Westinghouse Non-Proprietary Class 3

WCAP-15079
Revision 1

Wolf Creek Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Energy Systems

Westinghouse Non-Proprietary Class 3

WCAP - 15079
Revision 1

Wolf Creek Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse Energy Systems

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15079, Revision 1

Wolf Creek Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

September 1998

Work Performed Under Shop Order K6TP-139

Approved: ______________________
C. H. Boyd, Manager
Equipment & Materials Technology

Approved: ______________________
D. M. Trombola, Manager
Mechanical Systems Integration

Westinghouse Electric Company
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF TABLES - (Continued)

LIST OF FIGURES

PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek *Ed Terek*

EXECUTIVE SUMMARY

The purpose of this report is to generate pressure-temperature limit curves for Wolf Creek for normal operation at 20 and 32 EFPY using the methodology from the 1989 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T location. The 1/4T and 3/4T values are summarized in Table 4-14 and were calculated using the lower shell plate R2508-3 (i.e. The limiting beltline region material). The pressure-temperature limit curves were generated for heatup rates of 60 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100°F/hr. These curves can be found in Figures 5-1 through 5-4.

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 heatup and cooldown pressure-temperature limit curves for normal operation.

2 PURPOSE

The Wolf Creek Nuclear Operating Corporation has contracted Westinghouse to analyze surveillance capsule V from the Wolf Creek reactor vessel. As a part of this analysis Westinghouse generated new heatup and cooldown curves for 20 and 32 EFPY. The heatup and cooldown curves were generated without margins 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 the Wolf Creek heatup and cooldown curves for 20 and 32 EFPY. 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

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:

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

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:

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

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 the Wolf Creek reactor vessel.

The limiting unirradiated RT_{NDT} of 20°F (Table 4-6) occurs in the closure head and vessel flange of the Wolf Creek reactor vessel, so the minimum allowable temperature of this region is 140°F at pressures greater than 621 psig with no margin for uncertainties.

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:

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

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[6]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

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

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

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

where x inches (vessel beltline thickness is 8.63 inches[5]) 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 Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections[7] and the results are presented in Section 6 of WCAP-15078. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Tables 4-1 and 4-2, herein, contain the best estimate vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Wolf Creek reactor vessel. Additionally, the measured surveillance capsule fluence values are presented in Table 4-3.

TABLE 4-1
Summary of the Peak Pressure Vessel Neutron Fluence Values
at 20 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface	¼ T	¾ T
Intermediate Shell Plate R2005-1	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate Shell Plate R2005-2	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate Shell Plate R2005-3	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Lower Shell Plate R2508-1	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Lower Shell Plate R2508-2	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Lower Shell Plate R2508-3	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate and Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	6.82×10^{18}	4.06×10^{18}	1.44×10^{18}
Intermediate and Lower Shell Longitudinal Weld Seam 101-124B,C and 101-142B,C (210° & 330°Azimuth)	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}
Intermediate to Lower Shell Cirumferential Weld Seam 101-171	1.265×10^{19}	7.54×10^{18}	2.68×10^{18}

TABLE 4-2
Summary of the Peak Pressure Vessel Neutron Fluence Values
at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface	¼ T	¾ T
Intermediate Shell Plate R2005-1	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Intermediate Shell Plate R2005-2	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Intermediate Shell Plate R2005-3	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Lower Shell Plate R2508-1	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Lower Shell Plate R2508-2	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Lower Shell Plate R2508-3	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Intermediate and Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	1.06×10^{19}	6.32×10^{18}	2.24×10^{18}
Intermediate and Lower Shell Longitudinal Weld Seam 101-124B,C and 101-142B,C (210° & 330°Azimuth)	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}
Intermediate to Lower Shell Cirumferential Weld Seam 101-171	2.00×10^{19}	1.19×10^{19}	4.23×10^{18}

TABLE 4-3

Calculated Integrated Neutron Exposure of the Wolf Creek Surveillance Capsules Tested to Date

Capsule	Fluence
U	3.429×10^{18} n/cm^2, (E > 1.0 MeV)
Y	1.308×10^{19} n/cm^2, (E > 1.0 MeV)
V	2.528×10^{19} n/cm^2, (E > 1.0 MeV)

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 used. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates when surveillance capsule data is not used and 8.5°F for plates when surveillance capsule data is used. For welds, σ_Δ is 28°F when surveillance capsule data is not used and 14°F when surveillance capsule data is used. In addition, σ_Δ need not exceed one-half the mean value of ΔRT_{NDT}.

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials[7]. These measured shift values were obtained using CVGRAPH, Version 4.1[4], which is a hyperbolic tangent curve-fitting program.

TABLE 4-4
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[a]
Lower Shell Plate R2508-3	U	36.46°F
	Y	16.03°F
(Longitudinal Orientation)	V	52.03°F
Lower Shell Plate R2508-3	U	23.79°F
	Y	35.39°F
(Transverse Orientation)	V	54.53°F
Surveillance Program Weld Metal	U	27.21°F
	Y	45.09°F
	V	46.33°F
Heat Affected Zone	U	58.41°F
	Y	12.98°F
	V	55.91°F

Notes:

(a) Calculated using measured Charpy data and plotted using CVGRAPH[4]

Table 4-5 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-5 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-7. Table 4-6 provides the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-7.

