Attachment 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

WCAP-14637 Revision 3

Prairie Island Unit 2 Heatup and Cooldown Limit Curves Normal Operation

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14637, Revision 3

PRAIRIE ISLAND UNIT 2 HEATUP AND COOLDOWN LIMIT CURVES FOR NORMAL OPERATION

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PREFACE

The following revisions have been made to this report:

Revision 3:

This revision was completed to change the initial RT_{NDT} of the Intermediate Shell Forging and Lower Shell Forging to 14°F and -4°F respectively. This change is a result of a customer request. Only Page 13 is affected by this updated information.

Revision 2:

- Added Executive Summary.
- Revised Table 1 and added new Table 2 to include upper shell Cu and Ni content data.
- Renumbered Tables 2 through 9 to be 3 through 10.
- Revised Tables 3, 4, 5, 7 and 8 to include upper shell and update other material data and fluences, and revise calculations.
- Revised margin discussion in Section 4 to include provisions for surveillance capsule data which is not credible.
- Revised Figure 1 and 2 and Tables 9 and 10 to include new heatup and cooldown curves and data listing.

Revision 1:

- Revised Tables 4, 6 and 7 per updated fluences given in reference 5.
- Added Table 9.
- Deleted Executive Summary.

Verified By: Ed Terek

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EXECUTIVE SUMMARY

The pressure-temperature limit curves for the Prairie Island Unit 2 plant heatup and cooldown were calculated using the adjusted RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting beltline region material in the core region of the reactor vessel. The upper shell forging and the upper to intermediate shell weld seam were considered beltline materials along with the intermediate and lower shell forgings and the intermediate to lower shell weld seam because the upper shell extends below the top of the core.

The RT_{NDT} values for the materials were adjusted for the effects of exposure to fast-neutron radiation in accordance with NRC Regulatory Guide 1.99, Revision 2. The results of the Prairie Island Unit 2 Surveillance Capsule P analysis were included in the calculation of the Adjusted Reference Temperature (ART) values (initial RT_{NDT} + RT_{NDT} shift due to neutron exposure) at the 1/4T and 3/4T locations. The most limiting ART values were used in the generation of heatup and cooldown pressure-temperature limit curves at 35 EFPY (EOL). The upper to intermediate shell weld seam was found to be the limiting beltline material. See Figures 1 and 2 of this report for the curves which were generated.

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* Includes Vessel flange requirements of 98 °F and 621 psig per 10CFR50, Appendix G.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) 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. 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 the heatup and cooldown pressure-temperature limit curves.

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan[2]. The beltline material copper and nickel content values for the Prairie Island Unit 2 reactor vessel presented in Tables 1 and 2 are from References 3 through 7 and 12 through 17.

The average Cu and Ni values were used to calculate chemistry factor (CF) values per Tables 1 and 2 of Regulatory Guide 1.99, Revision 2. (See Table 3.) Additionally, surveillance capsule data is available for four capsules (Capsules V, T, R, and P) already removed from the Prairie Island Unit 2 reactor vessel. This surveillance capsule data was used to calculate chemistry factor (CF) values (Table 4) in addition to those calculated per Tables 1 and 2 of Regulatory Guide 1.99, Revision 2.

Table 1 Calculation of Average Cu and Ni Weight Percent Values for Beltline Base Materials

Ref.	Intermediate Shell Forging 22829		Lower Shell Forging 22642[a]		Upper Shell Forging 22231/39088[b]	
	Cu %	Ni %	Cu %	Ni %	Cu %	Ni %
3			0.085	0.70		
4			0.068	0.694		
5			0.068	0.575		
6	0.075	0.75				
6	0.070	0.75				
7			0.085	0.700		
7			0.085	0.700		
12					0.065	0.73
12					0.065	0.73
Avg.	0.0725	0.75	0.0782	0.6738	0.065	0.73

NOTES:

(a) Surveillance program base metal material.

(b) The upper shell forging is being included here since it is 6.833 inches below the top of the active fuel stack.

