-1650/72445	Position 2.1	138.0	0.727	125.7	48.3	-5	169.0
Intermediate Shell Longitudinal Welds L3 & L4 SA-1494/8T1554	Position 1.1	143.9	0.136	69.3	68.5	-5	132.8
Lower Shell Longitudinal Weld L1 SA-1494/8T1554	Position 1.1	143.9	0.123	66.0	68.5	-5	129.5
Lower Shell Longitudinal Weld L2 SA-1526/299L44	Position 1.1	220.6	0.123	101.2	69.5	-7	163.8

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-2 (cont'd): Calculation of the ART Values for the 3/4T Location at End of License							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 2 (30.1 EFPY)							
Nozzle Shell Forging 123V303VA1	Position 1.1	75.8	0.068	26.0	34.0	30	90.0
Intermediate Shell C4331-2	Position 1.1	83.0	0.799	77.8	34.0	-10	101.8
Intermediate Shell C4339-2	Position 1.1	73.4	0.799	68.8	34.0	-20	82.8
Lower Shell 4208-2	Position 1.1	107.3	0.799	100.5	34.0	-30	104.5
Lower Shell 4339-1	Position 2.1	68.4	0.799	64.1	17.0	-10	71.1
Nozzle to Intermediate Shell Circumferential Weld L737/4275	Position 1.1	160.5	0.068	55.1	68.8	0	123.9
Intermediate to Lower Shell Circumferential Weld R3008/0227	Position 2.1	128.0	0.799	120.0	48.8	0	168.8
Intermediate to Lower Shell Circumferential Weld L4 (ID 50%) WF-4/8T1762	Position 1.1	152.4	0.158	78.4	68.5	-5	141.9
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%) SA-1585/72445	Position 2.1	138.0	0.158	71.1	48.3	-5	114.4
Lower Shell Longitudinal Weld L2 (ID63%) and L1 (100%) WF-4/8T1762	Position 1.1	152.4	0.158	78.4	68.5	-5	141.9
Lower Shell Longitudinal Weld L2 (OD 37%) WF-8/8T1762	Position 1.1	152.4	0.158	78.4	68.5	-5	141.9

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-3: Calculation of the ART Values for the 1/4T Location at End of License Renewal							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 1 (47.6 EFPY)							
Nozzle Shell Forging 122V109VA1	Position 1.1	76.1	0.303	51.2	34.0	40.0	125.2
Intermediate Shell C4326-1	Position 1.1	73.5	3.294	96.5	34.0	10.0	140.5
Intermediate Shell C4326-2	Position 1.1	73.5	3.294	96.5	34.0	0	130.5
Lower Shell 4415-1	Position 2.1	85.0	3.294	111.6	17.0	20.0	148.6
Lower Shell 4415-2	Position 1.1	73.0	3.294	95.8	34.0	0	129.8
Nozzle to Intermediate Shell Circumferential Weld J726/25017	Position 1.1	152.0	0.303	102.2	68.8	0	171.0
Intermediate to Lower Shell Circumferential Weld (ID 40%) SA-1585/72445	Position 2.1	138.0	2.867	176.7	48.3	-5	220.0
Intermediate to Lower Shell Circumferential Weld (ID 60%) SA-1650/72445	Position 2.1	138.0	2.867	176.7	48.3	-5	220.0
Intermediate Shell Longitudinal Welds L3 & L4 SA-1494/8T1554	Position 1.1	143.9	0.558	120.4	68.5	-5	183.9
Lower Shell Longitudinal Weld L1 SA-1494/8T1554	Position 1.1	143.9	0.482	114.6	68.5	-5	178.1
Lower Shell Longitudinal Weld L2 SA-1526/299L44	Position 1.1	220.6	0.482	175.7	69.5	-7	238.2

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-3 (cont'd): Calculation of the ART Values for the 1/4T Location at End of License Renewal							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 2 (48.1 EFPY)							
Nozzle Shell Forging 123V303VA1	Position 1.1	75.8	0.287	50.0	34.0	30	114.0
Intermediate Shell C4331-2	Position 1.1	83.0	3.258	108.7	34.0	-10	132.7
Intermediate Shell C4339-2	Position 1.1	73.4	3.258	96.2	34.0	-20	110.2
Lower Shell 4208-2	Position 1.1	107.3	3.258	140.5	34.0	-30	144.5
Lower Shell 4339-1	Position 2.1	68.4	3.258	89.6	17.0	-10	96.6
Nozzle to Intermediate Shell Circumferential Weld L737/4275	Position 1.1	160.5	0.287	105.8	68.8	0	174.6
Intermediate to Lower Shell Circumferential Weld R3008/0227	Position 2.1	128.0	3.258	167.7	48.8	0	216.5
Intermediate to Lower Shell Circumferential Weld L4 (ID 50%) WF-4/8T1762	Position 1.1	152.4	0.659	134.5	68.5	-5	198.0
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%) SA-1585/72445	Position 2.1	138.0	0.659	121.9	48.3	-5	165.2
Lower Shell Longitudinal Weld L2 (ID63%) and L1 (100%) WF-4/8T1762	Position 1.1	152.4	0.659	134.5	68.5	-5	198.0
Lower Shell Longitudinal Weld L2 (OD 37%) WF-8/8T1762	Position 1.1	152.4	0.659	134.5	68.5	-5	198.0

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-4: Calculation of the ART Values for the 3/4T Location at End of License Renewal							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 1 (47.6 EFPY)							
Nozzle Shell Forging 122V109VA1	Position 1.1	76.1	0.113	33.6	34.0	40.0	107.6
Intermediate Shell C4326-1	Position 1.1	73.5	1.226	77.7	34.0	10.0	121.7
Intermediate Shell C4326-2	Position 1.1	73.5	1.226	77.7	34.0	0	111.7
Lower Shell 4415-1	Position 2.1	85.0	1.226	89.8	17.0	20.0	126.8
Lower Shell 4415-2	Position 1.1	73.0	1.226	77.1	34.0	0	111.1
Nozzle to Intermediate Shell Circumferential Weld J726/25017	Position 1.1	152.0	0.113	67.0	68.8	0	135.9
Intermediate to Lower Shell Circumferential Weld (ID 40%) SA-1585/72445	Position 2.1	138.0	1.067	140.5	48.3	-5	183.9
Intermediate to Lower Shell Circumferential Weld (ID 60%) SA-1650/72445	Position 2.1	138.0	1.067	140.5	48.3	-5	183.9
Intermediate Shell Long. Welds L3 & L4 SA-1494/8T1554	Position 1.1	143.9	0.208	83.2	68.5	-5	146.7
Lower Shell Longitudinal Weld L1 SA-1494/8T1554	Position 1.1	143.9	0.179	78.2	68.5	-5	141.7
Lower Shell Longitudinal Weld L2 SA-1526/299L44	Position 1.1	220.6	0.179	119.9	69.5	-7	182.5

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-4 (cont'd): Calculation of the ART Values for the 3/4T Location at End of License Renewal							
Material	RG 1.99 R2 Method	CF	Fluence (a)	ΔRT_{NDT}	Margin	IRT_{NDT}	ART
Surry Unit 2 (48.1 EFPY)							
Nozzle Shell Forging 123V303VA1	Position 1.1	75.8	0.107	32.6	34.0	30	96.6
Intermediate Shell C4331-2	Position 1.1	83.0	1.213	87.5	34.0	-10	111.5
Intermediate Shell C4339-2	Position 1.1	73.4	1.213	77.3	34.0	-20	91.3
Lower Shell 4208-2	Position 1.1	107.3	1.213	113.0	34.0	-30	117.0
Lower Shell 4339-1	Position 2.1	68.4	1.213	72.0	17.0	-10	79.0
Nozzle to Intermediate Shell Circumferential Weld L737/4275	Position 1.1	160.5	0.107	69.1	68.8	0	137.9
Intermediate to Lower Shell Circumferential Weld R3008/0227	Position 2.1	128.0	1.213	134.9	48.8	0	183.7
Intermediate to Lower Shell Circumferential Weld L4 (ID 50%) WF-4/8T1762	Position 1.1	152.4	0.245	94.3	68.5	-5	157.8
Intermediate Shell Longitudinal Welds L3 (100%) & L4 (50%) SA-1585/72445	Position 2.1	138.0	0.245	85.5	48.3	-5	128.8
Lower Shell Longitudinal Weld L2 (ID 63%) and L1 (100%) WF-4/8T1762	Position 1.1	152.4	0.245	94.3	68.5	-5	157.8
Lower Shell Longitudinal Weld L2 (OD 37%) WF-8/8T1762	Position 1.1	152.4	0.245	94.3	68.5	-5	157.8

Notes:
(a) Fluences in units of (10^{19} n/cm^2, E > 1.0 MeV)

Table 8-5: Summary of Limiting ART Values to be Used in the Generation of the Surry Units 1 and 2 Reactor Vessel Heatup and Cooldown Curves *		
EFPY	1/4 T Limiting ART	3/4 Limiting ART
29.6 (Unit 1)	215.7	169.0
30.1 (Unit 2)	203.4	168.8
47.6 (Unit 1)	238.2	183.9
48.1 (Unit 2)	216.5	183.7

* The Unit 1 Lower Shell Longitudinal Weld L2, SA-1526/299L44 is the limiting material at the 1/4T location. The Intermediate to Lower Shell Circumferential Welds, SA-1585/72445 and SA-1650/72445, in Surry Unit 1 are the limiting materials at the 3/4T location.

9. HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

9.1 Introduction and Methodology:

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 8 of this report. Figures 9-1 to 9-8 define all of the above limits for ensuring prevention of nonductile failure for the WOG Reactor Vessel 60-Year Evaluation Minigroup reactor vessel. The pressure temperature limit data points are tabulated in Tables 9-1 to 9-4.

Figures 9-1 to 9-3 present the end of license heatup curves with heatup rates of 20, 40 and 60°F/hour (a heatup rate of 0°F/hour is defined by the steady state cooldown curve) and with margins of 0°F and 0 psi for possible instrumentation errors. Figure 9-4 presents the end of license cooldown curves with cooldown rates of 0, 20, 40, 60 and 100°F/hour and margins of 0°F and 0 psi for possible instrumentation errors. The data points for the end of license heatup and cooldown pressure temperature limits are presented in Tables 9-1 and 9-2.

Figures 9-5 to 9-7 present the end of license renewal heatup curves with heatup rates of 20, 40 and 60°F/hour (a heatup rate of 0°F/hour is defined by the steady state cooldown curve) and with margins of 0°F and 0 psi for possible instrumentation errors. Figure 9-8 presents the end of license renewal cooldown curves with cooldown rates of 0, 20, 40, 60 and 100°F/hour and margins of 0°F and 0 psi for possible instrumentation errors. The data points for the end of license renewal heatup and cooldown pressure temperature limits are presented in Tables 9-3 and 9-4.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 9-1 to 9-3 and Figures 9-5 to 9-7 (for the specific heatup rate and licensing period being utilized). 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 10 CFR Part 50. The governing equations for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code[3] as follows:

End of License (EOL) $1.5\ K_{Im} < K_{Ia}$ (9-1)

End of License Renewal (EOLR) $1.5\ K_{Im} < K_{Ic}$ (9-2)

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress

$K_{Ia} = 26.78 + 1.233\ e^{[0.0145\ (T - RT_{NDT} + 160)]}$

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\ (T - RT_{NDT})]}$

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 2. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

9.2 Instrumentation Error Margins (if they are to be applied and how they are determined):

TERR =	Temperature instrumentation error	=	0
PERR =	Pressure instrumentation error	=	0
PDELTA =	Pressure difference between gage and beltline region	=	0

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)
LIMITING ART VALUES AT EOL: ¼T, 215.7°F
¾T, 169.0°F

