Westinghouse Non-Proprietary Class 3

V. C. Summer Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

WCAP-15102
Revision 1

Westinghouse Electric Company LLC

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P PDR

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Ed Terek

July 1999

Work Performed Under Shop Order CJWP139Q

Prepared by the Westinghouse Electric Company, LLC
for the South Carolina Electric and Gas Company

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

Approved: Kevin R. Neuhart for CHB
C. H. Boyd, Manager
Equipment & Materials Technology

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

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PREFACE

This report has been technically reviewed and verified by:

Reviewer: T. J. Laubham

EXECUTIVE SUMMARY

Revision 0:

The purpose of this report is to generate pressure-temperature limit curves for V. C. Summer Unit 1 for normal operation at 20 and 32 EFPY using the methodology from WCAP 14040-NP-A which encompasses the requirements of 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 lower shell plates C9923-1 and C9923-2 (i.e. the limiting beltline region material). The pressure-temperature limit curves were generated for heatup rates of 50 and 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. These curves can be found in Figures 5-1 through 5-4.

Revision 1:

The purpose of this revision is to generate pressure pressure-temperature limit curves for V. C. Summer Unit 1 for normal operation at 20 and 32 EFPY utilizing updated methodology and without instrumentation error margins. The updated methodology that is being employed is the utilization of the 1996 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, along with ASME Code Case N-640. In addition, this report contains a justification for a reduced flange temperature requirement which was also incorporated into the pressure-temperature curves provided in this report. All other calculations/data remains unchanged.

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} (ART) 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 (transverse to the major rolling direction for late material) 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/initial 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 ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the ¼T and ¾T 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 BACKGROUND AND PURPOSE

Appendix G to the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure"[3] was updated in 1996 and ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[4], was approved in February of 1999. The 1996 ASME Section XI, Appendix G, provides a more accurate methodology for calculating stress intensity factors due to the thermal and pressure stresses at the ¼T and ¾T locations while Code Case N-640 allows the use of the K_{IC} methodology rather than the K_{IA} methodology.

In September of 1998 Westinghouse completed the analysis surveillance capsule W from the V. C. Summer Unit 1 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 developed per the methodology given in WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[5] and included generic margins for instrumentation errors. In Addition, these curves included 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 revision is to present the calculations and development of the South Carolina Electric & Gas Company V. C. Summer Unit 1 pressure-temperature curves for 20 and 32 EFPY utilizing the 1996 Appendix G to the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Fracture Toughness Criteria for Protection Against Failure" along with ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1". In addition, this report provides technical justification for relaxing the flange temperature requirement of Appendix G to 10 CFR Part 50 based on the use of the K_{IC} methodology rather than the K_{IA} methodology. These pressure-temperature curves are being developed for normal operation up to 20 and 32 EFPY and do not include margins for instrumentation errors.

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 Overall Approach

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_{Ic} , for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640 of ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

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

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-2, and SA-508-3 steels.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined as:

$$C * K_{Im} + K_{It} < K_{IC} \quad (2)$$

where,

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

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

K_{1C} = 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

For membrane tension, the K_I corresponding to membrane tension for the postulated defect is:

$$K_{Im} = M_m * (pR_i \div t) \quad (3)$$

Where M_m for an inside surface is given by:

$M_m = 1.85$ for $\sqrt{t} < 2$,
$M_m = 0.926 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$, and
$M_m = 3.21$ for $\sqrt{t} > 3.464$.

Similarly, M_m for an outside surface flaw is given by:

$M_m = 1.77$ for $\sqrt{t} < 2$,
$M_m = 0.893 \sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$, and
$M_m = 3.09$ for $\sqrt{t} > 3.464$.

where:

p = internal pressure,
Ri = vessel inner radius, and
t = vessel wall thickness.

For Bending Stress, the K_I corresponding to bending stress for the postulated defect is:

$K_{Ib} = M_b$ * maximum bending stress, where M_b is two-thirds of M_m

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated inside surface defect is:

$$K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5} \qquad (4)$$

where:
CR = the cooldown rate in °F/hr.

For the Radial Thermal Gradient, the maximum K_I produced by radial thermal gradient for the postulated outside surface defect is:

$$K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5} \qquad (5)$$

where:
HU = the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from ASME Section XI, Appendix G, Figure G-2214-1. The temperature at any radial distance from the vessel surface can be determined from ASME Section XI, Appendix G, Figure G-2214-2 for the maximum thermal K_I .

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2) of Appendix G to ASME Section XI.