TABLE 4-5
Reactor Vessel Beltline Material Unirradiated Toughness Properties[5 & 9]

Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}[a]**
Closure Head Flange R2504-1	- -	0.66	20°F
Vessel Flange R2501-1	- -	0.70	20°F
Intermediate Shell Plate R2005-1	0.04	0.66	-20°F
Intermediate Shell Plate R2005-2	0.04	0.64	-20°F
Intermediate Shell Plate R2005-3	0.05	0.63	-20°F
Lower Shell Plate R2508-1	0.09	0.67	0°F
Lower Shell Plate R2508-2	0.06	0.64	10°F
Lower Shell Plate R2508-3	0.09	0.58	40°F
Intermediate and Lower Shell Longitudinal Weld Seams[b]	0.04	0.08	-50°F
Intermediate to Lower Shell Circumferential Weld Seam[b]	0.04	0.08	-50°F
Surveillance Program Weld Metal[b]	0.07	0.10	- - -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) All welds, including the surveillance weld, were fabricated with weld wire heat number 90146. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate shell longitudinal weld seams 101-124A, B & C and lower shell longitudinal weld seams 101-142A, B, & C were fabricated with Flux Type 0091 Lot 0842. The surveillance weld metal was fabricated with weld wire heat number 90146, Flux Type 124 Lot number 1061. Per Regulatory Guide 1.99, Revision 2, "weight percent copper " and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld." The surveillance weld metal was made with the same weld wire heat as all of the vessel beltline weld seam and is therefore representative of all of the beltline weld seams.

TABLE 4-6

Calculation of Chemistry Factors using Wolf Creek Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell	U	0.3429	0.705	36.46°F	25.70°F	0.50
Plate R2508-3	Y	1.308	1.075	16.03°F	17.23°F	1.16
(Longitudinal)	V	2.528	1.249	52.03°F	64.99°F	1.56
Intermediate Shell	U	0.3429	0.705	23.79°F	16.77°F	0.50
Plate R2508-3	Y	1.308	1.075	35.39°F	38.04°F	1.16
(Transverse)	V	2.528	1.249	54.53°F	68.11°F	1.56
				SUM	230.84°F	6.44
	$CF_{R2508\text{-}3} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (230.8) \div (6.44) = 35.8°F$					
Beltline Region	U	0.3429	0.705	19.75°F[d]	13.92°F	0.50
Weld Metal	Y	1.308	1.075	32.74°F[d]	35.20°F	1.16
	V	2.528	1.249	33.64°F[e]	42.02°F	1.56
				SUM	91.14°F	3.22
	$CF_{Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (91.14) \div (3.22) = 28.3°F$					

Notes:

(a) f = Measured fluence from capsule V dosimetry analysis results[7], (x 10^{19} n/cm^2, E > 1.0 MeV).

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

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values.

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 0.726.

TABLE 4-7

Summary of the Wolf Creek Reactor Vessel Beltline Material Chemistry Factors Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1[a]	Position 2.1[a]
Intermediate Shell Plate R2005-1	26.0°F	---
Intermediate Shell Plate R2005-2	26.0°F	---
Intermediate Shell Plate R2005-3	31.0°F	---
Lower Shell Plate R2508-1	58.0°F	---
Lower Shell Plate R2508-2	37.0°F	---
Lower Shell Plate R2508-3	58.0°F	35.8°F-
Intermediate and Lower Shell Longitudinal Weld Seam 101-124A & 101-142A[b]	31.6°F	28.3°F
Intermediate and Lower Shell Longitudinal Weld Seam 101-124B,C and 101-142B,C[b]	31.6°F	28.3°F
Intermediate to Lower Shell Cirumferential Weld Seam 101-171[b]	31.6°F	28.3°F
Surveillance Program Weld Metal[b]	43.5°F	---

Notes:

(a) Regulatory Guide 1.99, Revision 2, Position 1.1 or Position 2.1 methodology.

(b) All welds, including the surveillance weld, were fabricated with weld wire heat number 90146. The intermediate to lower shell circumferential weld seam 101-171 was fabricated with Flux Type 124 Lot Number 1061. The intermediate shell longitudinal weld seams 101-124A, B & C and lower shell longitudinal weld seams 101-142A, B, & C were fabricated with Flux Type 0091 Lot 0842. The surveillance weld metal was fabricated with weld wire heat number 90146, Flux Type 124 Lot number 1061

Contained in Tables 4-8 and 4-9 are summaries of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Wolf Creek reactor vessel beltline materials for 20 EFPY and 32 EFPY.

TABLE 4-8
Summary of the Calculated Fluence Factors used for the Generation of the
20 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm^2, E > 1.0 MeV)	¼ T FF[a]	¾ T f (n/cm^2, E >1.0 MeV)	¾ T FF[b]
Intermediate Shell Plate R2005-1	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Intermediate Shell Plate R2005-2	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Intermediate Shell Plate R2005-3	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Lower Shell Plate R2508-1	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Lower Shell Plate R2508-2	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Lower Shell Plate R2508-3	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Intermediate and Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	4.06×10^{18}	0.75	1.44×10^{18}	0.49
Intermediate and Lower Shell Longitudinal Weld Seam 101-124B,C and 101-142B,C (210° & 330°Azimuth)	7.54×10^{18}	0.92	2.68×10^{18}	0.64
Intermediate to Lower Shell Cirumferential Weld Seam 101-171	7.54×10^{18}	0.92	2.68×10^{18}	0.64

Notes:

(a) Fluence Factor at the 1/4T vessel thickness location.

(b) Fluence Factor at the 3/4T vessel thickness location.

TABLE 4-9

Summary of the Calculated Fluence Factors used for the Generation of the 32 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm², E > 1.0 MeV)	¼ T FF[(a)]	¾ T f (n/cm², E > 1.0 MeV)	¾ T FF[(b)]
Intermediate Shell Plate R2005-1	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Intermediate Shell Plate R2005-2	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Intermediate Shell Plate R2005-3	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Lower Shell Plate R2508-1	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Lower Shell Plate R2508-2	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Lower Shell Plate R2508-3	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Intermediate and Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	6.32×10^{18}	0.87	2.24×10^{18}	0.60
Intermediate and Lower Shell Longitudinal Weld Seam 101-124B,C and 101-142B,C (210° & 330°Azimuth)	1.19×10^{19}	1.05	4.23×10^{18}	0.76
Intermediate to Lower Shell Cirumferential Weld Seam 101-171	1.19×10^{19}	1.05	4.23×10^{18}	0.76

Notes:

(a) Fluence Factor at the 1/4T vessel thickness location.