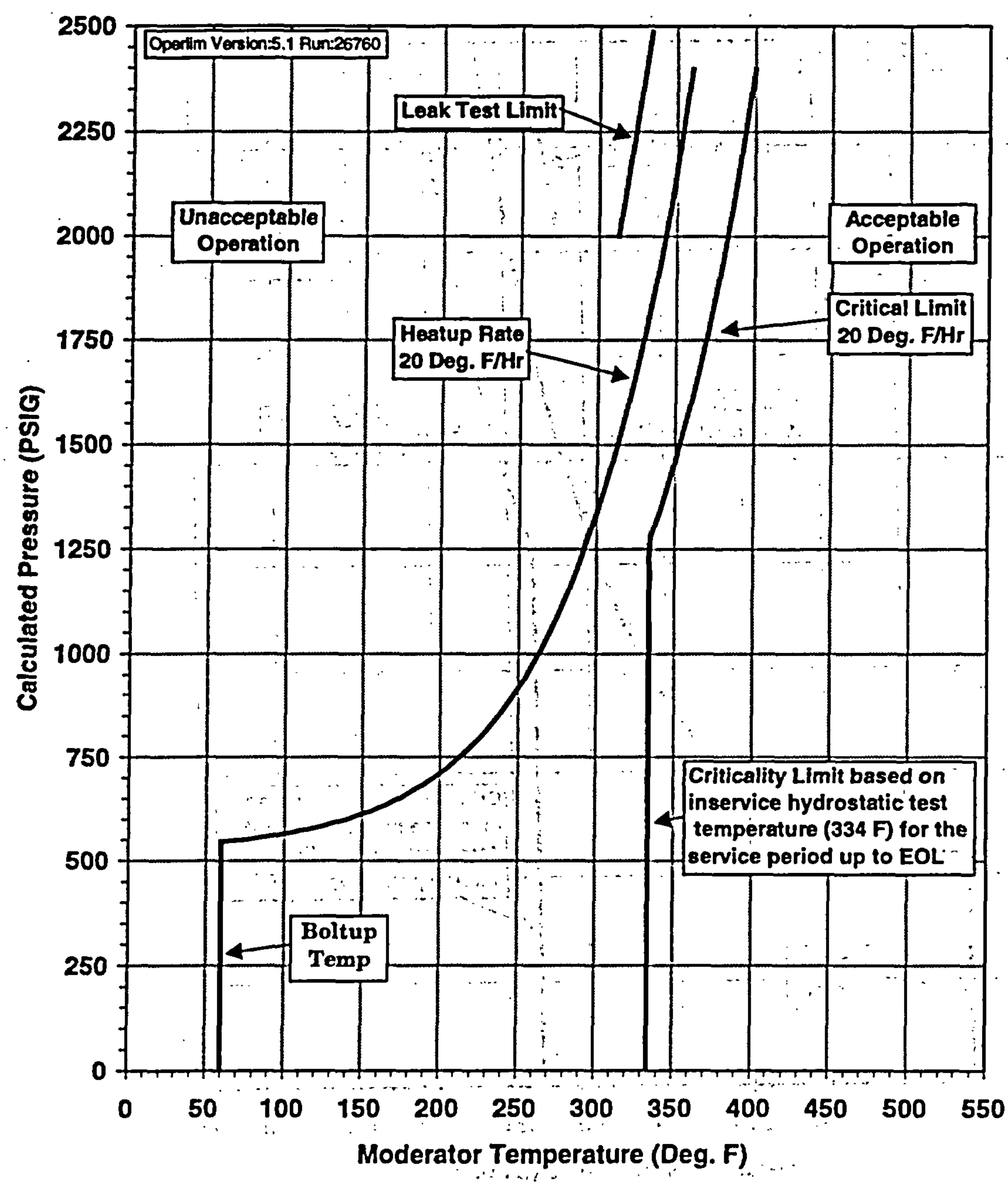

FIGURE 9-1: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 20°F/hr) Applicable to End of License (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)

LIMITING ART VALUES AT EOL: ¼T, 215.7°F
¾T, 169.0°F

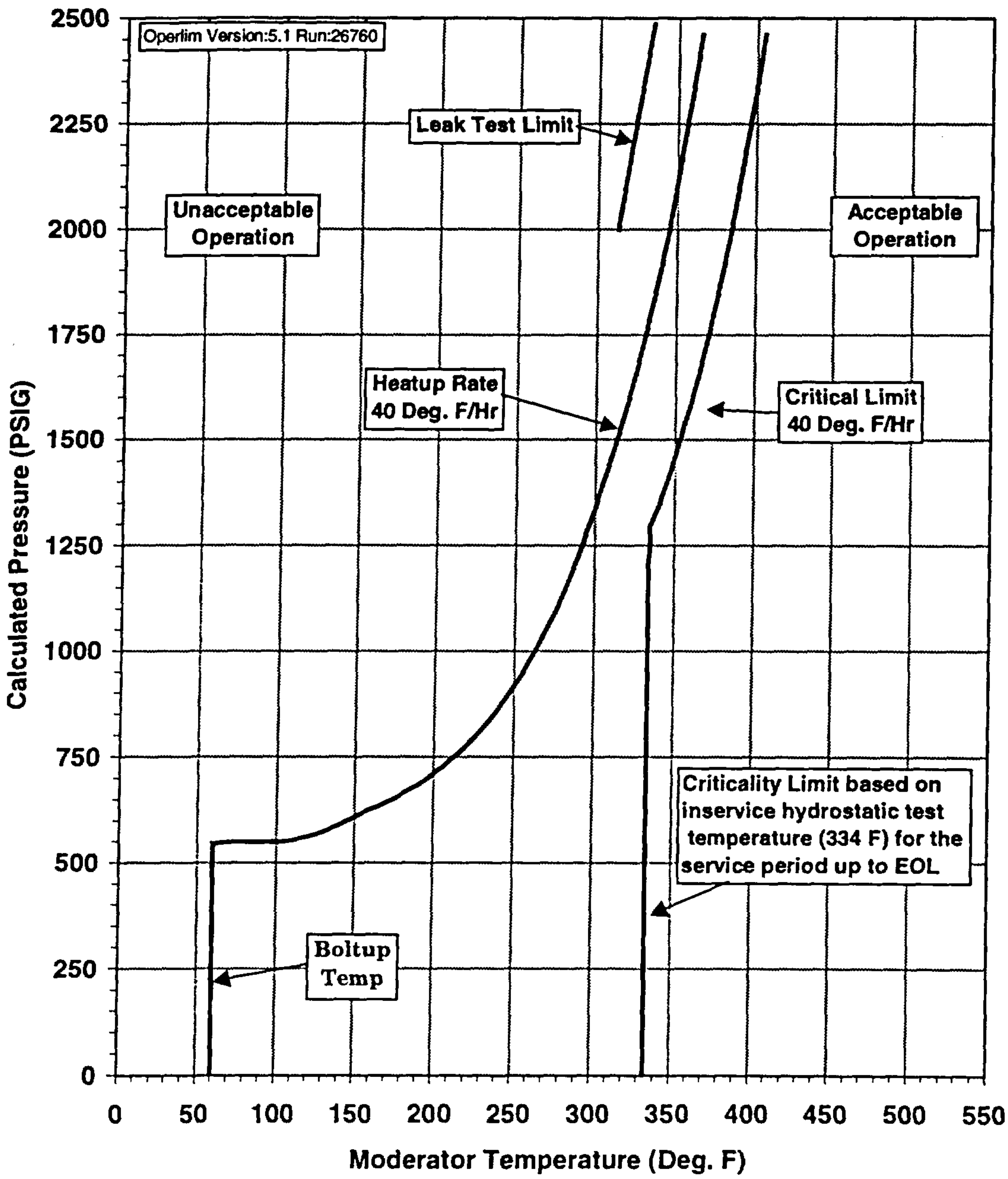

FIGURE 9-2: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 40°F/hr) Applicable to End of License (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)
LIMITING ART VALUES AT EOL: ¼T, 215.7°F
¾T, 169.0°F

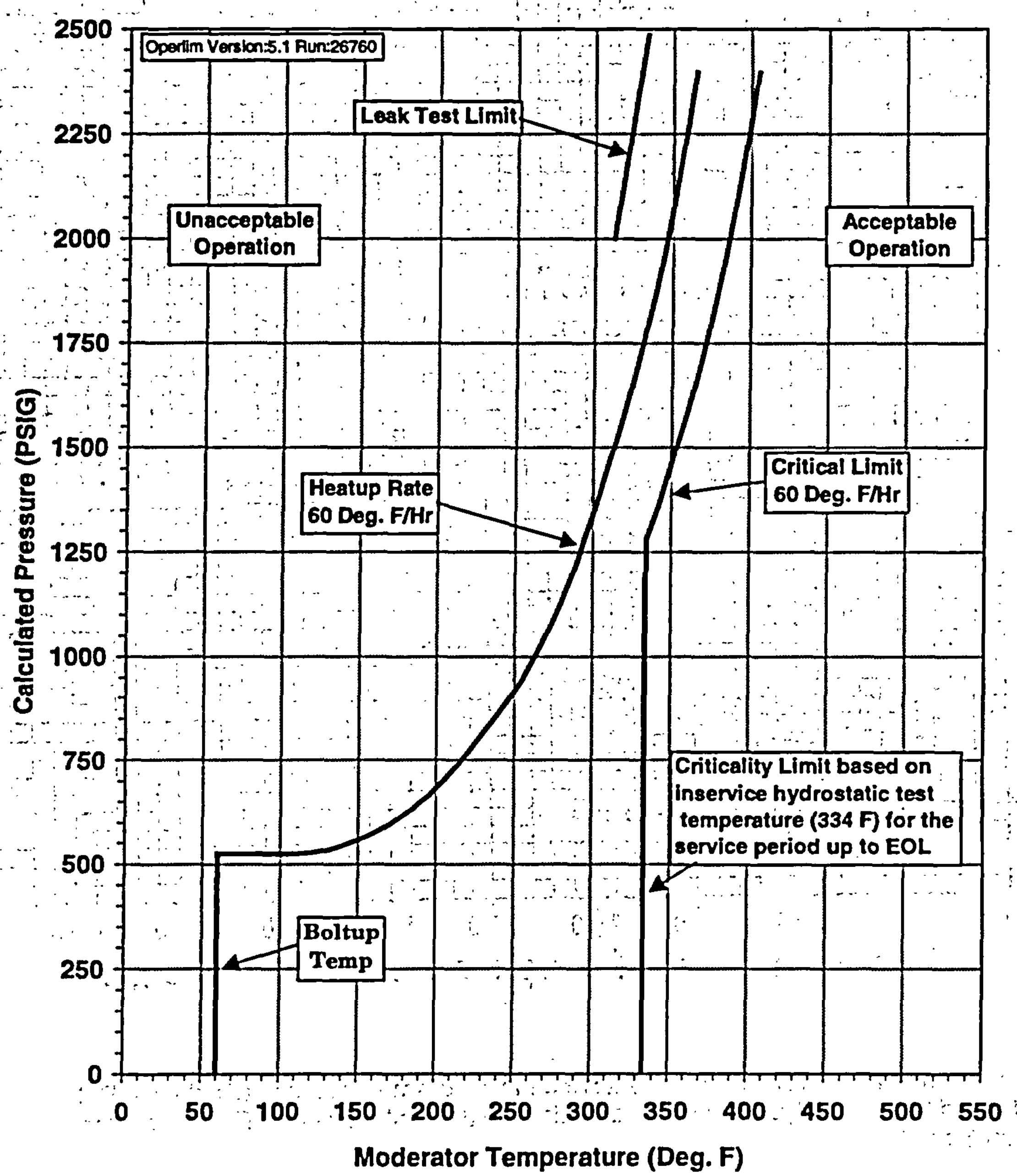

FIGURE 9-3: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable to End of License (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)