Table 2 Calculation of the Average Cu and Ni Weight Percent Values for the Prairie Island Unit 2 Weld Materials						
Ref.	Surveillance Program Weld Metal[(a)]		Inter. to Lower Shell Weld Seam W3[(a)]		Upper to Inter. Shell Weld Seam W2[(b)]	
	Cu %	Ni %	Cu %	Ni %	Cu %	Ni %
3	0.082	0.072				
4	0.076	0.071				
5	0.094	0.103				
5	0.081	0.087				
5	0.078	0.081				
13			0.090	0.130		
			0.0822[(c)]	0.0828[(c)]		
14					0.14	0.14
15					0.14	0.17
16					0.105	0.11
17					0.14[(d)]	0.11[(d)]
Average	0.0822	0.0828	0.0861	0.1064	0.13125	0.1325

NOTES:
(a) The Surveillance Program weld metal is identical to the intermediate to lower shell girth weld seam W3 and was fabricated with Weld Wire Type UM40, Heat No. 2721, Flux Type UM89, Lot No. 1263.
(b) Weld seam W2 is being included here, since it is 6.833 inches below the top of the active fuel stack. Weld seam W2 was fabricated with Weld Wire Type UM40, Heat No. 1752, FluxType UM89, Lot No. 1263.
(c) These values are the average of all data points from the Prairie Island Unit 2 surveillance weld which was fabricated with the same heat of weld wire.
(d) These values are the average of all data points from Reference 18 for the Prairie Island Unit 1 surveillance weld which was fabricated with the same heat of weld wire.

Table 3 Interpolation of Chemistry Factors From Regulatory Guide 1.99, Revision 2, Position 1.1		
Material	**Ni, wt %**	**Chemistry Factor, °F**
Upper Shell Forging 22231/39088 Given Cu wt% = 0.07	0.73	44
Intermediate Shell Forging 22829 Given Cu wt% = 0.07	0.75	44
Lower Shell Forging 22642 GivenCu wt% = 0.08	0.67	51
Upper to Intermediate Shell Girth Weld Given Cu wt% = 0.13	0.13	69.7
Intermediate to Lower Shell Girth Weld Given Cu wt% = 0.09	0.11	51.6
Surveillance Weld (Unit 2) Given Cu wt% = 0.0822	0.08	44.8
Surveillance Weld (Unit 1) Given Cu wt% = 0.14	0.11	70.9

Table 4 Calculation of Chemistry Factors Using Surveillance Capsule Data Per Regulatory Guide 1.99, Revision 2, Position 2.1

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Forging 22642 (Axial Orientation)	V	0.6206	0.866	35.28	30.55	0.75
	T	1.199	1.051	29.93	31.46	1.10
	R	4.376	1.375	84.73	116.50	1.89
	P	4.165	1.365	103.87	141.78	1.86
Lower Shell Forging 22642 (Tangential Orientation)	V	0.6206	0.866	32.89	28.48	0.75
	T	1.199	1.051	55.69	58.53	1.10
	R	4.376	1.375	90.02	123.78	1.89
	P	4.165	1.365	99.91	136.38	1.86
				SUM	667.46	11.2
	$CF_{\text{Lower Shell Forging}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = 667.46 ÷ 11.2 = **59.6 °F**					
Weld Metal	V	0.6206	0.866	80.58[d]	69.78	0.75
	T	1.199	1.051	66.39[d]	69.78	1.10
	R	4.376	1.375	115.36[d]	158.62	1.89
	P	4.165	1.365	110.68[d]	151.08	1.86
				SUM	449.26	5.6
	$CF_{\text{Weld Metal}} = \Sigma(FF * \Delta RT_{NDT}) \div \Sigma(FF^2)$ = 449.26 ÷ 5.6 = **80.2°F**					

NOTES:

(a) f = fluence (10^{19} n/cm^2, E > 1.0 MeV). All updated fluence values were taken from the Capsule P analysis. WCAP-14613[5].

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$

(c) The ΔRT_{NDT} values are measured values obtained from the Capsule P analysis[5]

(d) The measured ΔRT_{NDT} values obtained from the Capsule P analysis[5] were multiplied by a ratio factor of 1.15. ($CF_{Vessel} \div CF_{S/C\ Weld}$ = 51.6 ÷ 44.8 = 1.15).

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[8] 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"[9], *Vessels*, 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:

$$K_{Ia} = 26.78 + 1.223 * e^{[0.0145(T-RT_{NDT}+160)]} \qquad (1)$$

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:

$$C * K_{Im} + K_{It} < K_{Ia} \qquad (2)$$

where,

K_{Im}	=	stress intensity factor caused by membrane (pressure) stress
K_{It}	=	stress intensity factor caused by the thermal gradients
K_{Ia}	=	function of temperature relative to the RT_{NDT} of the material
C	=	2.0 for Level A and Level B service limits
C	=	1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{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 ΔT (temperature) developed during cooldown results in a higher value of K_{Ia} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ia} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures 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 (3106 psig), which is 621 psig for Prairie Island Unit 2.