(b) Fluence Factor at the 3/4T vessel thickness location.

Contained in Tables 4-10 through 4-13 are the calculations of the ART values used for the generation of the 20 EFPY and 32 EFPY heatup and cooldown curves.

TABLE 4-10

Calculation of the ART Values for the 1/4T Location @ 20 EFPY

Material	RG 1.99 R2 Method	CF	FF	IRT_{NDT}[(a)]	ΔRT_{NDT}[(c)]	Margin	ART[(b)]
Intermediate Shell Plate R2005-1	Position 1.1	26.0°F	0.92	-20°F	23.9°F	23.9°F	28°F
Intermediate Shell Plate R2005-2	Position 1.1	26.0°F	0.92	-20°F	23.9°F	23.9°F	28°F
Intermediate Shell Plate R2005-3	Position 1.1	31.0°F	0.92	-20°F	28.5°F	28.5°F	37°F
Lower Shell Plate R2508-1	Position 1.1	58.0°F	0.92	0°F	53.4°F	34°F	87°F
Lower Shell Plate R2508-2	Position 1.1	37.0°F	0.92	10°F	34.0°F	34°F	78°F
Lower Shell Plate R2508-3	Position 1.1	58.0°F	0.92	40°F	53.4°F	34°F	127°F
	Position 2.1	35.8°F	0.92	40°F	32.9°F	17°F	90°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	Position 1.1	31.6°F	0.75	-50°F	23.7°F	23.7°F	-3°F
	Position 2.1	28.3°F	0.75	-50°F	21.2°F	21.2°F	-8°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124B,C & 101-142B,C (210° & 330° Azimuth)	Position 1.1	31.6°F	0.92	-50°F	21.9°F	21.9°F	8°F
	Position 2.1	28.3°F	0.92	-50°F	26.0°F	26.0°F	2°F
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	31.6°F	0.92	-50°F	29.1°F	29.1°F	8°F
	Position 2.1	28.3°F	0.92	-50°F	26.0°F	26.0°F	2°F

Notes:

(a) Initial RT_{NDT} values are measured values (see Table 4-5).

(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(c) ΔRT_{NDT} = CF * FF

TABLE 4-11
Calculation of the ART Values for the 3/4T Location @ 20 EFPY

Material	RG 1.99 R2 Method	CF	FF	IRT_{NDT} [(a)]	ΔRT_{NDT} [(c)]	Margin	ART[(b)]
Intermediate Shell Plate R2005-1	Position 1.1	26.0°F	0.64	-20°F	16.6°F	16.6°F	13°F
Intermediate Shell Plate R2005-2	Position 1.1	26.0°F	0.64	-20°F	16.6°F	16.6°F	13°F
Intermediate Shell Plate R2005-3	Position 1.1	31.0°F	0.64	-20°F	19.8°F	19.8°F	20°F
Lower Shell Plate R2508-1	Position 1.1	58.0°F	0.64	0°F	37.1°F	34.0°F	71°F
Lower Shell Plate R2508-2	Position 1.1	37.0°F	0.64	10°F	23.7°F	23.7°F	57°F
Lower Shell Plate R2508-3	Position 1.1	58.0°F	0.64	40°F	37.1°F	34°F	111°F
	Position 2.1	35.8°F	0.64	40°F	22.9°F	17°F	80°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	Position 1.1	31.6°F	0.49	-50°F	15.5°F	15.5°F	-19°F
	Position 2.1	28.3°F	0.49	-50°F	13.9°F	13.9°F	-22°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124B,C & 101-142B,C (210° & 330° Azimuth)	Position 1.1	31.6°F	0.64	-50°F	20.2°F	20.2°F	-10°F
	Position 2.1	28.3°F	0.64	-50°F	18.1°F	18.1°F	-14°F
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	31.6°F	0.64	-50°F	20.2°F	20.2°F	-10°F
	Position 2.1	28.3°F	0.64	-50°F	18.1°F	18.1°F	-14°F

Notes:

(a) Initial RT_{NDT} values are measured values (see Table 4-5).

(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(c) ΔRT_{NDT} = CF * FF

TABLE 4-12

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT} [a]	ΔRT_{NDT} [c]	Margin	ART [b]
Intermediate Shell Plate R2005-1	Position 1.1	26.0°F	1.05	-20°F	27.3°F	27.3°F	35°F
Intermediate Shell Plate R2005-2	Position 1.1	26.0°F	1.05	-20°F	27.3°F	27.3°F	35°F
Intermediate Shell Plate R2005-3	Position 1.1	31.0°F	1.05	-20°F	32.6°F	32.6°F	45°F
Lower Shell Plate R2508-1	Position 1.1	58.0°F	1.05	0°F	60.9°F	34°F	95°F
Lower Shell Plate R2508-2	Position 1.1	37.0°F	1.05	10°F	38.9°F	34°F	83°F
Lower Shell Plate R2508-3	Position 1.1	58.0°F	1.05	40°F	60.9°F	34°F	135°F
	Position 2.1	35.8°F	1.05	40°F	37.6°F	17°F	95°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	Position 1.1	31.6°F	0.87	-50°F	27.5°F	27.5°F	5°F
	Position 2.1	28.3°F	0.87	-50°F	24.6°F	24.6°F	-1°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124B,C & 101-142B,C (210° & 330° Azimuth)	Position 1.1	31.6°F	1.05	-50°F	33.2°F	33.2°F	16°F
	Position 2.1	28.3°F	1.05	-50°F	29.7°F	28.0°F	8°F
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	31.6°F	1.05	-50°F	33.2°F	33.2°F	16°F
	Position 2.1	28.3°F	1.05	-50°F	29.7°F	28.0°F	8°F

Notes:

(a) Initial RT_{NDT} values are measured values (see Table 4-5).