LIMITING ART VALUES AT EOL: 1/4T, 215.7°F
3/4T, 169.0°F

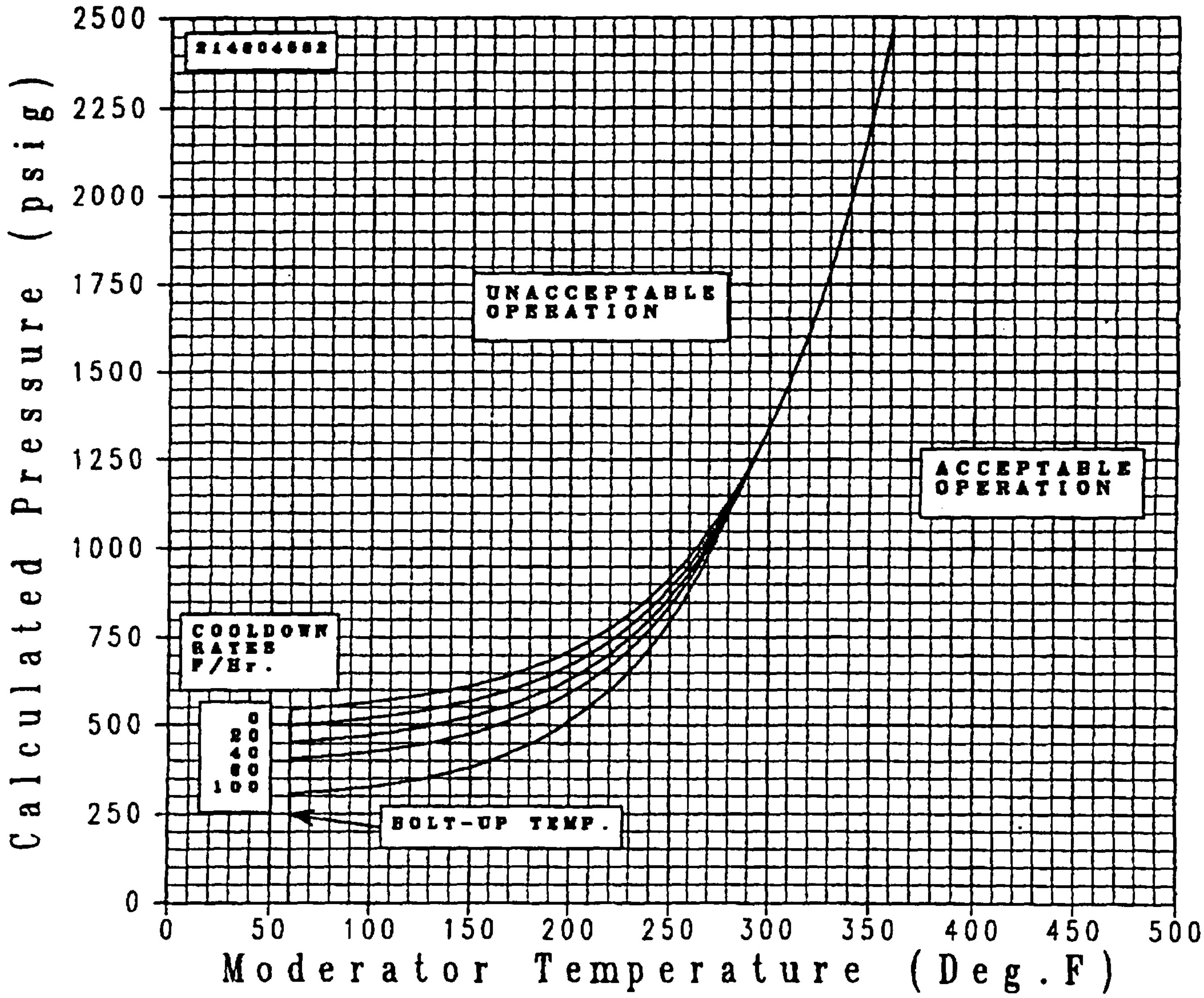

Figure 9-4: Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100 °F/hr) Applicable to End of License (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)
LIMITING ART VALUES AT EOL: ¼T, 238.2°F
¾T, 183.9°F

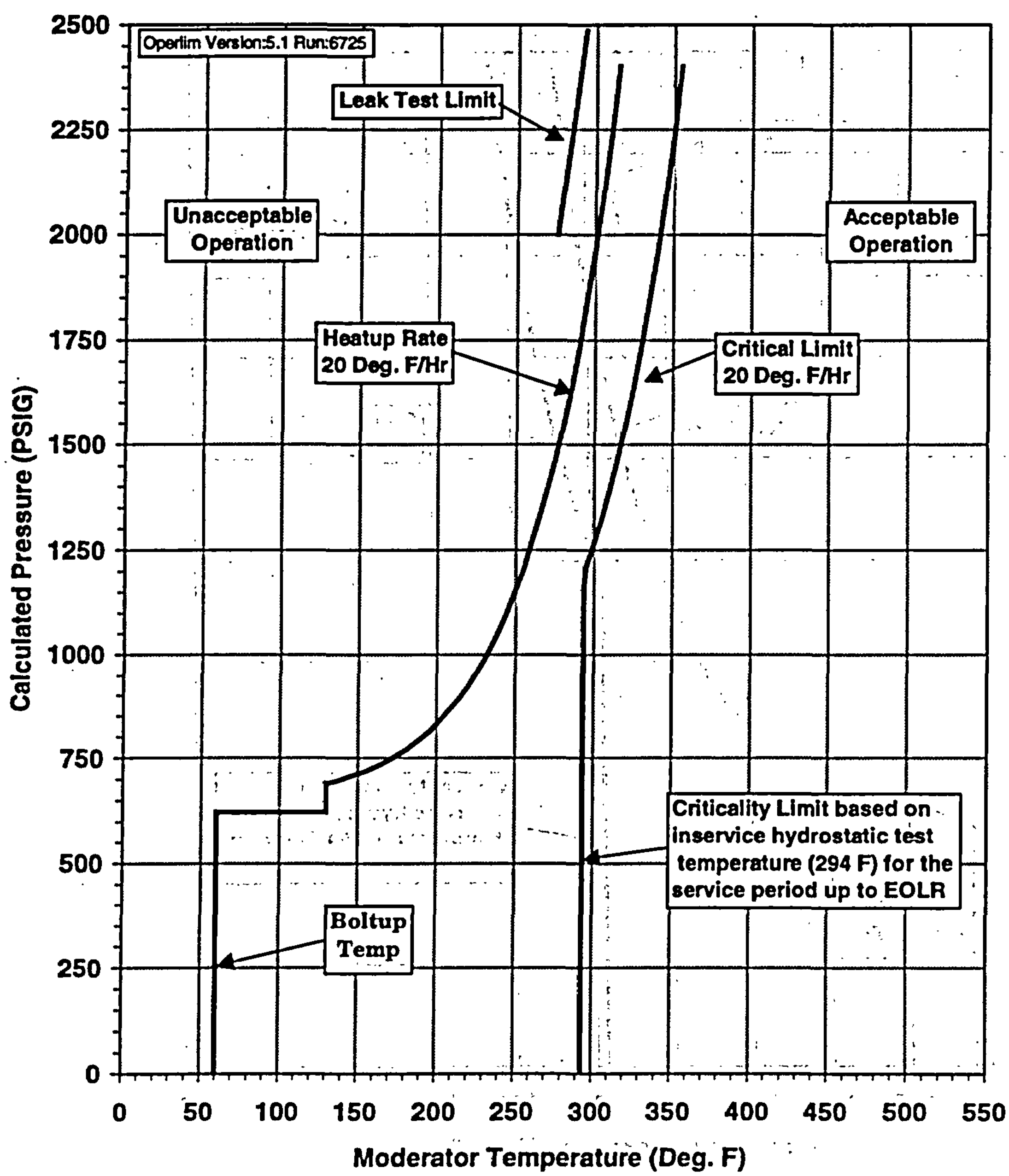

FIGURE 9-5: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 20°F/hr) Applicable to End of License Renewal (With Margins of 0°Fand 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)
LIMITING ART VALUES AT EOL: ¼T, 238.2°F
¾T, 183.9°F

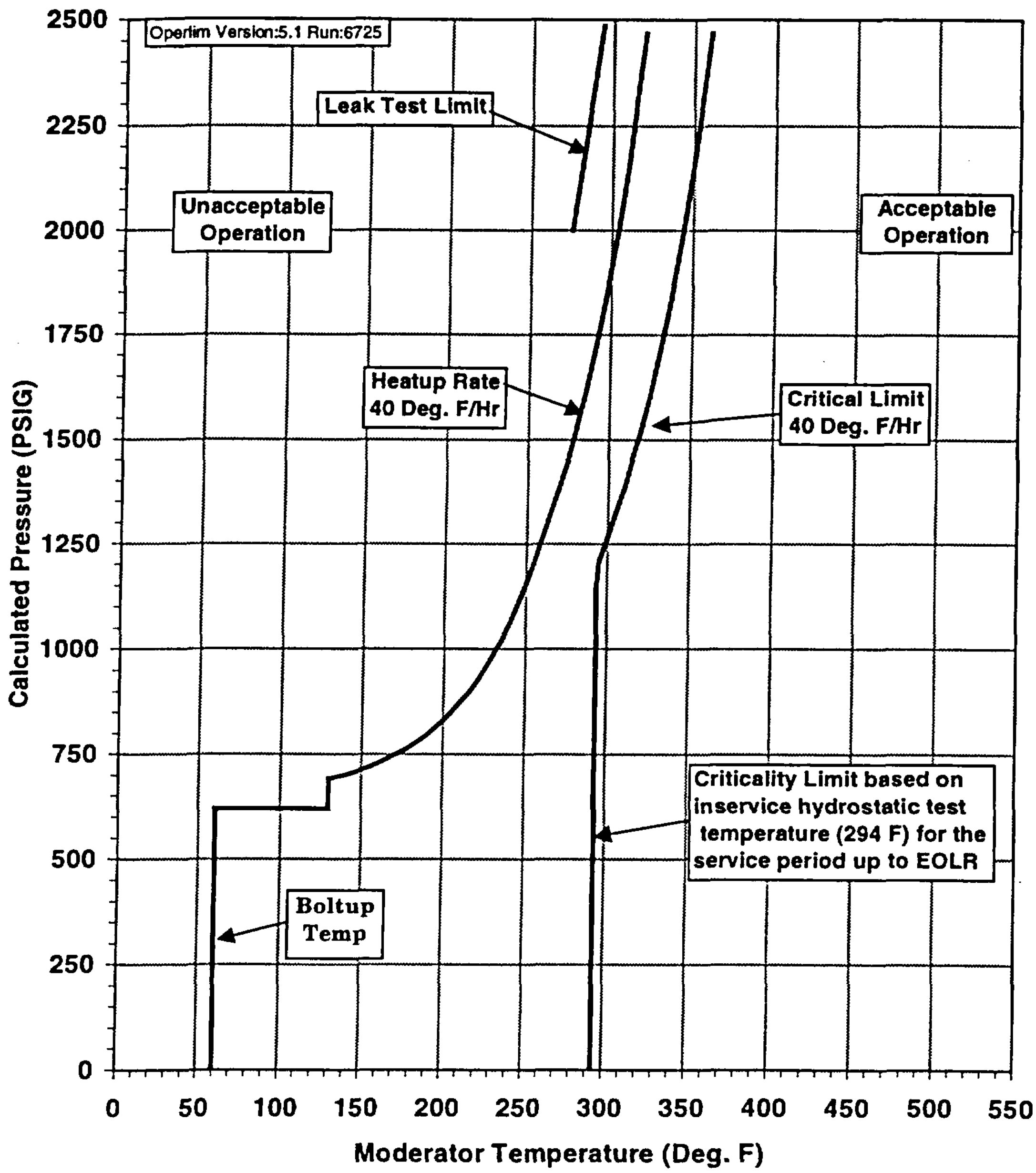

FIGURE 9-6: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 40°F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)
LIMITING ART VALUES AT EOLR: ¼T, 238.2°F
¾T, 183.9°F

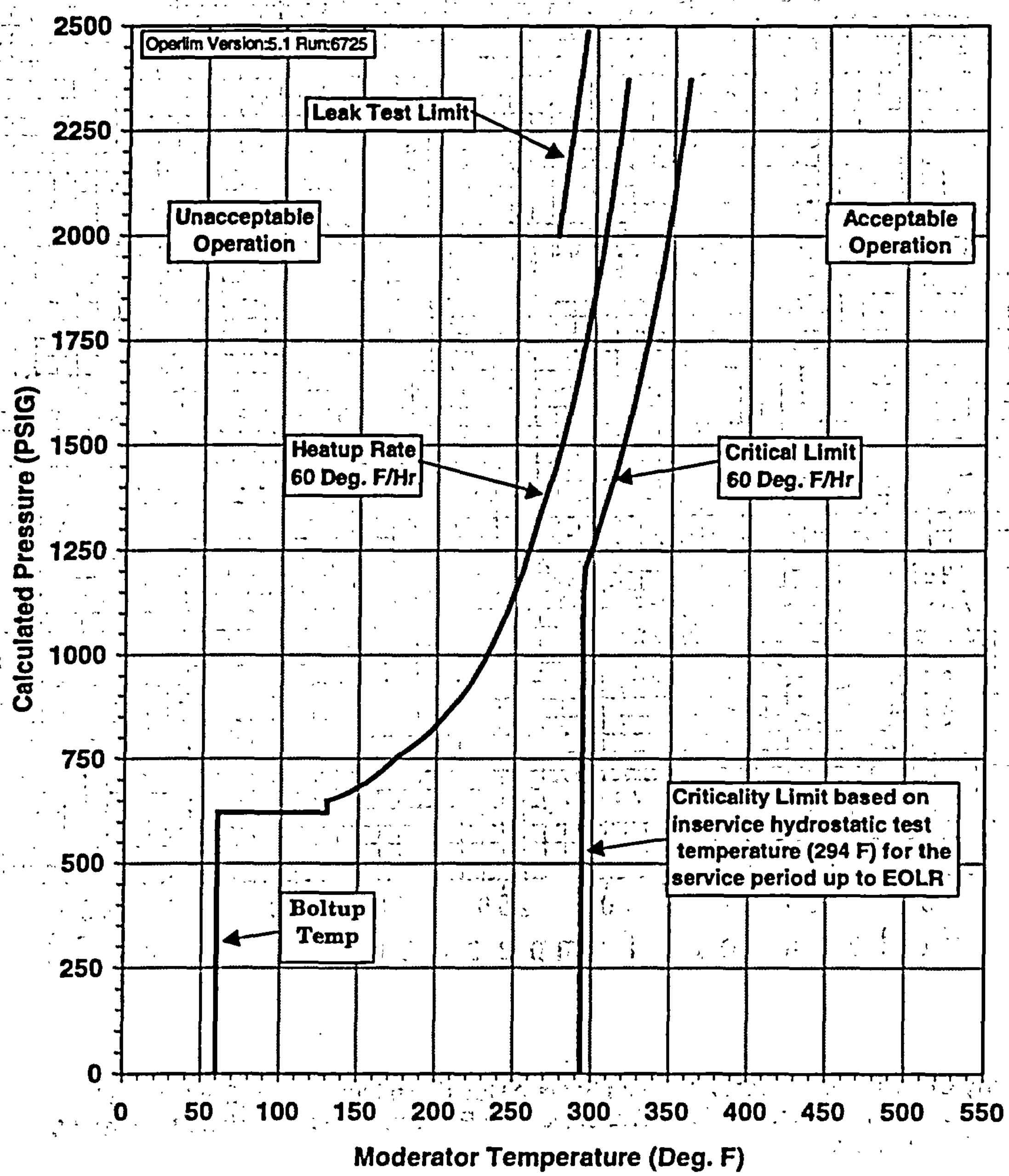

FIGURE 9-7: Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD (1/4T)
INTERMEDIATE TO LOWER SHELL CIRC. WELDS (3/4T)
LIMITING ART VALUES AT EOLR: 1/4T, 238.2°F
3/4T, 183.9°F