The limiting unirradiated RT_{NDT} of -22°F occurs in the vessel flange of the Prairie Island Unit 2 reactor vessel, so the minimum allowable temperature of this region is 98°F at pressures greater than 621 psig. This limit (where the horizontal line indicates that the pressure shall not exceed 621 psig for temperatures less than 98°F) is shown as a notch in the curves, presented wherever applicable in Figures 1 and 2.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

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

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \tag{3}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[10]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

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

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

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \tag{5}$$

where x inches (vessel beltline thickness is 6.692 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 4 to calculate the ΔRT_{NDT} at the specific depth. The fluence (E > 1.0 MeV) values on the pressure vessel clad/base metal interface for the Prairie Island Unit 2 reactor vessel are presented in Table 5.

Table 5 Peak Fluence (10^{19} n/cm^2, E > 1.0 MeV) on the Pressure Vessel Clad/Base Metal Interface for Prairie Island Unit 2 (See Ref. 5)	
EFPY	**0°**
Best Estimate @ Core Mid-Plane	
17.24	2.44
24	3.11
32	3.89
35	4.18
Best Estimate @ Top of Core	
17.24	1.39
24	1.76
32	2.21
35	2.38

The chemistry factor values obtained from Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, were determined in Table 3 using the copper and nickel content values reported in Tables 1 and 2 of this report. Chemistry factors were also calculated using surveillance capsule data as shown in Table 4.

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings, and 8.5°F for plates or forgings (half the value) when credible surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. σ_Δ need not exceed one-half the mean value of ΔRT_{NDT}. See Table 6. In the case of Prairie Island Unit No. 2, the surveillance capsule forging material data is deemed to be **not** credible in accordance with the Regulatory Guide 1.99, Revision 2[5] credibility criteria. Since data points for all of the calculated ΔRT_{NDT} values do not fall within the prescribed margins for the surveillance capsule data, NSP has chosen to use the full σ_Δ margin of 17°F for forgings. In the case of the Prairie Island Unit No. 1 surveillance capsule weld material, which is used to evaluate Unit No. 2 upper to intermediate shell weld seam W2, the data is also deemed **not** credible in accordance with the Regulatory Guide 1.99, Revision 2 credibility criteria[18], and the full σ_Δ margin of 28°F for welds is used. The Prairie Island Unit 2 surveillance weld data is credible[5]. Hence, for the intermediate to lower shell weld seam W3, a σ_Δ margin of 14°F was used.

Table 6 Margins for Adjusted Reference Temperature (ART) Calculations per Regulatory Guide 1.99, Revision 2

Material Properties	Surv. Capsule Data NOT Used	Surv. Capsule Data Used
PLATES or FORGINGS		
Measured IRT_{NDT}	34	34[(a)]
Generic IRT_{NDT}	48	38
WELD METAL		
Measured IRT_{NDT}	56	28[(b)]
Generic IRT_{NDT}	66	44

NOTES:

(a) The Prairie Island Unit No. 2 surveillance capsule forging data is not credible. Therefore, a full σ_Δ of 17°F for forgings in used in the margin term providing a total margin of 34°F.

(b) The Prairie Island Unit No. 1 surveillance capsule weld data is not credible. Therefore, a full σ_Δ of 28°F for welds is used in the margin term for upper to intermediate shell weld seam W2 providing a total margin of 56°F. The Prairie Island Unit No. 2 surveillance capsule weld data is credible. Therefore, a ½ σ_Δ is used for weld seam W3.

All materials in the beltline region of the Prairie Island Unit 2 reactor vessel were considered in determining the limiting material. Sample calculations to determine the ART values for weld seam W2 are shown in Table 7. The resulting ART values for all beltline materials at the 1/4T and 3/4T locations are summarized in Table 8. From Table 8, it can be seen that the limiting material is Upper to Intermediate Shell Weld Seam W2. Therefore, the 1/4T and 3/4T ART values for Weld Seam W2 will be used in the generation of the heatup and cooldown curves.