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{6}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{7}$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{8}$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3 through 8 were added to the OPERLIM computer program, which is the Westinghouse computer program used to generate pressure-temperature limit curves. No other changes were made to the OPERLIM computer program with regard to the pressure-temperature curve calculation methodology. Hence, the pressure-temperature curve methodology described in WCAP-14040[5] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) remains valid for the generation of the pressure-temperature curves documented in this report with the exceptions described above.

At any time during the heatup or cooldown transient, K_{IC} is determined by the metal temperature at the tip of a postulated flaw at the ¼T and ¾T 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 ¼T 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 (temperature) developed during cooldown results in a higher value of K_{IC} at the ¼T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{IC} 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 ¼T 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 ¼T 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_{IC} for the ¼T crack during heatup is lower than the K_{IC} for the ¼T 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_{IC} 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 ¼T 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 ¼T flaw located at the ¼T 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.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G contains the requirements for 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 pre-service hydrostatic test pressure (3106 psig), which is 621 psig for the V. C. Summer Unit 1 reactor vessel.

This requirement was originally based on concerns about the fracture margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120F and the pressure limitation of 20 percent of hydrotest pressure were developed using the K_{ia} fracture toughness, in the mid 1970s.

Improved knowledge of fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{ic} in the development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1". The following discussion uses a similar approach (i.e. using K_{ic}) is used here to develop equivalent flange requirements.

The geometry of the closure head flange region for a typical Westinghouse four loop plant reactor vessel which is more conservative than the geometry of a three loop plant reactor vessel such as the V. C. Summer Unit 1 reactor vessel is shown in Figure 3-1. The stresses in this region are highest near the outside surface of the head. Hence, a outside reference flaw of 25 percent of the wall thickness parallel to the dome to flange weld (i.e. in the direction of welding) was postulated in this region. To be consistent with ASME Section XI, Appendix G, a safety factor of two was applied and a fracture calculation performed.

Figure 3-2 shows the crack driving force or stress intensity factor for the postulated flaw in this region, along with a second curve which incorporates the safety factor of two. Note that the stress intensity factor with a safety factor of one for this region does not exceed 55 ksi√in., even for postulated flaws of up to 50 percent of the wall thickness. For the reference flaw, with the safety factor of two, the applied stress intensity factor is 85.15 ksi√in. at 25 percent of the wall thickness.

The determination of the bolt-up, or flange requirement, is shown in Figure 3-3, where the fracture toughness is plotted as a function of the temperature. In this figure, the intersection between the stress intensity factor curve and the K_{ia} toughness curve occurs at a value slightly higher than $T\text{-}RT_{NDT}$ = 100°F, which is in the range of the existing 120°F requirement. The reference calculation used for the original requirement (which is no longer available) resulted in a temperature requirement of $T - RT_{NDT}$ = 120°F. This corresponds to a K_{ia} (with a safety factor of 2) of 98 ksi√in. Note that the use of the K_{ic} curve to determine this requirement results in a revised requirement of $T\text{-}RT_{NDT}$ = 45°F, as seen in Figure 3-3.

Therefore, the appropriate flange requirement for use with the K_{Ic} curve is as follows:

The pressure in the vessel should not exceed 20 percent of the pre-service hydro-test pressure until the temperature exceeds $T - RT_{NDT} = 45°F$. This requirement has been implemented with the curves presented in this report.

The limiting unirradiated RT_{NDT} of 10°F (Table 4-5 in of this report) occurs in closure head flange 5297-V1 of the V. C. Summer Unit 1 reactor vessel, so the minimum allowable temperature of this region is 55°F at pressures greater than 621 psig with no margin for uncertainties.