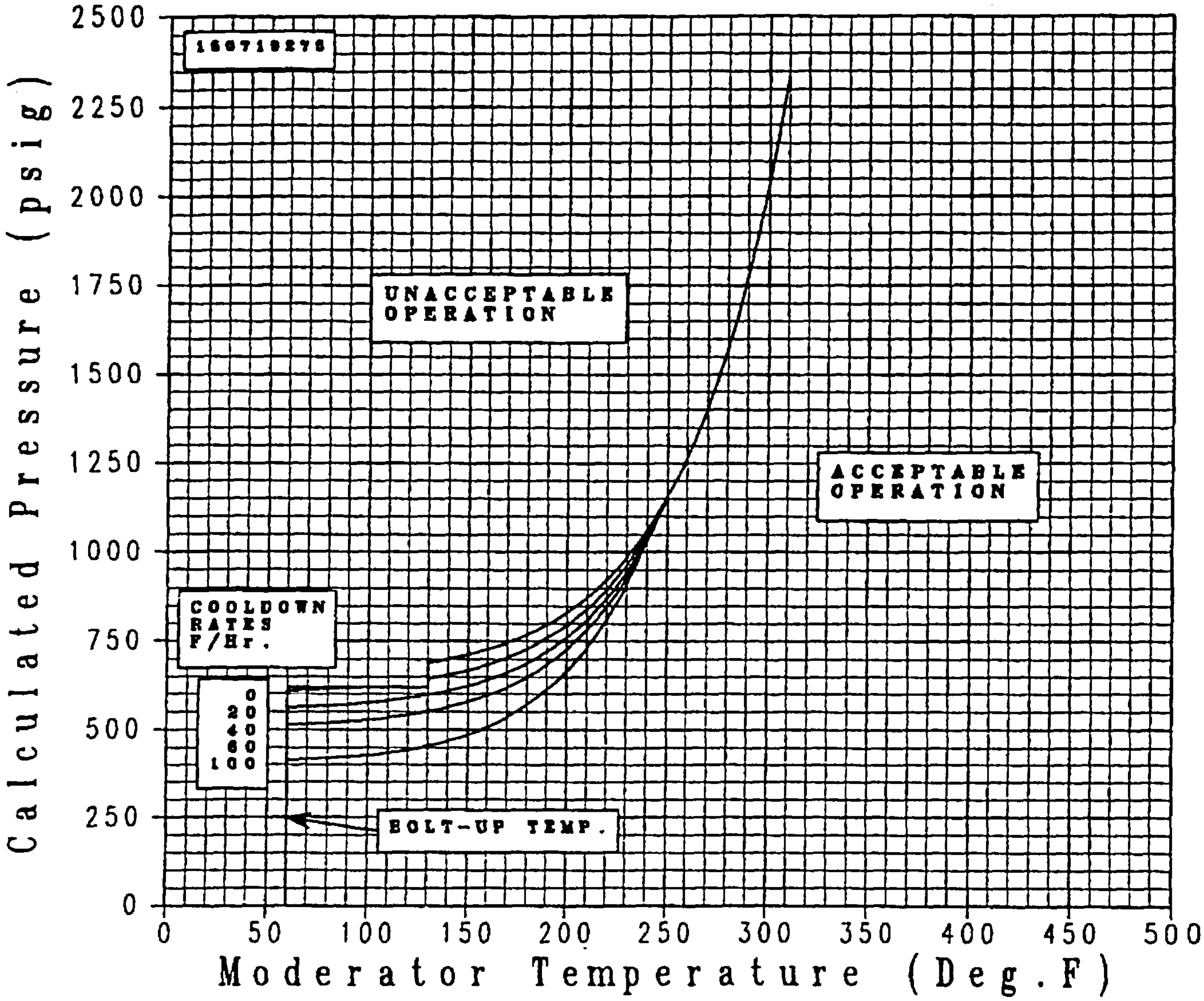

Figure 9-8: Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100 °F/hr) Applicable to End of License Renewal (With Margins of 0°F and 0 psi for Instrumentation Errors)

TABLE 9-1: WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup Data at End of License with Margins of 0°F and 0 psi for Instrumentation Errors (Includes Vessel Flange Requirements of 130°F and 621 psig per 10CFR50)

HEATUP RATE(S) (DEG. F/HR.) = 20.0

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)
1	60	545	21	175	653	40	270	1049
2	65	547	22	180	663	41	275	1089
3	85	556	23	185	674	42	280	1132
4	90	559	24	190	686	43	285	1178
5	95	561	25	195	698	44	290	1227
6	100	565	26	200	712	45	295	1281
7	105	568	27	205	726	46	300	1338
8	110	572	28	210	742	47	305	1398
9	115	576	29	215	758	48	310	1460
10	120	580	30	220	776	49	315	1526
11	125	584	31	225	795	50	320	1597
12	130	589	32	230	816	51	325	1674
13	135	594	33	235	839	52	330	1756
14	140	600	34	240	863	53	335	1845
15	145	606	35	245	888	54	340	1940
16	150	613	36	250	916	55	345	2042
17	155	620	37	255	946	56	350	2152
18	160	627	38	260	978	57	355	2270
19	165	635	39	265	1012	58	360	2397
20	170	644						

HEATUP RATE(S) (DEG. F/HR.) = 40.0

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)
1	60	545	21	175	653	41	275	1089
2	65	547	22	180	663	42	280	1132
3	85	549	23	185	674	43	285	1178
4	90	549	24	190	686	44	290	1227
5	95	549	25	195	698	45	295	1281
6	100	549	26	200	712	46	300	1338
7	105	550	27	205	726	47	305	1399
8	110	553	28	210	742	48	310	1458
9	115	557	29	215	758	49	315	1520
10	120	562	30	220	776	50	320	1586
11	125	568	31	225	795	51	325	1658
12	130	574	32	230	816	52	330	1735
13	135	581	33	235	839	53	335	1817
14	140	590	34	240	863	54	340	1906
15	145	598	35	245	888	55	345	2002
16	150	608	36	250	916	56	350	2104
17	155	618	37	255	946	57	355	2214
18	160	627	38	260	978	58	360	2332
19	165	635	39	265	1012	59	365	2459
20	170	644	40	270	1049			

Table 9-1, Continued

HEATUP RATE(S) (DEG. F/HR.) = 60.0

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)
1	60	524	21	175	607	41	275	1089
2	65	524	22	180	620	42	280	1132
3	85	524	23	185	634	43	285	1178
4	90	524	24	190	649	44	290	1227
5	95	524	25	195	665	45	295	1281
6	100	524	26	200	683	46	300	1338
7	105	524	27	205	702	47	305	1399
8	110	524	28	210	722	48	310	1459
9	115	525	29	215	744	49	315	1517
10	120	526	30	220	767	50	320	1579
11	125	529	31	225	793	51	325	1646
12	130	533	32	230	816	52	330	1718
13	135	538	33	235	839	53	335	1796
14	140	543	34	240	863	54	340	1879
15	145	550	35	245	888	55	345	1968
16	150	557	36	250	916	56	350	2063
17	155	566	37	255	946	57	355	2166
18	160	575	38	260	978	58	360	2276
19	165	585	39	265	1012	59	365	2395
20	170	595	40	270	1049			

Table 9-2: WOG Reactor Vessel 60-year Evaluation Minigroup Cooldown Data at End of License with Margins of 0°F and 0 psi for Instrumentation Errors (Includes Vessel Flange Requirements of 130°F and 621 psig per 10CFR50)

(0 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	544.77	22	165.000	635.18	42	265.000	1012.28
2	65.000	546.66	23	170.000	643.87	43	270.000	1049.31
3	70.000	548.70	24	175.000	653.22	44	275.000	1089.13
4	75.000	550.90	25	180.000	663.27	45	280.000	1131.94
5	80.000	553.26	26	185.000	674.07	46	285.000	1177.97
6	85.000	555.79	27	190.000	685.69	47	290.000	1227.45
7	90.000	558.52	28	195.000	698.18	48	295.000	1280.66
8	95.000	561.45	29	200.000	711.61	49	300.000	1337.86
9	100.000	564.60	30	205.000	726.04	50	305.000	1399.37
10	105.000	567.99	31	210.000	741.57	51	310.000	1465.49
11	110.000	571.64	32	215.000	758.25	52	315.000	1536.59
12	115.000	575.55	33	220.000	776.20	53	320.000	1613.03
13	120.000	579.76	34	225.000	795.49	54	325.000	1695.21
14	125.000	584.29	35	230.000	816.23	55	330.000	1783.57
15	130.000	589.16	36	235.000	838.53	56	335.000	1878.57
16	135.000	594.40	37	240.000	862.51	57	340.000	1980.71
17	140.000	600.02	38	245.000	888.28	58	345.000	2090.53
18	145.000	606.08	39	250.000	916.00	59	350.000	2208.60
19	150.000	612.58	40	255.000	945.80	60	355.000	2335.54
20	155.000	619.58	41	260.000	977.83	61	360.000	2472.02
21	160.000	627.10						

(20 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	499.36	17	140.000	554.96	33	220.000	740.05
2	65.000	501.20	18	145.000	561.21	34	225.000	760.52
3	70.000	503.16	19	150.000	567.94	35	230.000	782.53
4	75.000	505.30	20	155.000	575.21	36	235.000	806.24
5	80.000	507.59	21	160.000	583.02	37	240.000	831.72
6	85.000	510.09	22	165.000	591.46	38	245.000	859.17
7	90.000	512.78	23	170.000	600.53	39	250.000	888.68
8	95.000	515.71	24	175.000	610.33	40	255.000	920.46
9	100.000	518.85	25	180.000	620.85	41	260.000	954.63
10	105.000	522.26	26	185.000	632.20	42	265.000	991.41
11	110.000	525.93	27	190.000	644.41	43	270.000	1030.96
12	115.000	529.90	28	195.000	657.57	44	275.000	1073.53
13	120.000	534.18	29	200.000	671.72	45	280.000	1119.30
14	125.000	538.81	30	205.000	686.98	46	285.000	1168.57
15	130.000	543.78	31	210.000	703.37	47	290.000	1221.54
16	135.000	549.17	32	215.000	721.05	48	295.000	1278.55

Table 9-2, Continued

(40 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	453.02	17	140.000	509.11	33	220.000	704.14
2	65.000	454.75	18	145.000	515.60	34	225.000	725.90
3	70.000	456.64	19	150.000	522.59	35	230.000	749.32
4	75.000	458.71	20	155.000	530.16	36	235.000	774.57
5	80.000	460.96	21	160.000	538.32	37	240.000	801.73
6	85.000	463.43	22	165.000	547.14	38	245.000	831.01
7	90.000	466.10	23	170.000	556.65	39	250.000	862.52
8	95.000	469.01	24	175.000	566.92	40	255.000	896.46
9	100.000	472.16	25	180.000	577.99	41	260.000	932.98
10	105.000.	475.60	26	185.000	589.94	42	265.000	972.33
11	110.000	479.32	27	190.000	602.82	43	270.000	1014.65
12	115.000	483.36	28	195.000	616.71	44	275.000	1060.25
13	120.000	487.72	29	200.000	631.68	45	280.000	1109.29
14	125.000	492.46	30	205.000	647.82	46	285.000	1162.12
15	130.000	497.58	31	210.000	665.21	47	290.000	1218.94
16	135.000	503.12	32	215.000	683.96	48	295.000	1280.12