(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

(c) ΔRT_{NDT} = CF * FF

TABLE 4-13

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT} [(a)]	ΔRT_{NDT} [(c)]	Margin	ART [(b)]
Intermediate Shell Plate R2005-1	Position 1.1	26.0°F	0.76	-20°F	19.8°F	19.8°F	20°F
Intermediate Shell Plate R2005-2	Position 1.1	26.0°F	0.76	-20°F	19.8°F	19.8°F	20°F
Intermediate Shell Plate R2005-3	Position 1.1	31.0°F	0.76	-20°F	23.6°F	23.6°F	27°F
Lower Shell Plate R2508-1	Position 1.1	58.0°F	0.76	0°F	44.1°F	34.0°F	78°F
Lower Shell Plate R2508-2	Position 1.1	37.0°F	0.76	10°F	28.1°F	28.1°F	66°F
Lower Shell Plate R2508-3	Position 1.1	58.0°F	0.76	40°F	44.1°F	34°F	118°F
	Position 2.1	35.8°F	0.76	40°F	27.2°F	17°F	84°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124A & 101-142A (90° Azimuth)	Position 1.1	31.6°F	0.60	-50°F	19.0°F	19°F	-12°F
	Position 2.1	28.3°F	0.60	-50°F	17.0°F	17°F	-16°F
Intermediate & Lower Shell Longitudinal Weld Seam 101-124B,C & 101-142B,C (210° & 330° Azimuth)	Position 1.1	31.6°F	0.76	-50°F	24.0°F	24°F	-2°F
	Position 2.1	28.3°F	0.76	-50°F	21.5°F	21.5°F	-7°F
Intermediate to Lower Shell Girth Weld Seam 101-171	Position 1.1	31.6°F	0.76	-50°F	24.0°F	24°F	-2°F
	Position 2.1	28.3°F	0.76	-50°F	21.5°F	21.5°F	-7°F

Notes:

(a) Initial RT_{NDT} values are measured values (see Table 4-5).

(b) $ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$ (°F)

(c) $\Delta RT_{NDT} = CF * FF$

The lower shell plate R2508-3 is the limiting beltline material for all heatup and cooldown curves to be generated. Contained in Table 4-14 is a summary of the limiting ARTs to be used in the generation of the Wolf Creek reactor vessel heatup and cooldown curves.

TABLE 4-14
Summary of the Limiting ART Values Used in the
Generation of the Wolf Creek Heatup/Cooldown Curves

EFPY	1/4T Limiting ART	3/4T Limiting ART
20	90°F	80°F
32	95°F	84°F

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 discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1and 5-3 present the heatup curves with no margins for possible instrumentation errors for a heatup rates of 60 and 100°F/hr. These curves are applicable for 20 EFPY and 32 EFPY respectively, for the Wolf Creek reactor vessel. Additionally, Figures 5-2 and 5-4 present the cooldown curves with no margins for possible instrumentation errors for cooldown rates of 0, 20, 40, 60, and 100°F/hr. These curves are also applicable for 20 EFPY and 32 EFPY, respectively, for the Wolf Creek reactor vessel. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed 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 Figures 5-1 and 5-3 (for the specific heatup rate 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 equation for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code[3] as follows:

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

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ia} = 26.78 + 1.233\, e^{[0.0145\,(T - RTNDT + 160)]}$,
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 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.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of nonductile failure for the Wolf Creek reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-4 are presented in Tables 5-1 through 5-4, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2508-3
LIMITING ART VALUES AT 20 EFPY: 1/4T, 90°F
3/4T, 80°F