Table 7	**Calculation of ART Values for the Limiting Prairie Island Unit 2 Reactor Vessel Material – Upper to Intermediate Weld Seam W2**	
Parameter		
Operating Time	35 EFPY	
Location	1/4T	3/4T
Chemistry Factor, CF (°F)	80.8	80.8
Fluence, f (10^{19} n/cm^2)[(a)]	1.59	0.71
Fluence Factor, FF	1.13	0.90
ΔRT_{NDT} = CF x FF (°F)	91.3	72.9
Initial RT_{NDT}, I (°F)	-13	-13
Margin, M (°F)	56	56
Adjusted Reference Temperature (ART), (°F)	134	116

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV) = 2.379 at 35 EFPY.

(b) The Prairie Island Unit 2 reactor vessel wall thickness is 6.692 inches at the beltline region.

Table 8 Prairie Island Unit 2 1/4T and 3/4T ART Calculations at 35 EFPY								
Material	CF	f @35 EFPY	1/4T f 3/4T f(a)	1/4T FF 3/4T FF	I (°F)	M (°F)	ΔRT_{NDT} (°F)	ART (°F)
1/4T Calculations								
Upper Shell Forging B	44.0	2.379	1.59	1.13	-13	34	49.7	71
Upper to Inter. Shell Weld W2	70.0	2.379	1.59	1.13	-13	56	79.1	122
Using Unit 1 S/C Data(b)	80.8	2.379	1.59	1.13	-13	56	91.3	**134**
Intermediate Shell Forging C	44.0	4.183	2.80	1.27	14	34	55.9	104
Inter. to Lower Shell Weld W3	52.0	4.183	2.80	1.27	-31	56	66.0	91
Using S/C Data	80.0	4.183	2.80	1.27	-31	28	101.6	99
Lower Shell Forging D	51.0	4.183	2.80	1.27	-4	34	64.8	95
Using S/C Data	60.0	4.183	2.80	1.27	-4	34	76.2	106
3/4T Calculations								
Upper Shell Forging B	44.0	2.379	0.71	0.90	-13	34	39.6	61
Upper to Inter. Shell Weld W2	70.0	2.379	0.71	0.90	-13	56	63.0	106
Using Unit 1 S/C Data(b)	80.8	2.379	0.71	0.90	-13	56	72.7	**116**
Intermediate Shell Forging C	44.0	4.183	1.25	1.06	14	34	46.6	95
Inter. to Lower Shell Weld W3	52.0	4.183	1.25	1.06	-31	55.1	55.1	79
Using S/C Data	80.0	4.183	1.25	1.06	-31	28	84.8	82
Lower Shell Forging D	51.0	4.183	1.25	1.06	-4	34	54.1	84
Using S/C Data	60.0	4.183	1.25	1.06	-4	34	63.6	94

NOTE:

(a) Fluence values are x 10^{19} n/cm^2 (E > 1.0 MeV). In addition, the values used are the Best-Estimate values since thay are higher than the calculated values.

(b) This calculation is using the chemistry factor based on the surveillance capsule data from the Prairie Island Unit 1 surveillance program. Per WCAP-14779 Rev. 1, the surveillance weld data is not credible, therefore, a full σ_Δ of 28°F was used in the margin term.

(c) When two or more credible surveillance data sets become available, the data sets may be used to determine ART values as described in Reg. Guide 1.99, Rev. 2, Pos.2.1. If the ART values based on surveillance capsule data are larger than those calculated per Reg. Guide 1.99, Rev. 2, Pos. 1.1, the surveillance data must be used. If the surveillance capsule data gives lower values, either may be used. In the case of Prairie Island Unit 2, where the surveillance data for the forging material is not credible, and the Unit 1 surveillance weld data used to evaluate the upper to intermediate shell weld seam is not credible, the ART values based on the surveillance capsule data with the full margins included are larger. Therefore, the ART values based on the surveillance capsule data were used.)

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

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[11] discussed in Section 3 and 4 of this report. Since indication of reactor vessel beltline pressure is not available directly to the operator, the pressure difference between the wide-range pressure transmitter and the limiting beltline region must be accounted for when using the pressure-temperature limits presented in Figures 1 and 2.