(60 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	405.62	17	140.000	462.47	33	220.000	668.57
2	65.000	407.26	18	145.000	469.21	34	225.000	691.75
3	70.000	409.09	19	150.000	476.51	35	230.000	716.74
4	75.000	411.11	20	155.000	484.42	36	235.000	743.68
5	80.000	413.32	21	160.000	492.96	37	240.000	772.71
6	85.000	415.76	22	165.000	502.22	38	245.000	804.00
7	90.000	418.42	23	170.000	512.22	39	250.000	837.71
8	95.000	421.33	24	175.000	523.03	40	255.000	874.04
9	100.000	424.51	25	180.000	534.71	41	260.000	913.17
10	105.000	427.99	26	185.000	547.33	42	265.000	955.34
11	110.000	431.77	27	190.000	560.95	43	270.000	1000.74
12	115.000	435.89	28	195.000	575.66	44	275.000	1049.65
13	120.000	440.36	29	200.000	591.53	45	280.000	1102.32
14	125.000	445.24	30	205.000	608.66	46	285.000	1159.05
15	130.000	450.52	31	210.000	627.14	47	290.000	1220.12
16	135.000	456.26	32	215.000	647.08			

(100 DEG-F/HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	307.52	17	140.000	366.79	32	215.000	574.65
2	65.000	309.03	18	145.000	374.18	33	220.000	599.21
3	70.000	310.75	19	150.000	382.21	34	225.000	625.75
4	75.000	312.69	20	155.000	390.95	35	230.000	654.39
5	80.000	314.85	21	160.000	400.43	36	235.000	685.32
6	85.000	317.26	22	165.000	410.74	37	240.000	718.69
7	90.000	319.93	23	170.000	421.90	38	245.000	754.70
8	95.000	322.90	24	175.000	434.02	39	250.000	793.54
9	100.000	326.17	25	180.000	447.14	40	255.000	835.44
10	105.000	329.78	26	185.000	461.35	41	260.000	880.62
11	110.000	333.75	27	190.000	476.73	42	265.000	929.35
12	115.000	338.10	28	195.000	493.39	43	270.000	981.86
13	120.000	342.87	29	200.000	511.39	44	275.000	1038.48
14	125.000	348.09	30	205.000	530.86	45	280.000	1099.49
15	130.000	353.79	31	210.000	551.90	46	285.000	1165.25
16	135.000	360.02						

TABLE 9-3: WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup Data at End of License Renewal with Margins of 0°F and 0 psi for Instrumentation Errors (Includes Vessel Flange Requirements of 130°F and 621 psig per 10CFR50)

HEATUP RATE(S) (DEG. F/HR.) = 20.0

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)
1	60	621	18	155	720	35	240	1061
2	65	621	19	160	728	36	245	1105
3	85	621	20	165	737	37	250	1153
4	90	621	21	170	747	38	255	1207
5	95	621	22	175	758	39	260	1266
6	100	621	23	180	770	40	265	1326
7	105	621	24	185	783	41	270	1392
8	110	621	25	190	797	42	275	1465
9	115	621	26	195	814	43	280	1545
10	120	621	27	200	831	44	285	1634
11	125	621	28	205	851	45	290	1732
12	130	621	29	210	873	46	295	1840
13	130	690	30	215	897	47	300	1960
14	135	695	31	220	924	48	305	2092
15	140	700	32	225	953	49	310	2239
16	145	706	33	230	985	50	315	2400
17	150	713	34	235	1021			

HEATUP RATE(S) (DEG. F/HR.) = 40.0

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)
1	60	621	18	155	720	35	240	1061
2	65	621	19	160	728	36	245	1105
3	85	621	20	165	737	37	250	1153
4	90	621	21	170	747	38	255	1207
5	95	621	22	175	758	39	260	1266
6	100	621	23	180	770	40	265	1325
7	105	621	24	185	783	41	270	1386
8	110	621	25	190	797	42	275	1452
9	115	621	26	195	814	43	280	1525
10	120	621	27	200	831	44	285	1606
11	125	621	28	205	851	45	290	1696
12	130	621	29	210	873	46	295	1795
13	130	690	30	215	897	47	300	1904
14	135	695	31	220	924	48	305	2024
15	140	700	32	225	953	49	310	2157
16	145	706	33	230	985	50	315	2304
17	150	713	34	235	1021	51	320	2466

Table 9-3, Continued

HEATUP RATE(S) (DEG. F/HR.) = 60.0

	INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG. F)	INDICATED PRESSURE (PSIG)
1	60	621	18	155	694	35	240	1061
2	65	621	19	160	707	36	245	1105
3	85	621	20	165	721	37	250	1153
4	90	621	21	170	738	38	255	1207
5	95	621	22	175	756	39	260	1266
6	100	621	23	180	770	40	265	1329
7	105	621	24	185	783	41	270	1384
8	110	621	25	190	797	42	275	1445
9	115	621	26	195	814	43	280	1512
10	120	621	27	200	831	44	285	1586
11	125	621	28	205	851	45	290	1668
12	130	621	29	210	873	46	295	1758
13	130	649	30	215	897	47	300	1858
14	135	656	31	220	924	48	305	1967
15	140	663	32	225	953	49	310	2089
16	145	672	33	230	985	50	315	2222
17	150	682	34	235	1021	51	320	2370

Table 9-4: WOG Reactor Vessel 60-year Evaluation Minigroup Cooldown Data at End of License Renewal with Margins of 0°F and 0 psi for Instrumentation Errors (Includes Vessel Flange Requirements of 130°F and 621 psig per 10CFR50)

(STEADY-STATE COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	621.00	19	145.000	706.38	36	230.000	985.43
2	65.000	621.00	20	150.000	712.94	37	235.000	1021.34
3	70.000	621.00	21	155.000	720.19	38	240.000	1061.02
4	75.000	621.00	22	160.000	728.20	39	245.000	1104.88
5	80.000	621.00	23	165.000	737.06	40	250.000	1153.35
6	85.000	621.00	24	170.000	746.84	41	255.000	1206.91
7	90.000	621.00	25	175.000	757.66	42	260.000	1266.11
8	95.000	621.00	26	180.000	769.61	43	265.000	1331.54
9	100.000	621.00	27	185.000	782.82	44	270.000	1403.85
10	105.000	621.00	28	190.000	797.42	45	275.000	1483.76
11	110.000	621.00	29	195.000	813.55	46	280.000	1572.08
12	115.000	621.00	30	200.000	831.38	47	285.000	1669.68
13	120.000	621.00	31	205.000	851.09	48	290.000	1777.56
14	125.000	621.00	32	210.000	872.87	49	295.000	1896.77
15	130.000	621.00	33	215.000	896.94	50	300.000	2028.53
16	130.000	690.22	34	220.000	923.54	51	305.000	2174.14
17	135.000	695.08	35	225.000	952.94	52	310.000	2335.06
18	140.000	700.45						

(20 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	609.42	15	130.000	621.00	28	190.000	759.39
2	65.000	610.55	16	130.000	644.51	29	195.000	776.89
3	70.000	611.78	17	135.000	649.62	30	200.000	796.23
4	75.000	613.17	18	140.000	655.27	31	205.000	817.67
5	80.000	614.72	19	145.000	661.56	32	210.000	841.36
6	85.000	616.47	20	150.000	668.51	33	215.000	867.60
7	90.000	618.41	21	155.000	676.25	34	220.000	896.61
8	95.000	620.59	22	160.000	684.79	35	225.000	928.73
9	100.000	621.00	23	165.000	694.29	36	230.000	964.23
10	105.000	621.00	24	170.000	704.79	37	235.000	1003.53
11	110.000	621.00	25	175.000	716.44	38	240.000	1046.97
12	115.000	621.00	26	180.000	729.31	39	245.000	1095.04
13	120.000	621.00	27	185.000	743.60	40	250.000	1148.19
14	125.000	621.00						

(40 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	562.49	14	125.000	593.23	27	190.000	721.72
2	65.000	563.48	15	130.000	598.05	28	195.000	740.76
3	70.000	564.63	16	135.000	603.45	29	200.000	761.83
4	75.000	565.95	17	140.000	609.44	30	205.000	785.21
5	80.000	567.45	18	145.000	616.13	31	210.000	811.08
6	85.000	569.16	19	150.000	623.56	32	215.000	839.76
7	90.000	571.08	20	155.000	631.83	33	220.000	871.50
8	95.000	573.27	21	160.000	641.01	34	225.000	906.67
9	100.000	575.71	22	165.000	651.23	35	230.000	945.59
10	105.000	578.47	23	170.000	662.55	36	235.000	988.69
11	110.000	581.56	24	175.000	675.14	37	240.000	1036.39
12	115.000	585.03	25	180.000	689.08	38	245.000	1089.20
13	120.000	588.89	26	185.000	704.57	39	250.000	1147.63

Table 9-4, Continued

(60 DEG-F / HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	514.47	14	125.000	545.74	27	190.000	684.59
2	65.000	515.37	15	130.000	550.85	28	195.000	705.36
3	70.000	516.44	16	135.000	556.58	29	200.000	728.40
4	75.000	517.70	17	140.000	562.97	30	205.000	753.97
5	80.000	519.16	18	145.000	570.13	31	210.000	782.31
6	85.000	520.84	19	150.000	578.10	32	215.000	813.75
7	90.000	522.77	20	155.000	587.01	33	220.000	848.58
8	95.000	524.97	21	160.000	596.91	34	225.000	887.20
9	100.000	527.47	22	165.000	607.95	35	230.000	929.98
10	105.000	530.30	23	170.000	620.22	36	235.000	977.39
11	110.000	533.50	24	175.000	633.87	37	240.000	1029.89
12	115.000	537.12	25	180.000	649.03	38	245.000	1088.06
13	120.000	541.17	26	185.000	665.89	39	250.000	1152.46

(100 DEG-F/HR COOLDOWN)

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSIG)
1	60.000	415.07	14	125.000	448.40	27	190.000	612.86
2	65.000	415.82	15	130.000	454.25	28	195.000	637.87
3	70.000	416.78	16	135.000	460.85	29	200.000	665.66
4	75.000	417.95	17	140.000	468.26	30	205.000	696.57
5	80.000	419.37	18	145.000	476.60	31	210.000	730.88
6	85.000	421.06	19	150.000	485.94	32	215.000	769.00
7	90.000	423.04	20	155.000	496.40	33	220.000	811.31
8	95.000	425.35	21	160.000	508.10	34	225.000	858.27
9	100.000	428.02	22	165.000	521.18	35	230.000	910.36
10	105.000	431.11	23	170.000	535.77	36	235.000	968.15
11	110.000	434.63	24	175.000	552.05	37	240.000	1032.23
12	115.000	438.65	25	180.000	570.18	38	245.000	1103.29
13	120.000	443.22	26	185.000	590.38			

10. Enable Temperature Calculation

10.1 ASME Code Case N-514 Methodology

ASME Code Case N-514 requires that the LTOP or COMS system be in operation at coolant temperatures less than 200°F or at coolant temperatures less than a temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. RT_{NDT} is the highest adjusted reference temperature (ART) for the limiting beltline material at a distance one fourth of the vessel section thickness from the vessel inside surface (i.e., clad/base metal interface), as determined by Regulatory Guide 1.99, Revision 2 [1]

10.2 Enable Temperature Calculation:

10.2.1 End of License Enable Temperature

The highest calculated 1/4T ART for the Surry Units 1 and 2 reactor vessel beltline region at the end of license EFPY is 215.7°F.

From the OPERLIM computer code output for the N. Anna Units 1 and 2 end of license pressure temperature limit curves without margins the maximum DT_{metal} is:

Cooldown Rate (Steady-State Cooldown):
max (DT_{metal}) at 1/4T = 0°F

Heatup Rate of 60°F/Hr:
max (DT_{metal}) at 1/4T = 36.1°F

Minimum Enable Temperature (ENBT) = RT_{NDT} + 50 + max (DT_{metal}), °F
= (215.7 + 50 + 36.1) °F
= 301.8°F

The minimum required enable temperature for the Surry Units 1 and 2 Reactor Vessel are conservatively chosen to be 305°F for the end of license pressure temperature limits.

10.2.2 End of License Renewal Enable Temperature Calculation

The highest calculated 1/4T ART for the Surry Units 1 and 2 reactor vessel beltline region at the end of license renewal EFPY is 238.2°F.