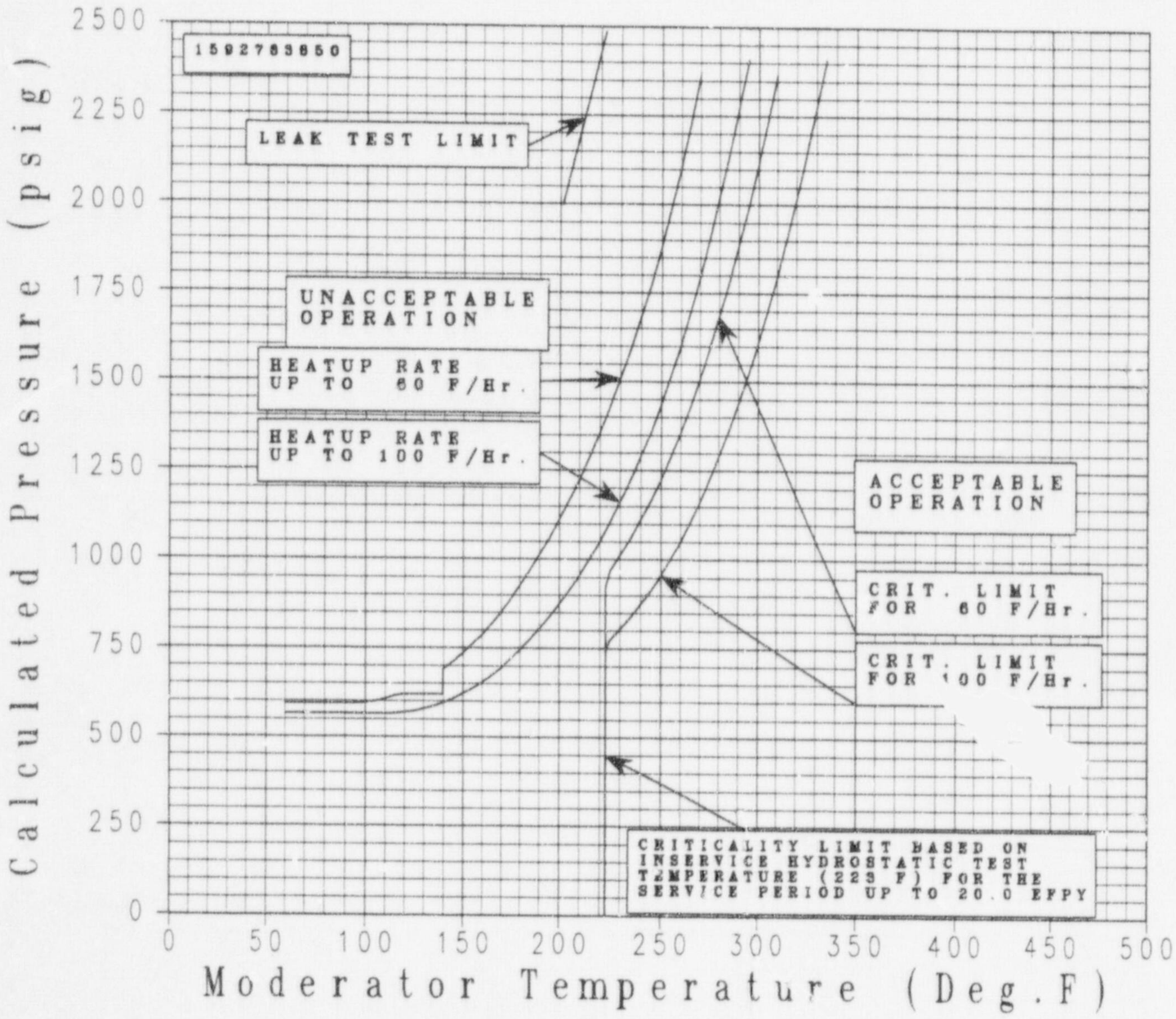

FIGURE 5-1 Wolf Creek Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable to 20 EFPY (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2508-3

LIMITING ART VALUES AT 20 EFPY: 1/4T, 90°F

3/4T, 80°F

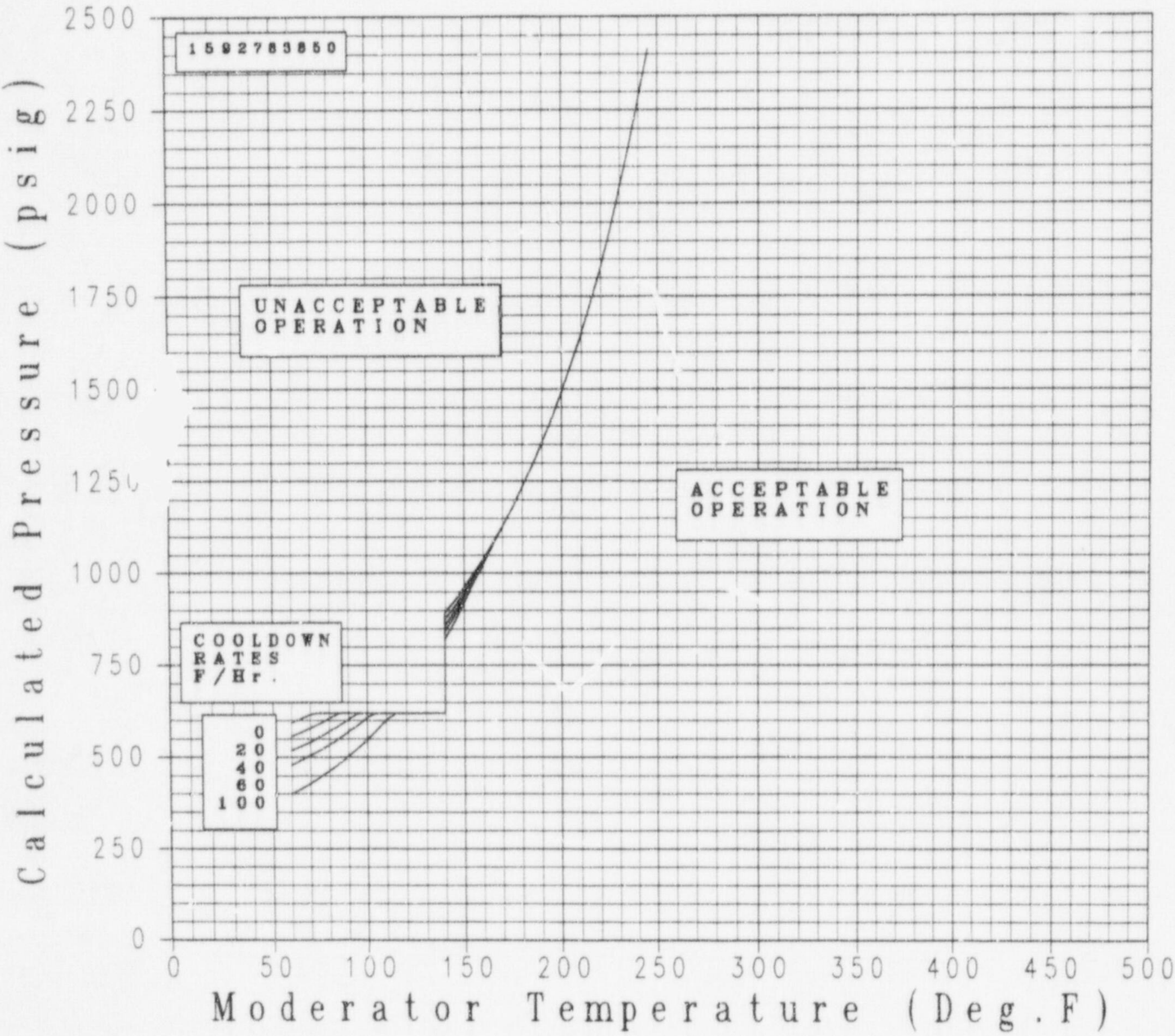

FIGURE 5-2 Wolf Creek Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable to 20 EFPY (Without Margins for Instrumentation Errors)