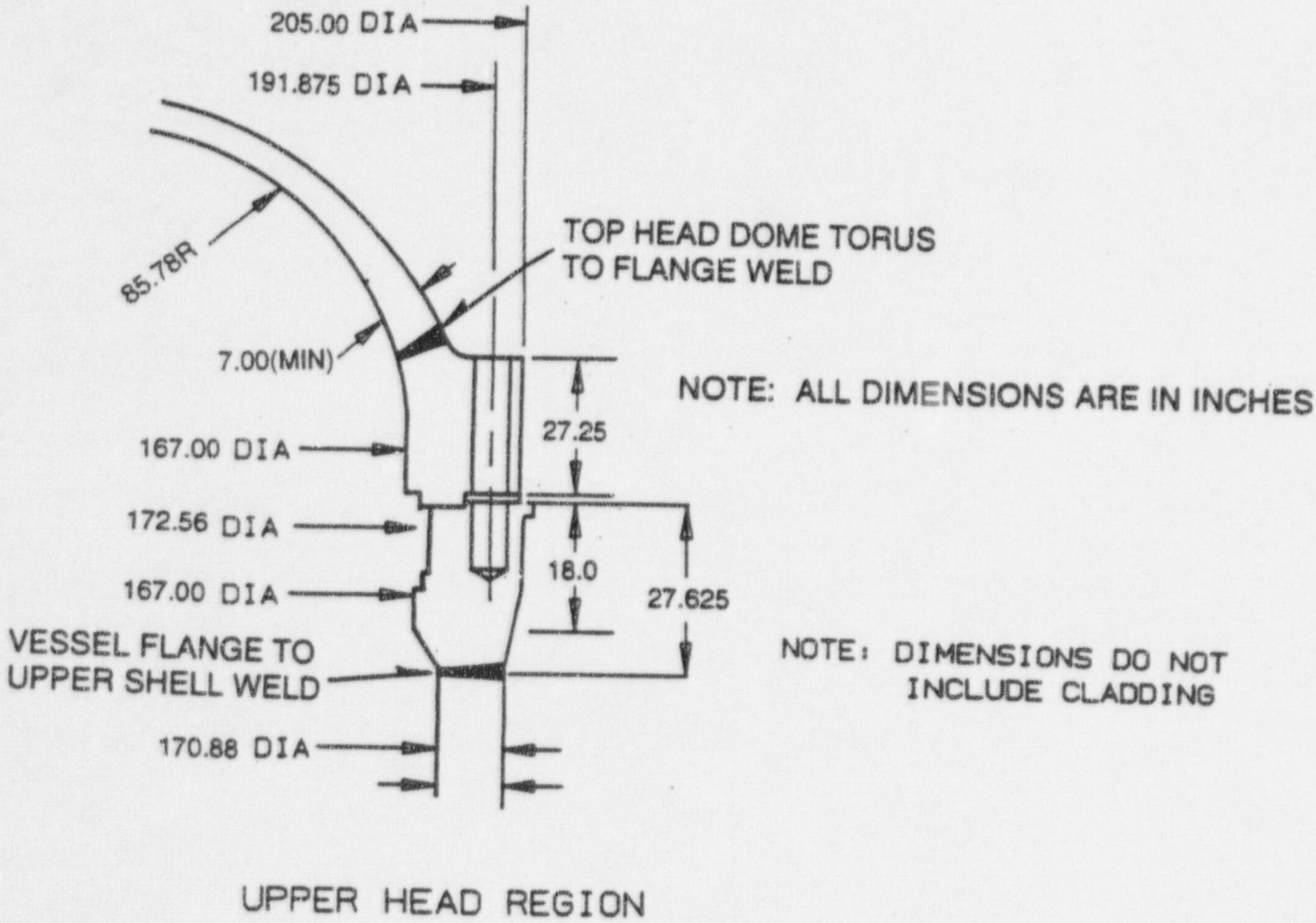

Figure 3-1 Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

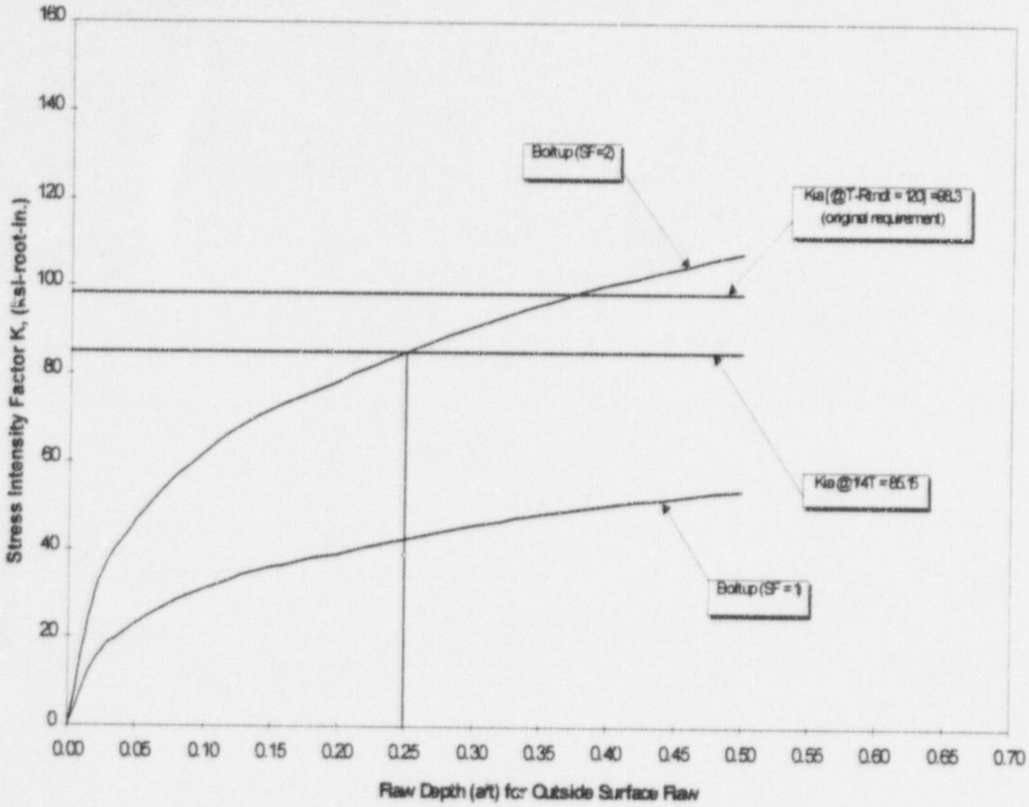

Figure 3-2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

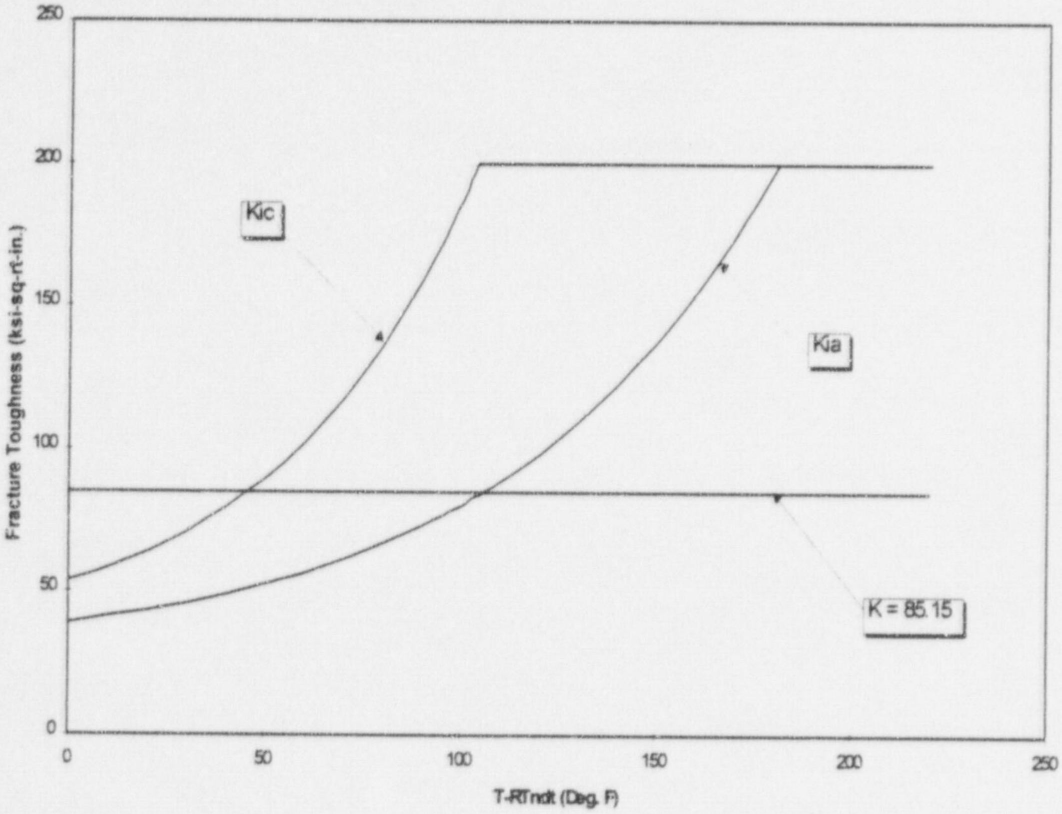

Figure 3-3 Determination of Boltup Requirement, using K_{IC}

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:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \tag{9}$$

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 is calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10 \log f)} \tag{10}$$

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.