From the OPERLIM computer code output for the Surry Units 1 and 2 end of license renewal pressure temperature limit curves without margins, the maximum DT_{metal} is:

Cooldown Rate (Steady-State Cooldown):
max (DT_{metal}) at 1/4T = 0°F

Heatup Rate of 60°F/Hr:
max (DT_{metal}) at 1/4T = 36.1°F

Minimum Enable Temperature (ENBT) = RT_{NDT} + 50 + max (DT_{metal}), °F
= (238.2 + 50 + 36.1) °F
= 324.3°F

The minimum required enable temperature for the Surry Units 1 and 2 Reactor Vessel are conservatively chosen to be 325°F for end of license renewal EFPY.

11. REFERENCES

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

2 10 CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995.

3 1989 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

4 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

5 WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", J. D. Andrachek, et al., January 1996.

6 Virginia Power Calculation Number SM-1008, "Reactor Vessel Calculations and Data to Support RV Aging Management Report and NRC RAI on Generic Letter 92-01 Supplement 1, Surry and Surry Units 1 and 2.

7 "Meeting Summary for November 12, 1997 Meeting with Owners Group Representatives and NEI Regarding Review of Responses to Generic Letter 92-01, Revision 1, Supplement 1 Responses," Memorandum to E.J. Sullivan from K.R. Wichman, Materials and Chemical Engineering Branch, dated November 19, 1997"

8 Virginia Power Topical Report VEP-NAF-3 "Reactor Vessel Fluence Methodology".

9 Letter from Virginia Power to USNRC Serial Number 98-252, June 18, 1998.

10 Electric Power Research Institute (EPRI) report NP-6114.

11 American Society for Testing and Materials (ASTM) report STP-1046.

APPENDIX E

Westinghouse Letter VPA-03-193 dated October 9, 2003 "Dominion Generation, Surry Units 1 and 2, Thermal Stress Intensity Factors and Vessel Wall Temperatures for PT Curves from WCAP-15130, Revision 1"

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Our ref: VPA-03-193

October 9, 2003

DOMINION GENERATION
SURRY UNITS 1 AND 2
Thermal Stress Intensity Factors and Vessel Wall Temperatures for PT Curves from WCAP-15130, Revision 1

Dear Mr. Margolis:

Reference: (1) WCAP-15130, Revision 1, "Surry Units 1 and 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," April 2001.

Per your request, Westinghouse has extracted the thermal stress intensity factors for the end-of-license-renewal PT Limit curves from Reference 1 above. In addition, the vessel wall temperatures (¼ & ¾ thickness only) were also obtained. All this information is present in Tables 1 and 2 of Attachment 1. As a note, for proprietary concerns, the requested information was obtained for just the maximum heatup and cooldown rates and therefore is non-proprietary. Based on past experiences, this should be sufficient to satisfy any NRC questions.

Please contact Mr. Tom Laubham at (412) 374-6788 or me on (412) 374-6345 if you have any questions regarding this information.

Very truly yours,

WESTINGHOUSE ELECTRIC COMPANY

William R Rice

W. R. Rice
Customer Projects Manager

Cc: J. Harrell

/

bcc: W. R. Rice
S. M. DiTommaso
T. Laubham
VRA File

Reference:

ATTACHMENT 1

TABLE 1
Kit Values for 60°F/hr Heatup Curve (EOLR)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 60°F/hr Heatup (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	Vessel Temperature @ 3/4T Location for 60°F/hr Heatup (°F)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
75	71.53	-1.0783	70.17	0.6035
80	74.86	-2.4166	70.94	1.6199
85	78.38	-3.3610	72.58	2.4071
PT Curves are Limited by the SS up to Temp. = 85°F, then limited by the ¾ T location up to Temp. = 210°F				
90	82.23	-4.1929	74.89	3.0490
95	86.31	-4.8132	77.72	3.5571
100	90.55	-5.3471	80.98	3.9718
105	94.96	-5.7538	84.58	4.3030
110	99.45	-6.1038	88.45	4.5747
115	104.07	-6.3735	92.52	4.7938
120	108.73	-6.6083	96.77	4.9753
125	113.47	-6.7908	101.15	5.1236
130	118.24	-6.9528	105.64	5.2481
135	123.06	-7.0801	110.21	5.3514
140	127.89	-7.1959	114.85	5.4397
145	132.76	-7.2882	119.55	5.5146
150	137.64	-7.3746	124.29	5.5799
155	142.55	-7.4447	129.06	5.6366
160	147.45	-7.5123	133.86	5.6874
165	152.37	-7.5683	138.69	5.7325
170	157.30	-7.6239	143.53	5.7739
175	162.23	-7.6710	148.38	5.8117
180	167.17	-7.7189	153.24	5.8472
185	172.11	-7.7604	158.12	5.8803
190	177.06	-7.8033	163.00	5.9120
195	182.00	-7.8413	167.89	5.9421
200	186.95	-7.8810	172.78	5.9714
205	191.90	-7.9168	177.67	5.9996
210	196.86	-7.9544	182.57	6.0273
PT Curves are limited by the SS from Temp. = 215°F to 260°F, then limited by the ¼ T location up to Temp. = 320°F				
215	201.81	-7.9888	187.47	6.0544

220	206.76	-8.0252	192.37	6.0812
225	211.72	-8.0588	197.27	6.1075
230	216.67	-8.0943	202.18	6.1337
235	221.62	-8.1275	207.08	6.1596
240	226.58	-8.1625	211.99	6.1854
245	231.53	-8.1954	216.90	6.2110
250	236.49	-8.2301	221.80	6.2367
255	241.45	-8.2630	226.71	6.2622
260	246.40	-8.2975	231.62	6.2877
265	251.36	-8.3304	236.53	6.3132
270	256.31	-8.3648	241.43	6.3387
275	261.27	-8.3977	246.34	6.3642
280	266.23	-8.4321	251.25	6.3897
285	271.18	-8.4652	256.16	6.4152
290	276.14	-8.4996	261.06	6.4408
295	281.09	-8.5328	265.97	6.4664
300	286.05	-8.5673	270.88	6.4921
305	291.00	-8.6006	275.78	6.5177
310	295.96	-8.6351	280.69	6.5434
315	300.91	-8.6686	285.60	6.5692
320	305.87	-8.7032	290.51	6.5950

Vessel Radius to the ¼T and ¾T Locations are as follows:

- 1/4T Radius = 81.130" &
- 3/4T Radius = 85.170"

TABLE 2
Kit Values for 100°F/hr Cooldown Curve (EOLR)

Water Temp. (°F)	Vessel Temperature @ 1/4T Location for 100°F/hr Cooldown (°F)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
From Temp. = 250°F to 310°F the 100°F/hr. the Cooldown Curve is limited by the lower rates or SS.		
245	269.20	14.8707
240	264.13	14.8127
235	259.05	14.7538
230	253.97	14.6953
225	248.90	14.6360
220	243.82	14.5771
215	238.74	14.5176
210	233.67	14.4585
205	228.59	14.3988
200	223.51	14.3396
195	218.43	14.2799
190	213.36	14.2206
185	208.28	14.1609
180	203.20	14.1017
175	198.12	14.0420
170	193.05	13.9829
165	187.97	13.9233
160	182.89	13.8643
155	177.81	13.8049
150	172.74	13.7460
145	167.66	13.6868
140	162.58	13.6281
135	157.50	13.5690
130	152.43	13.5105
125	147.35	13.4517
120	142.27	13.3934
115	137.20	13.3349
110	132.12	13.2768
105	127.04	13.2185
100	121.97	13.1607
95	116.89	13.1026
90	111.82	13.0450
85	106.74	12.9872
80	101.66	12.9299

75	96.59	12.8723
70	91.51	12.8153
65	86.44	12.7580
60	81.36	12.7004

ATTACHMENT 2

Mark-up of TS Pages

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 47.6 ~~28.8~~ Effective Full Power Years (EFPY) and 48.1 ~~29.4~~ EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (238.2 ~~228.4~~°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results are presented in UFSAR Section 4.1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 47.6 ~~28.8~~ EFPY and 48.1 ~~29.4~~ EFPY for Units 1 and 2, respectively (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 47.6 ~~28.8~~ EFPY or 48.1 ~~29.4~~ EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

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, $\cancel{K_{IR}}$ K_{IC}, for the metal temperature at that time. $\cancel{K_{IR}}$ K_{IC} is obtained from the reference fracture toughness curve, defined in ~~Appendix G~~ Section XI to the ASME Code. The $\cancel{K_{IR}}$ K_{IC} curve is given by the equation:

$$\cancel{K_{IR} = 26.78 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)]} \quad (1)$$

$$K_{IC} = 33.2 + 20.734 \exp[0.02(T - RT_{NDT})]$$

where $\cancel{K_{IR}}$ K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{It} \leq \cancel{K_{IR}}\ K_{IC} \quad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients

$\cancel{K_{IR}}$ K_{IC} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, $\cancel{K_{IR}}$ K_{IC} 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 temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 47.6 ~~28.8~~ EFPY and 48.1 ~~29.4~~ EFPY for Units 1 and 2, respectively. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of ≤ 39~~0~~5 psig and verify each PORV block valve is open at least once per 72 hours,

or

(5) The RCS shall be vented through one open PORV or an equivalent size opening as specified below:

(a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or

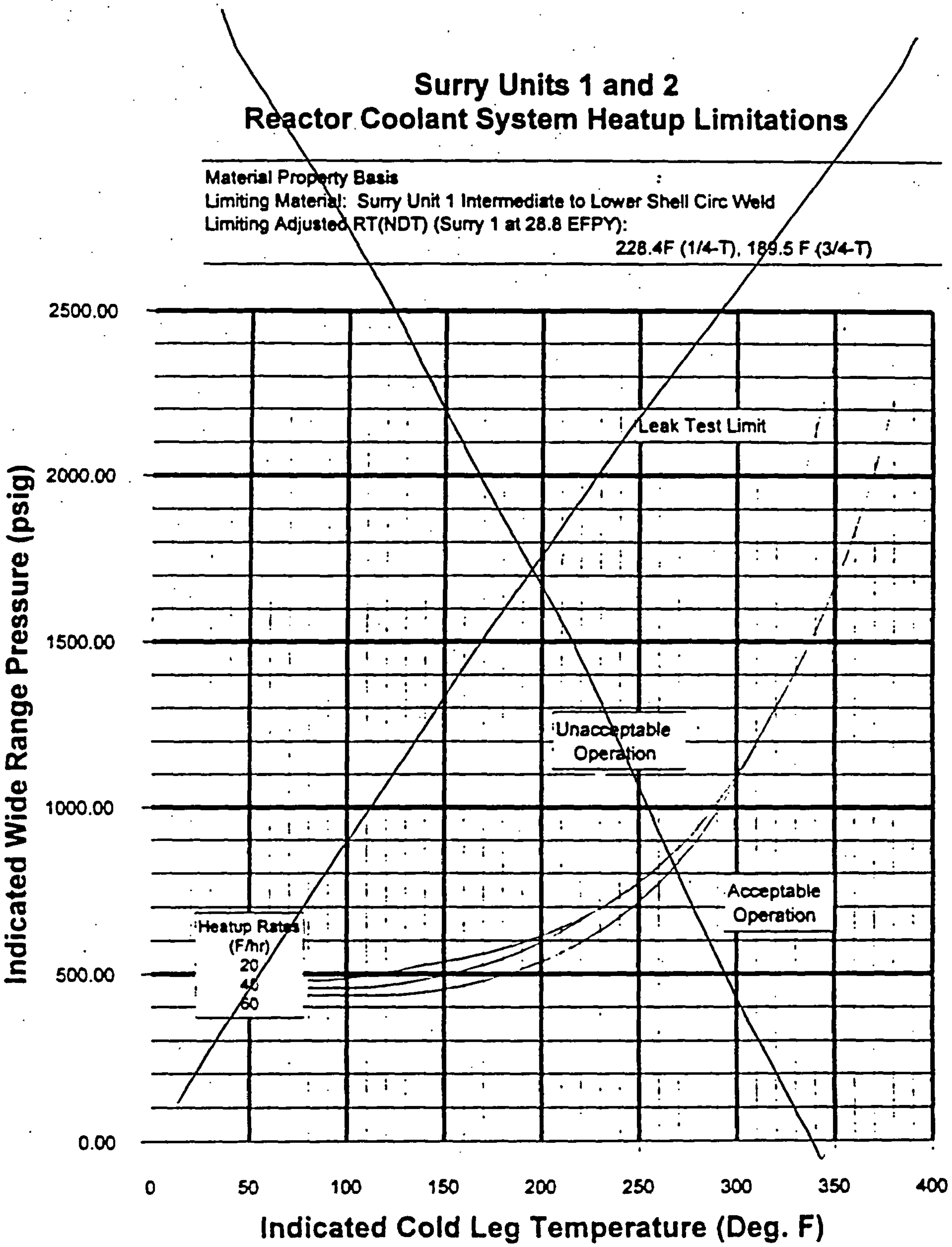
(b) with the RCS vented through a locked open vent path verify the path is open at least once per 31 days.