TABLE 5-1
Wolf Creek Heatup Data at 20 EFPY
Without Margins for Instrumentation Errors

Run = 1592783850

Heatup Curves

60 F/hr		60 F/hr Crit. Limit		100 F/hr		100 F/hr Crit. Limit		Leak Test Limit	
Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)
60	0	223	0	60	0	223	0	201	2000
60	595	223	605	60	567	223	567	223	2485
65	596	223	604	65	567	223	567		
85	596	223	598	85	567	223	567		
90	596	223	596	90	567	223	567		
95	596	223	596	95	567	223	567		
100	596	223	600	100	567	223	567		
105	600	223	606	105	567	223	567		
110	606	223	615	110	567	223	567		
115	615	223	626	115	567	223	569		
120	621	223	639	120	569	223	573		
125	621	223	654	125	573	223	579		
130	621	223	672	130	579	223	587		
135	621	223	691	135	587	223	596		
140	621	223	712	140	596	223	608		
140	691	223	736	145	608	223	622		
145	712	223	761	150	622	223	637		
150	736	223	789	155	637	223	655		
155	761	223	820	160	655	223	674		
160	789	223	852	165	674	223	696		
165	820	223	888	170	696	223	719		
170	852	223	926	175	719	223	745		
175	888	225	967	180	745	225	774		
180	926	230	1012	185	774	230	805		
185	967	235	1060	190	805	235	838		
190	1012	240	1111	195	838	240	875		
195	1060	245	1166	200	875	245	914		
200	1111	250	1225	205	914	250	956		
205	1166	255	1289	210	956	255	1002		
210	1225	260	1357	215	1002	260	1052		
215	1289	265	1430	220	1052	265	1105		
220	1357	270	1509	225	1105	270	1162		
225	1430	275	1593	230	1162	275	1223		
230	1509	280	1683	235	1223	280	1289		
235	1593	285	1779	240	1289	285	1360		
240	1683	290	1882	245	1360	290	1436		
245	1779	295	1992	250	1436	295	1517		
250	1882	300	2110	255	1517	300	1605		
255	1992	305	2235	260	1605	305	1698		
260	2110	310	2369	265	1698	310	1798		
265	2235			270	1798	315	1905		
270	2369			275	1905	320	2019		
				280	2019	325	2141		
				285	2141	330	2271		
				290	2271	335	2409		
				295	2409				

TABLE 5-2

Wolf Creek Cooldown Data at 20 EFPY

Without Margins for Instrumentation Errors

Run = 1592783850

Cooldown Curves

Steady State		20 F/hr		40 F/hr		60 F/hr		100 F/hr	
Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)	Temp. (F)	Press. (psi)
60	0	60	0	60	0	60	0	60	0
60	595	60	557	60	518	60	480	60	401
65	605	65	568	65	530	65	492	65	414
70	616	70	579	70	542	70	505	70	429
75	621	75	592	75	556	75	519	75	445
80	621	80	606	80	570	80	534	80	463
85	621	85	620	85	586	85	551	85	482
90	621	90	621	90	602	90	569	90	502
95	621	95	621	95	620	95	588	95	525
100	621	100	621	100	621	100	609	100	549
105	621	105	621	105	621	105	621	105	574
110	621	110	621	110	621	110	621	110	602
115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
140	894	140	876	140	859	140	844	140	822
145	927	145	911	145	897	145	885	145	870
150	962	150	948	150	937	150	929	150	920
155	1000	155	989	155	981	155	975	155	975
160	1040	160	1032	160	1028	160	1026	160	1034
165	1084	165	1079	165	1078	165	1080		
170	1130	170	1129						
175	1181								
180	1234								
185	1292								
190	1354								
195	1421								
200	1492								
205	1569								
210	1651								
215	1739								
220	1833								
225	1934								
230	2042								
235	2158								
240	2281								
245	2413								

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2508-3
LIMITING ART VALUES AT 32 EFPY: 1/4T, 95°F
3/4T, 84°F