Figure 1 presents the heatup curves without margins for possible instrumentation errors using heatup rates up to 100°F/hr applicable for the first 35 EFPY. Figure 2 presents the cooldown curves without margins for possible instrumentation errors using cooldown rates up to 100°F/hr applicable for 35 EFPY. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 1 and 2. This is in addition to other criteria which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 1. 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 equation for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code as follows:

$$1.5\,K_{Im} < K_{Ia} \qquad (6)$$

where,
K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ia} = 26.78 + 1.223\, e^{[0.0145\,(T - RTNDT + 160)]}$,
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 10. 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 minimum temperatures for the inservice hydrostatic leak tests for the Prairie Island Unit 1 reactor vessel at 35 EFPY is 246°F. 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.

Figures 1 and 2 define all of the above limits for ensuring prevention of nonductile failure for the Prairie Island Unit 2 reactor vessel.

The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 and 2 are presented in Tables 9 and 10.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: UPPER TO INTERMEDIATE SHELL WELD SEAM W2

LIMITING ART VALUES AT 35 EFPY: 1/4T, 134°F

3/4T, 116°F

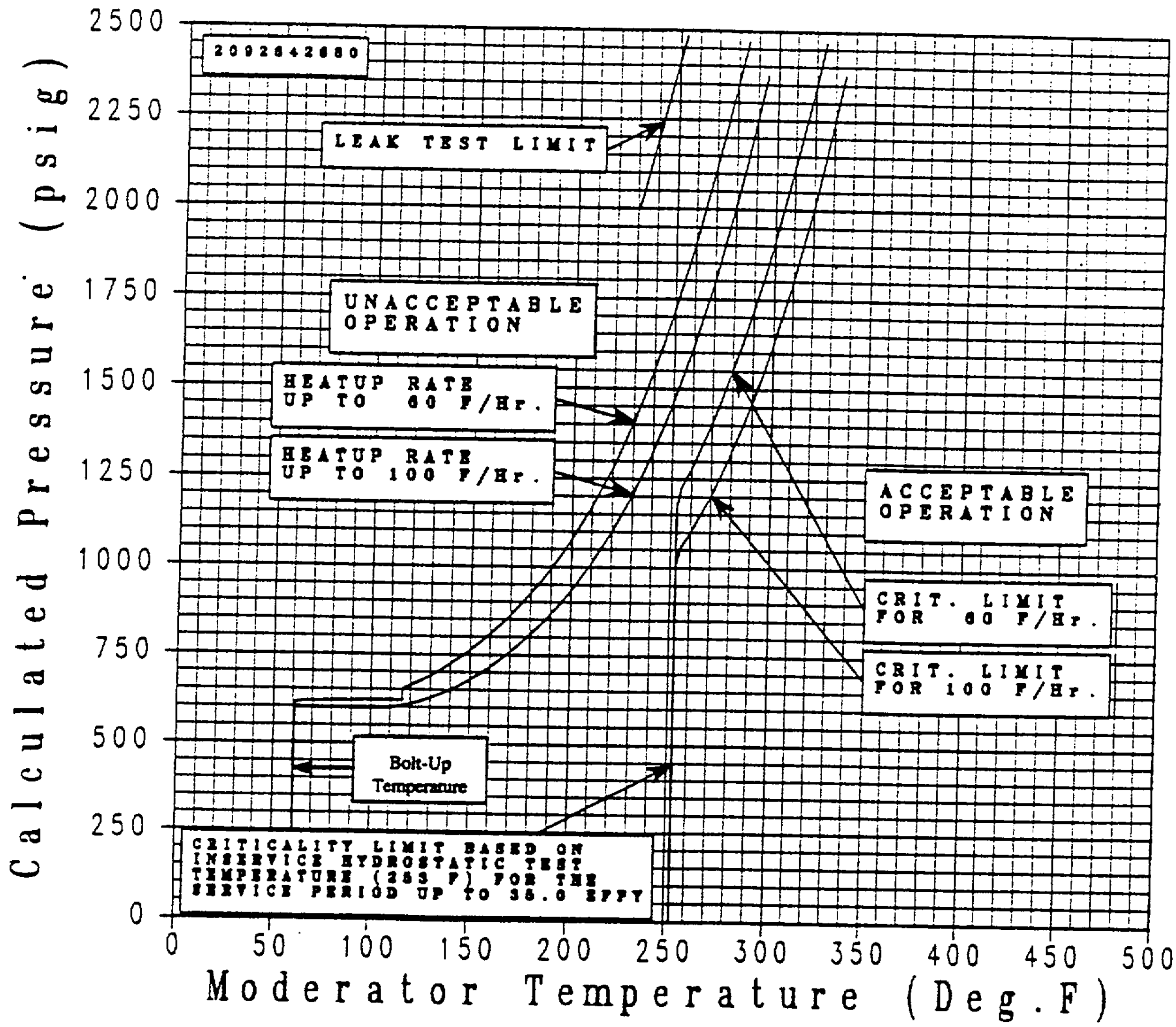

FIGURE 1 Prairie Island Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 100°F/hr) Applicable for the First 35 EFPY (Without Margins for Instrumentation Errors)