2. The requirements of Specification 3.1.G.1.c.(4) may be modified as follows:

a One PORV may be inoperable in INTERMEDIATE SHUTDOWN with the RCS average temperature > 200°F but < 350°F for a period not to exceed 7 days. If the inoperable PORV is not restored to OPERABLE status within 7 days, then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within the next 8 hours.

b One PORV may be inoperable in COLD SHUTDOWN or REFUELING SHUTDOWN with the reactor vessel head bolted for a period not to exceed 24 hours. If the inoperable PORV is not restored to OPERABLE status within 24 hours then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within 8 hours.

Replace w/ new Fig 3.1-1



Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

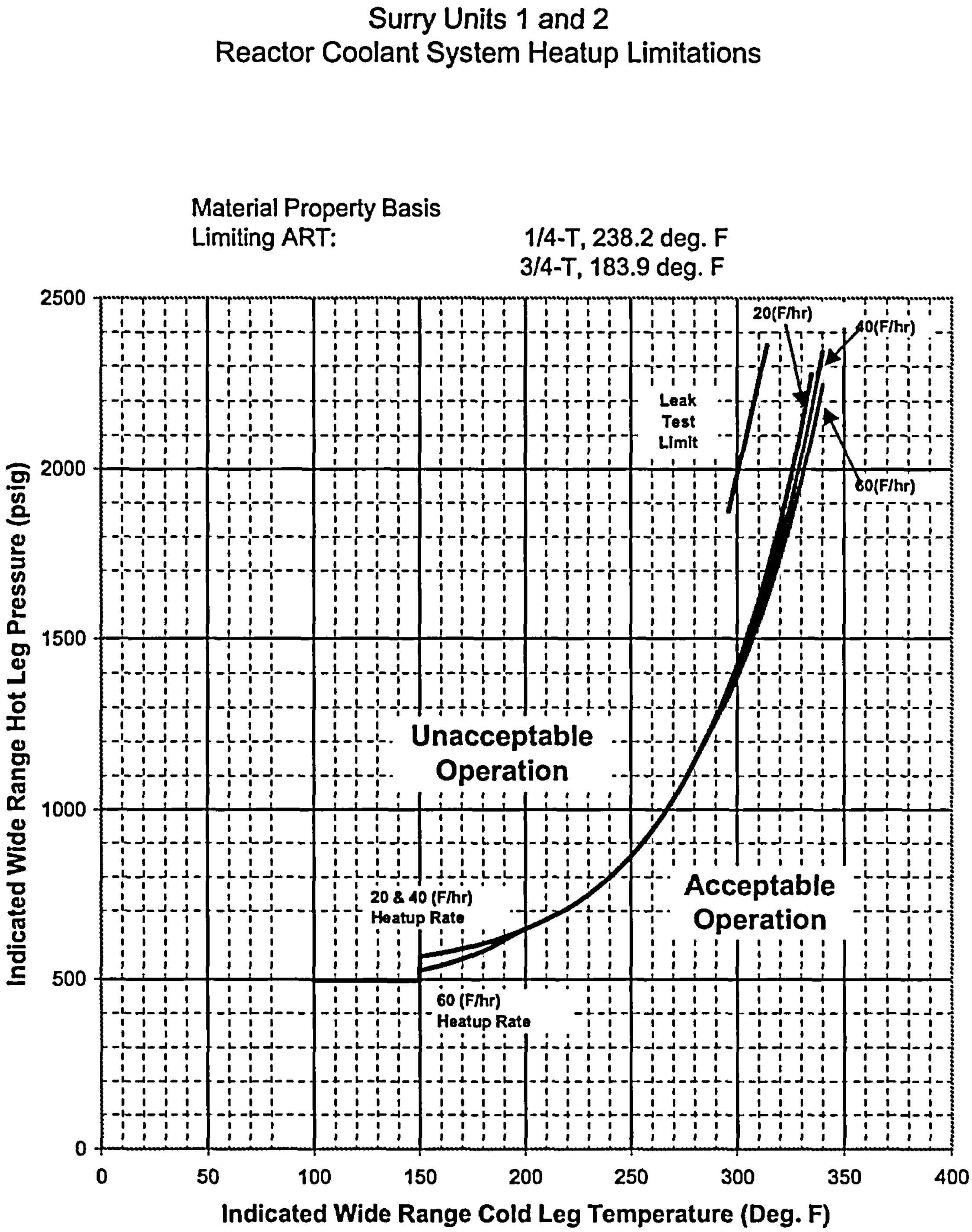
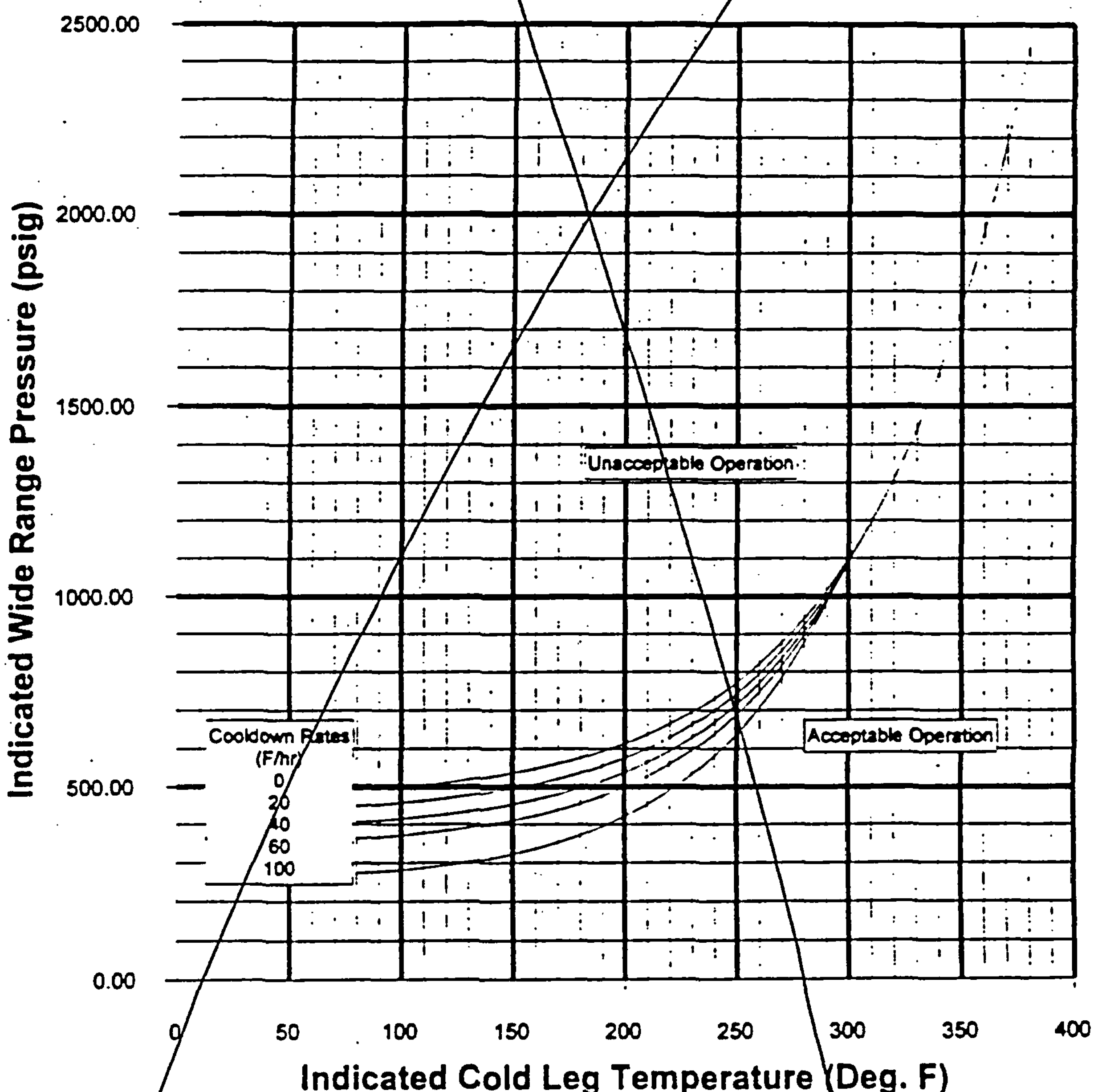


Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

Replace w/ new
Fig 3.1-2

Surry Units 1 and 2
Reactor Coolant System Cooldown Limitations

Material Property Basis
Limiting Material: Surry Unit 1 Intermediate to Lower Shell Circ Weld
Limiting Adjusted RT(NDT) (Surry 1 at 28.8 EFPY):
228.4 F (1/4-T), 189.5 F (3/4-T)



Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

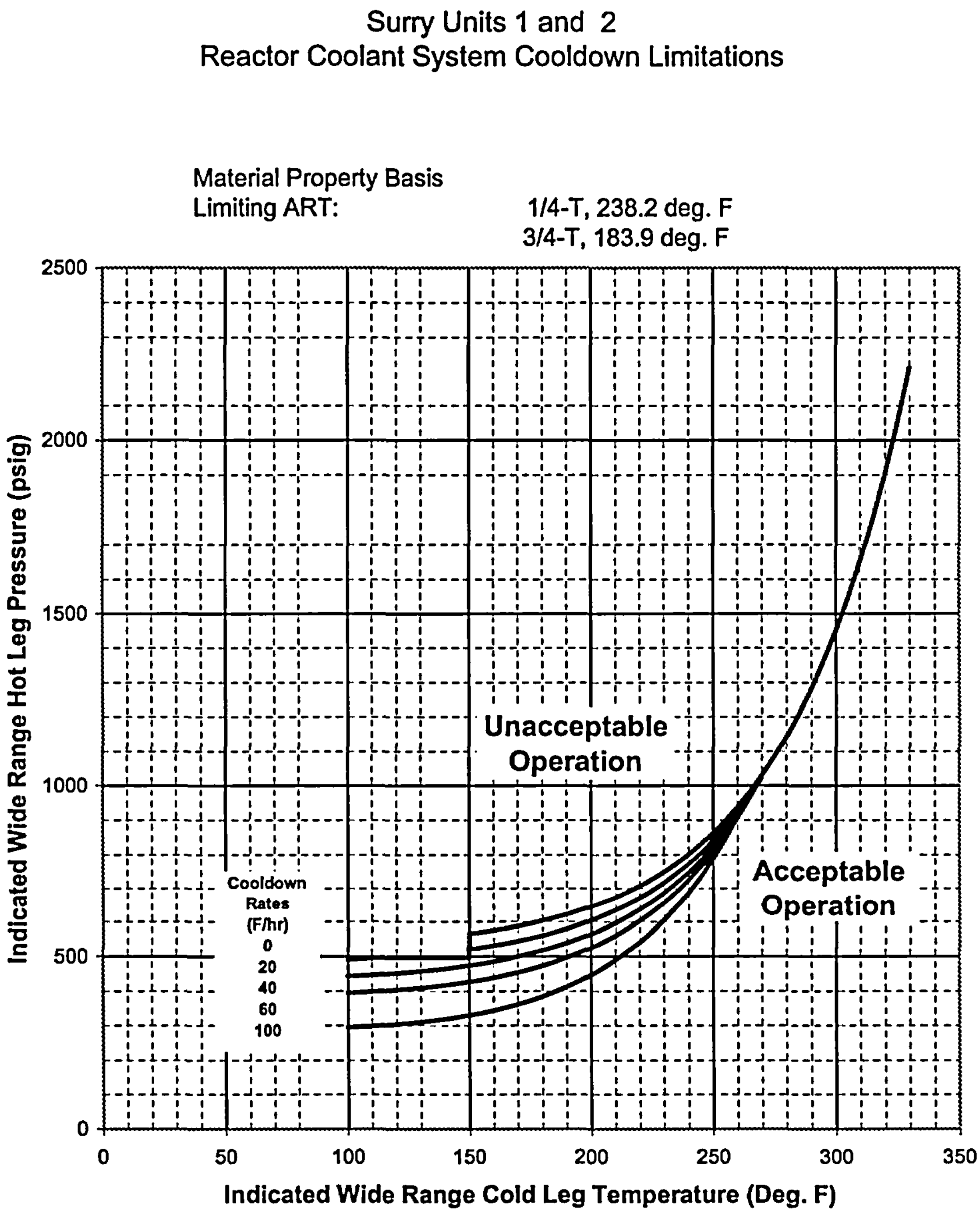


Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

Serial No. 04-755
Docket Nos. 50-280, 281

ATTACHMENT 3

Proposed TS Pages

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

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$$K_{IC} = 33.2 + 20.734 \exp [0.02(T - RT_{NDT})] \qquad (1)$$

where K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

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(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of ≤ 395 psig and verify each PORV block valve is open at least once per 72 hours,

or

(5) The RCS shall be vented through one open PORV or an equivalent size opening as specified below:

(a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or

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2. The requirements of Specification 3.1.G.1.c.(4) may be modified as follows:

a One PORV may be inoperable in INTERMEDIATE SHUTDOWN with the RCS average temperature > 200°F but < 350°F for a period not to exceed 7 days. If the inoperable PORV is not restored to OPERABLE status within 7 days, then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within the next 8 hours.