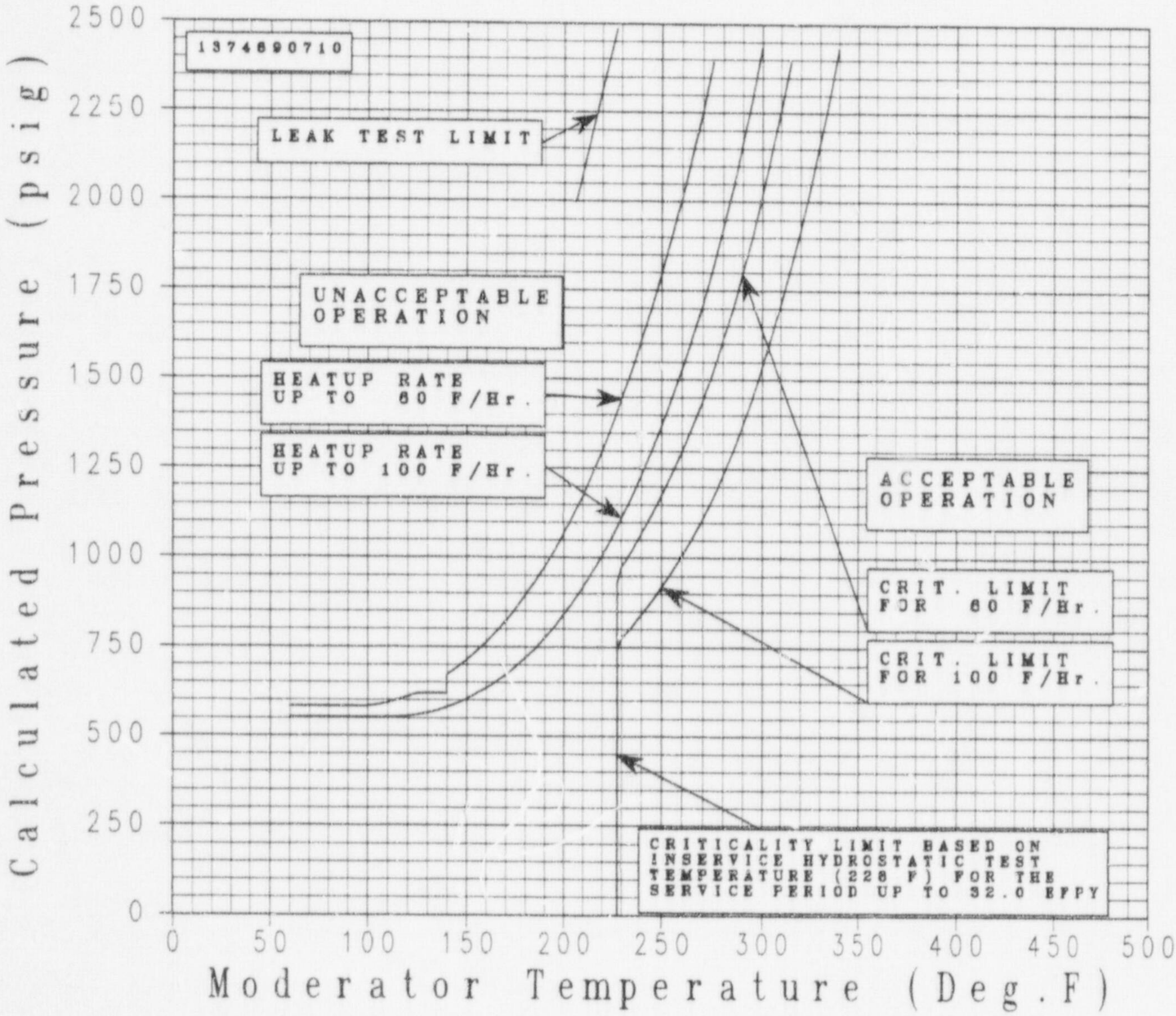

FIGURE 5-3 Wolf Creek Reactor Coolant System Heatup Limitations (Heatup Rate of 60 and 100°F/hr) Applicable to 32 EFPY (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATE R2508-3
LIMITING ART VALUES AT 32 EFPY: 1/4T, 95°F
3/4T, 84°F