Includes Vessel flange requirements of 98°F and 621 psig per 10CFR50, Appendix G.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: UPPER TO INTERMEDIATE SHELL WELD SEAM W2
LIMITING ART VALUES AT 35 EFPY: 1/4T, 134°F
3/4T, 116°F

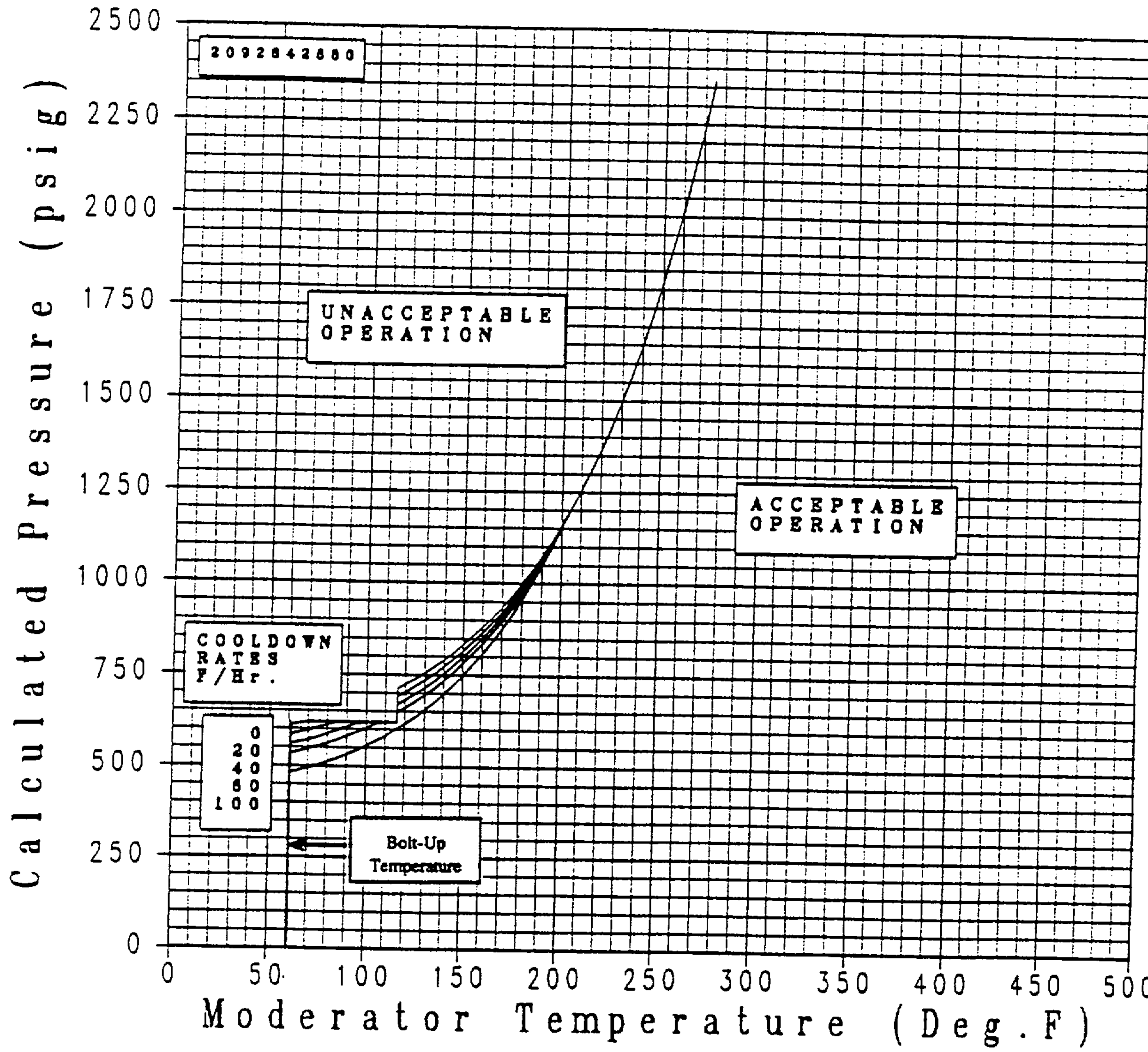

FIGURE 2 Prairie Island Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 35 EFPY (Without Margins for Instrumentation Errors)