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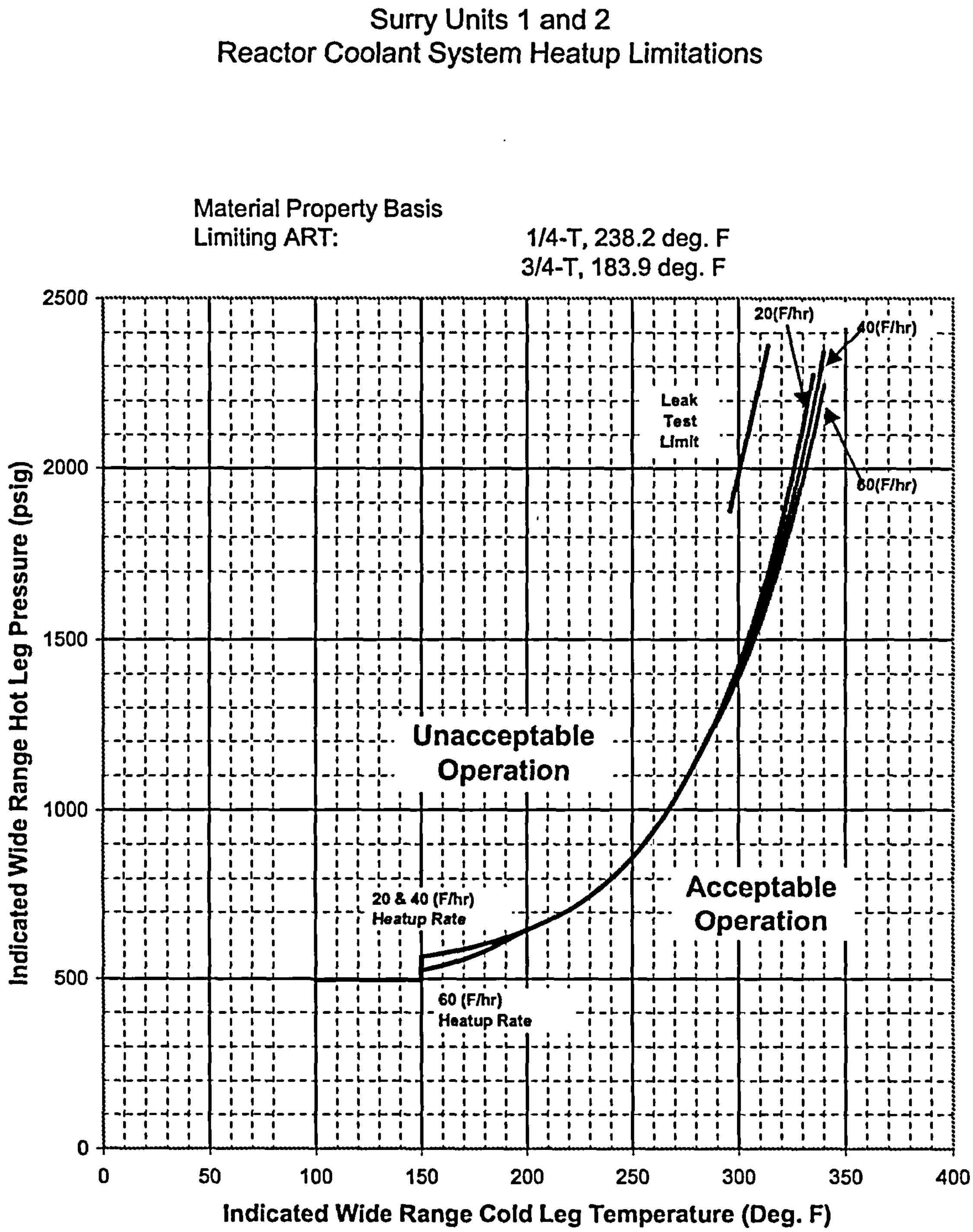

Figure 3.1-1 : Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

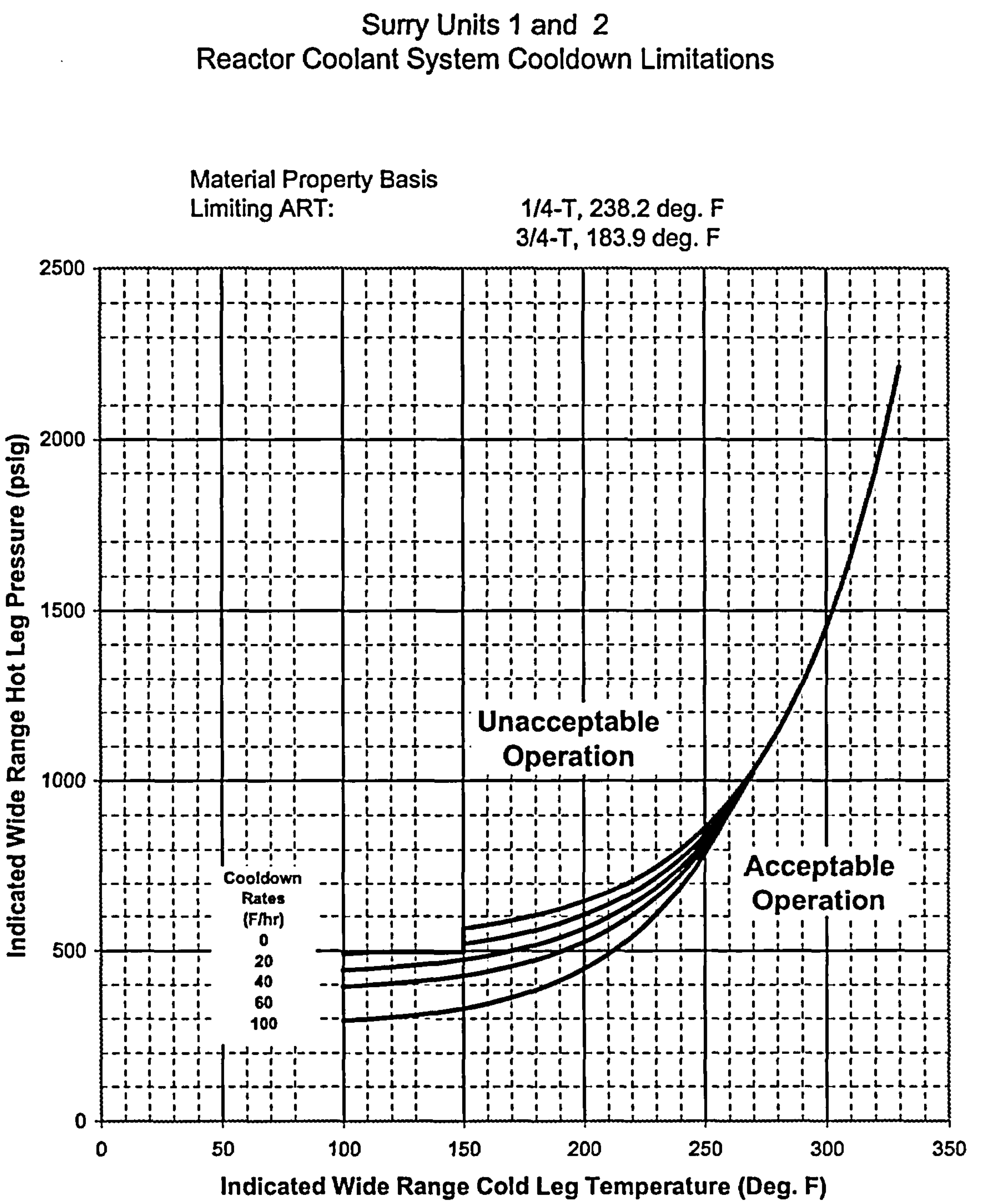

Figure 3.1-2 : Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the first 47.6 EFPY for Unit 1 and 48.1 EFPY for Unit 2 (Including Margins for Instrumentation Errors)

ATTACHMENT 4

Regulatory Basis And Request For Exemption

Surry Power Station
Units 1 and 2
Virginia Electric and Power Company
(Dominion)

REGULATORY BASIS AND REQUEST FOR EXEMPTION

Virginia Electric and Power Company (Dominion) requests modification of the Surry Units 1 and 2 reactor vessel beltline material initial properties basis for the Linde 80 weld heat materials to reflect Topical Report BAW-2308 Revision 1 (Reference 1). The proposed material initial properties basis utilize ASME Code Case N-629, which supports use of a conservative but less restrictive model for the determination of initial material properties. The proposed material initial properties basis could be used in the future by Dominion to make various plant safety improvements (e.g., reduced probability of undesired PORV lifts during reactor coolant pump startups). Please note that the acceptance of this exemption is not required for approval of the proposed Technical Specifications change requested in this submittal as the proposed TS change is supported by the current material properties basis.

In support of the proposed alternate material properties basis for Surry Units 1 and 2, exemptions are hereby being requested to 10 CFR 50.61, and 10 CFR 50 Appendix G, which specifically refer to ASME Code paragraph NB-2331 as the method for determination of initial (i.e., unirradiated) RT_{NDT}. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that: 1) the exemption is authorized by law, 2) the exemption will not result in an undue risk to public health and safety, 3) the exemption is consistent with the common defense and security, and 4) special circumstances, as defined in 10 CFR 50.12(a)(2) are present. The requested exemption to allow the use of BAW-2308 Revision 1 as the basis for the Linde 80 weld heat material initial properties at Surry Units 1 and 2 satisfy these requirements as described below.

1. The requested exemption is authorized by law.

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternatives to 10 CFR 50, Appendix G when an exemption is granted by the Commission under 10 CFR 50.12. In addition, 10 CFR 50.61 permits other methods for use in determining the initial material properties provided such methods are approved by the Director, Office of Nuclear Reactor Regulation.

2. The requested exemption does not present an undue risk to the public health and safety.

   The proposed material initial properties basis utilizes Reference 1 and ASME Code Case N-629, which supports use of a conservative but less restrictive model for the determination of initial material properties. In addition, Reference 1 contains additional conservatisms to ensure that use of the proposed initial material properties basis does not increase the probability of occurrence or the consequences of an accident at Surry Units 1 and 2, and will not create the possibility for a new or different type of accident that could pose a risk to public health and safety. In addition, Dominion will employ NRC approved methods for any future application of the margin arising from the proposed initial material properties basis (e.g., revised RCS P/T Limits, LTOPS PORV setpoints, etc). Such applications would be submitted for NRC review and approval.

3. <u>The requested exemption will not endanger the common defense and security.</u>

   The use of the proposed initial material properties from Reference 1 will not adversely affect the operation of Surry Power Station or endanger the common defense and security.

4. <u>Special circumstances are present which necessitate the request for an exemption to the regulations of 10 CFR 50.44 and 10 CFR 50.46.</u>

   Pursuant to 10 CFR 50.12(a)(2), the NRC will not consider granting an exemption to the regulations unless special circumstances are present. The requested exemptions meet the special circumstances of paragraph (a)(2)(ii) in that application of these regulations in this particular circumstance is not necessary to achieve the underlying purpose of the regulations.

   The underlying purpose of 10 CFR 50.61 and 10 CFR 50 Appendix G is to protect the integrity of the reactor coolant pressure boundary. Application of paragraph NB-2331 of ASME Section III in the determination of initial material properties was conservatively developed based on the level of knowledge existing in 1974 concerning reactor pressure vessel materials and the estimated effects of operation. Since 1974, the level of knowledge about these topics has been greatly expanded. This increased knowledge permits relaxation of the ASME III NB-2331 requirements via application of Reference 1, while maintaining the underlying purpose of the ASME Code and the NRC regulations to ensure an acceptable margin of safety.

   Therefore, the intent of 10 CFR 50.61 and 10 CFR 50 Appendix G (i.e., protection of the integrity of the reactor coolant pressure boundary) will continue to be satisfied for the proposed change in reactor vessel material initial properties basis. Issuance of an exemption from the criteria of these regulations for the use of Reference 1 in Surry Units 1 and 2 will not compromise the safe operation of the reactors.

Reference 1: BAW-2308, "Initial RTN~T of Linde 80 Weld Materials," Revision 1, dated August 2003.