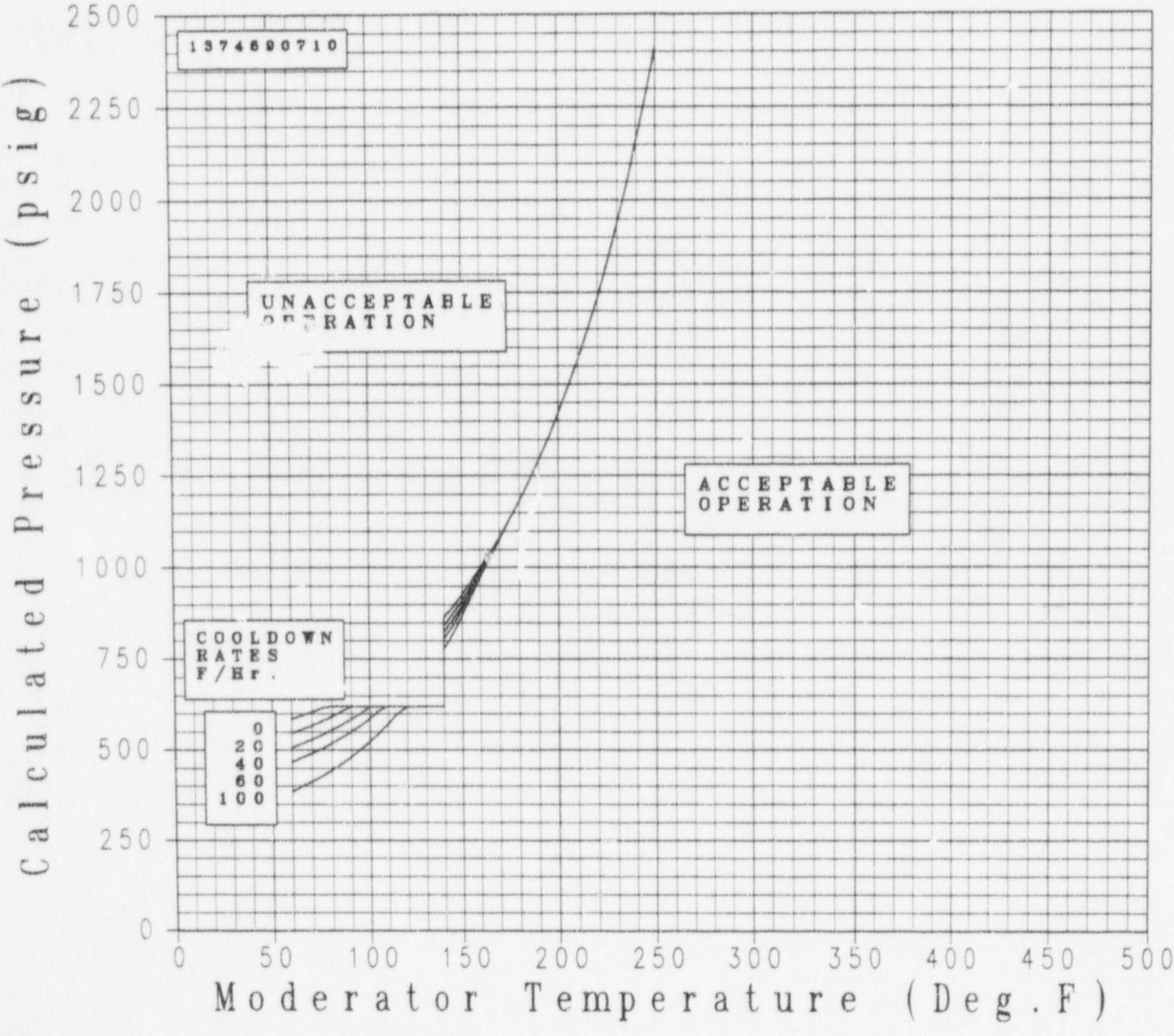

FIGURE 5-4 Wolf Creek Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable to 32 EFPY (Without Margins for Instrumentation Errors)

TABLE 5-3
Wolf Creek Heatup Data at 32 EFPY
Without Margins for Instrumentation Errors

Run = 1374690710

Heatup Curves

60 F/hr		60 F/hr Crit. Limit		100 F/hr		100 F/hr Crit. Limit		Leak Test Limit	
Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)
60	0	228	0	60	0	228	0	206	2000
60	584	228	595	60	554	228	554	228	2485
65	584	228	594	65	554	228	554		
85	584	228	587	85	554	228	554		
90	584	228	584	90	554	228	554		
95	584	228	585	95	554	228	554		
100	585	228	588	100	554	228	554		
105	588	228	593	105	554	228	554		
110	593	228	601	110	554	228	554		
115	601	228	611	115	554	228	556		
120	611	228	624	120	556	228	559		
125	621	228	638	125	559	228	564		
130	621	228	654	130	564	228	572		
135	621	228	672	135	572	228	581		
140	621	228	692	140	581	228	591		
140	672	228	714	145	591	228	604		
145	692	228	738	150	604	228	618		
150	714	228	764	155	618	228	635		
155	738	228	793	160	635	228	653		
160	764	228	824	165	653	228	673		
165	793	228	857	170	673	228	695		
170	824	228	893	175	695	228	719		
175	857	228	932	180	719	228	746		
180	893	230	974	185	746	230	775		
185	932	235	1019	190	775	235	807		
190	974	240	1068	195	807	240	841		
195	1019	245	1120	200	841	245	878		
200	1068	250	1176	205	878	250	918		
205	1120	255	1236	210	918	255	961		
210	1176	260	1300	215	961	260	1008		
215	1236	265	1369	220	1008	265	1058		
220	1300	270	1444	225	1058	270	1112		
225	1369	275	1523	230	1112	275	1170		
230	1444	280	1608	235	1170	280	1232		
235	1523	285	1699	240	1232	285	1299		
240	1608	290	1797	245	1299	290	1371		
245	1699	295	1901	250	1371	295	1448		
250	1797	300	2013	255	1448	300	1530		
255	1901	305	2132	260	1530	305	1619		
260	2013	310	2259	265	1619	310	1713		
265	2132	315	2394	270	1713	315	1814		
270	2259			275	1814	320	1923		
275	2394			280	1923	325	2038		
				285	2038	330	2161		
				290	2161	335	2293		
				295	2293	340	2433		
				300	2433				

TABLE 5-4
Wolf Creek Cooldown Data at 32 EFPY
Without Margins for Instrumentation Errors

Run = 1374690710

Cooldown Curves

Steady State		20 F/hr		40 F/hr		60 F/hr		100 F/hr	
Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)	Temp. (F)	Pres. (psi)
60	0	60	0	60	0	60	0	60	0
60	585	60	547	60	507	60	468	60	387
65	595	65	557	65	518	65	479	65	400
70	605	70	568	70	530	70	491	70	413
75	616	75	579	75	542	75	504	75	428
80	621	80	592	80	555	80	519	80	445
85	621	85	605	85	570	85	534	85	462
90	621	90	620	90	585	90	550	90	481
95	621	95	621	95	602	95	568	95	502
100	621	100	621	100	620	100	588	100	524
105	621	105	621	105	621	105	608	105	548
110	621	110	621	110	621	110	621	110	574
115	621	115	621	115	621	115	621	115	602
120	621	120	621	120	621	120	621	120	621
125	621	125	621	125	621	125	621	125	621
130	621	130	621	130	621	130	621	130	621
135	621	135	621	135	621	135	621	135	621
140	621	140	621	140	621	140	621	140	621
140	864	140	843	140	824	140	806	140	778
145	894	145	876	145	859	145	844	145	822
150	927	150	911	150	896	150	885	150	869
155	962	155	948	155	937	155	928	155	920
160	1000	160	989	160	981	160	975	160	975
165	1040	165	1032	165	1027	165	1026	165	1034
170	1084	170	1079	170	1078	170	1080		
175	1130	175	1129						
180	1181								
185	1234								
190	1292								
195	1354								
200	1421								
205	1492								
210	1569								
215	1651								
220	1739								
225	1833								
230	1934								
235	2042								
240	2158								
245	2281								
250	2413								

6 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, Appendix G, "Fracture Toughness Requirements", 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 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

5 WCAP-13365, Rev. 1, "Analysis of Capsule Y from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Program", J.M. Chicots, et al., April 1993.

6 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

7 WCAP-15078, "Analysis of Capsule V from the Wolf Creek Nuclear Operating Corporation Wolf Creek Reactor Vessel Radiation Surveillance Program", Ed Terek, et al., August 1998.

8 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.

9 CE NPSD-1039, Rev. 2, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds, Appendix A, CE Reactor Vessel Weld Properties Database, Volume 1," CEOG Task 902, June 1997.