Includes Vessel flange requirements of 98°F and 621 psig per 10CFR50, Appendix G.

TABLE 9
35 EFPY Heatup Curve Data Points
(Without Instrumentation Error Margins)

Heatup Curves									
60 Heatup		Critical. Limit		100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P	T	P	T	P
60.0	0.0	253.0	0.0	60.0	0.0	253.0	0.0	231.0	2000.0
60.0	610.4	253.0	616.7	60.0	597.9	253.0	597.9	253.0	2485.0
65.0	616.7	253.0	620.7	65.0	597.9	253.0	597.9		
85.0	618.9	253.0	618.9	85.0	597.9	253.0	597.9		
90.0	618.9	253.0	620.8	90.0	597.9	253.0	597.9		
95.0	620.8	253.0	625.3	95.0	597.9	253.0	597.9		
100.0	621.0	253.0	632.2	100.0	597.9	253.0	598.2		
105.0	621.0	253.0	640.9	105.0	598.2	253.0	600.5		
110.0	621.0	253.0	651.3	110.0	600.5	253.0	604.8		
115.0	621.0	253.0	663.1	115.0	604.8	253.0	610.9		
116.0	621.0	253.0	676.5	120.0	610.9	253.0	618.8		
116.0	653.7	253.0	691.2	125.0	618.8	253.0	628.1		
120.0	663.1	253.0	707.2	130.0	628.1	253.0	639.1		
125.0	676.5	253.0	724.8	135.0	639.1	253.0	651.6		
130.0	691.2	253.0	743.9	140.0	651.6	253.0	665.5		
135.0	707.2	253.0	764.3	145.0	665.5	253.0	681.0		
140.0	724.8	253.0	786.6	150.0	681.0	253.0	698.1		
145.0	743.9	253.0	810.3	155.0	698.1	253.0	716.6		
150.0	764.3	253.0	836.2	160.0	716.6	253.0	737.1		
155.0	786.6	253.0	863.7	165.0	737.1	253.0	759.0		
160.0	810.3	253.0	893.6	170.0	759.0	253.0	783.1		
165.0	836.2	253.0	925.6	175.0	783.1	253.0	808.8		
170.0	863.7	253.0	959.8	180.0	808.8	253.0	836.9		
175.0	893.6	253.0	996.7	185.0	836.9	253.0	867.0		
180.0	925.6	253.0	1036.5	190.0	867.0	253.0	899.4		
185.0	959.8	253.0	1079.1	195.0	899.4	253.0	934.5		
190.0	996.7	253.0	1124.8	200.0	934.5	253.0	972.1		
195.0	1036.5	253.0	1173.8	205.0	972.1	253.0	1012.4		
200.0	1079.1	255.0	1226.5	210.0	1012.4	255.0	1055.9		
205.0	1124.8	260.0	1283.0	215.0	1055.9	260.0	1102.5		
210.0	1173.8	265.0	1343.5	220.0	1102.5	265.0	1152.6		
215.0	1226.5	270.0	1408.7	225.0	1152.6	270.0	1206.5		
220.0	1283.0	275.0	1478.3	230.0	1206.5	275.0	1264.2		
225.0	1343.5	280.0	1553.1	235.0	1264.2	280.0	1326.1		
230.0	1408.7	285.0	1633.0	240.0	1326.1	285.0	1392.5		
235.0	1478.3	290.0	1718.9	245.0	1392.5	290.0	1463.7		
240.0	1553.1	295.0	1810.7	250.0	1463.7	295.0	1540.0		
245.0	1633.0	300.0	1908.9	255.0	1540.0	300.0	1621.7		
250.0	1718.9	305.0	2013.9	260.0	1621.7	305.0	1709.2		
255.0	1810.7	310.0	2123.0	265.0	1709.2	310.0	1802.8		
260.0	1908.9	315.0	2231.9	270.0	1802.8	315.0	1902.9		
265.0	2013.9	320.0	2347.8	275.0	1902.9	320.0	2009.8		
270.0	2123.0	325.0	2472.0	280.0	2009.8	325.0	2124.4		
275.0	2231.9			285.0	2124.4	330.0	2246.7		
280.0	2347.8			290.0	2246.7	335.0	2376.8		
285.0	2472.0			295.0	2376.8				

TABLE 10
35 EFPY Cooldown Curve Data Points
(Without Instrumentation Error Margins)

Cooldown Curves									
Steady State		20F		40F		60F		100F	
T	P	T	P	T	P	T	P	T	P
60.0	0.0	60.0	0.0	60.0	0.0	60.0	0.0	60.0	0.0
60.0	610.4	60.0	585.0	60.0	559.3	60.0	533.4	60.0	480.5
65.0	616.7	65.0	591.5	65.0	565.9	65.0	540.2	65.0	487.7
70.0	621.0	70.0	598.4	70.0	573.1	70.0	547.5	70.0	495.5
75.0	621.0	75.0	605.8	75.0	580.7	75.0	555.3	75.0	503.9
80.0	621.0	80.0	613.8	80.0	589.0	80.0	563.8	80.0	512.9
85.0	621.0	85.0	621.0	85.0	597.9	85.0	573.0	85.0	522.7
90.0	621.0	90.0	621.0	90.0	607.4	90.0	582.9	90.0	533.3
95.0	621.0	95.0	621.0	95.0	617.7	95.0	593.7	95.0	544.9
100.0	621.0	100.0	621.0	100.0	621.0	100.0	605.1	100.0	557.1
105.0	621.0	105.0	621.0	105.0	621.0	105.0	617.5	105.0	570.6
110.0	621.0	110.0	621.0	110.0	621.0	110.0	621.0	110.0	585.0
115.0	621.0	115.0	621.0	115.0	621.0	115.0	621.0	115.0	600.7
116.0	621.0	116.0	621.0	116.0	621.0	116.0	621.0	120.0	617.4
116.0	714.9	116.0	692.8	116.0	670.6	116.0	648.5	125.0	635.6
120.0	726.0	120.0	704.2	120.0	682.6	120.0	660.8	130.0	655.0
125.0	741.0	125.0	719.8	125.0	698.7	125.0	677.6	135.0	676.2
130.0	756.9	130.0	736.5	130.0	716.0	130.0	695.7	140.0	699.0
135.0	774.3	135.0	754.3	135.0	734.7	135.0	715.0	145.0	723.5
140.0	792.8	140.0	773.7	140.0	754.6	140.0	736.0	150.0	749.9
145.0	812.7	145.0	794.4	145.0	776.3	145.0	758.4	155.0	778.4
150.0	834.2	150.0	816.7	150.0	799.4	150.0	782.7	160.0	808.9
155.0	857.1	155.0	840.7	155.0	824.5	155.0	808.7	165.0	842.0
160.0	882.0	160.0	866.5	160.0	851.3	160.0	836.8	170.0	877.6
165.0	908.5	165.0	894.2	165.0	880.4	165.0	867.0	175.0	915.8
170.0	937.1	170.0	924.0	170.0	911.4	170.0	899.3	180.0	957.0
175.0	967.8	175.0	956.0	175.0	944.9	175.0	934.5	185.0	1001.3
180.0	1000.7	180.0	990.5	180.0	980.9	180.0	972.1	190.0	1049.0
185.0	1036.2	185.0	1027.5	185.0	1019.6	185.0	1012.5	195.0	1100.4
190.0	1074.3	190.0	1067.3	190.0	1061.1	190.0	1055.9	200.0	1155.6
195.0	1115.2	195.0	1109.9	195.0	1105.7	195.0	1102.7		
200.0	1159.1	200.0	1155.8	200.0	1153.7	200.0	1152.9		
205.0	1206.3	205.0	1205.1	205.0	1205.4				
210.0	1256.9								
215.0	1311.2								
220.0	1369.6								
225.0	1432.0								
230.0	1499.4								
235.0	1571.5								
240.0	1648.8								
245.0	1731.5								
250.0	1820.1								
255.0	1915.1								
260.0	2017.0								
265.0	2126.0								
270.0	2242.3								
275.0	2366.7								

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