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \tag{11}$$

where x inches (vessel beltline thickness is 7.75 inches[7]) 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 and the results are presented in Section 6 of WCAP-15101[8]. 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"[5]. Tables 4-1 and 4-2, herein, contain the calculated 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 V. C. Summer Unit 1 reactor vessel. Additionally, the calculated 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)

Azimuth	Surface	¼ T	¾ T
Intermediate Shell Plate A9154-1	2.45×10^{19}	1.54×10^{19}	6.07×10^{18}
Intermediate Shell Plate A9153-2	2.45×10^{19}	1.54×10^{19}	6.07×10^{18}
Lower Shell Plate C9923-1	2.45×10^{19}	1.54×10^{19}	6.07×10^{18}
Lower Shell Plate C9923-2	2.45×10^{19}	1.54×10^{19}	6.07×10^{18}
Intermediate Lower Shell Longitudinal Weld Seam BC, BD and BA, BB (0° Azimuth)	0.892×10^{19}	5.60×10^{18}	2.21×10^{18}
Intermediate to Lower Shell Cirumferential Weld Seam AB	2.45×10^{19}	1.54×10^{19}	6.07×10^{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 A9154-1	3.84×10^{19}	2.41×10^{19}	9.52×10^{18}
Intermediate Shell Plate A9153-1	3.84×10^{19}	2.41×10^{19}	9.52×10^{18}
Lower Shell Plate C9923-1	3.84×10^{19}	2.41×10^{19}	9.52×10^{18}
Lower Shell Plate C9923-2	3.84×10^{19}	2.41×10^{19}	9.52×10^{18}
Intermediate & Lower Shell Longitudinal Weld Seam BC, BD and BA, BB (45° Azimuth)	1.43×10^{19}	8.98×10^{18}	3.54×10^{18}
Intermediate to Lower Shell Cirumferential Weld Seam AB	3.84×10^{19}	2.41×10^{19}	9.52×10^{18}

TABLE 4-3

Calculated Integrated Neutron Exposure of the V. C. Summer Unit 1 Surveillance Capsules Tested to Date

Capsule	Fluence
U	6.542×10^{18} n/cm^2, (E > 1.0 MeV)
V	1.538×10^{19} n/cm^2, (E > 1.0 MeV)
X	2.543×10^{19} n/cm^2, (E > 1.0 meV)
W	4.664×10^{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[8]. These measured shift values were obtained using CVGRAPH, Version 4.1[9], 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)]
Intermediate Shell Plate A9154-1	U	36.0°F
	V	52.6°F
	X	37.7°F
(Longitudinal Orientation)	W	65.7°F
Intermediate Shell Plate A9154-1	U	14.5°F
	V	32.4°F
	X	26.0°F
(Transverse Orientation)	W	57.8°F
Surveillance Program	U	22.2°F
	V	46.5°F
Weld Metal	X	22.4°F
	W	43.3°F
	U	35.6°F
	V	50.1°F
Heat Affected Zone	X	54.1°F
	W	60.3°F

Notes:

(a) From capsule W dosimetry analysis results[8], ($x10^{19}$ n/cm2, E>1.0 .MeV).

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[7 & 10]

Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}[(a)]**
Closure Head Flange 5297-V1[(b)]	- -	- -	10°F[(b)]
Vessel Flange 5301-V-1	- -	- -	0°F[(b)]
Intermediate Shell Plate A9154-1	0.10	0.51	30°F
Intermediate Shell Plate A9153-2	0.09	0.45	-20°F
Lower Shell Plate C9923-1	0.08	0.41	10°F
Lower Shell Plate C9923-2	0.08	0.41	10°F
Intermediate Shell Longitudinal Welds Seams BC & BD	0.05	0.91	-44°F
Intermediate Shell Longitudinal Welds Seams BA & BB	0.05	0.91	-44°F
Intermediate Lower Shell Plate Circumferential Weld Seam A5	0.05	0.91	-44°F
Surveillance Program Weld Metal	0.04	0.95	- - -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data per WCAP-12867[7]

(b) In the past the closure head flange was reported as Heat A9231 with a IRT_{NDT} of -20°F. Based on a review of Westinghouse files, the correct data is Heat # 5297-V1 with a IRT_{NDT} of 10°F. Also, the vessel flange was reported a IRT_{NDT} of 10°F., however, based on a review Westinghouse files, the correct IRT_{NDT} of 0°F.

TABLE 4-6

Calculation of Chemistry Factors using V. C. Summer Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF^*\Delta RT_{NDT}$	FF^2
Intermediate Shell	U	0.654	0.881	36.0	31.7	0776
Plate A9154-1	V	1.538	1.119	52.6	58.9	1.252
	X	2.543	1.250	37.7	47.1	1.563
(Longitudinal)	W	4.664	1.388	65.7	91.2	1.927
Intermediate Shell	U	0.654	0.881	14.5	12.8	0.776
Plate A9154-1	V	1.538	1.119	32.4	36.3	1.252
	X	2.543	1.250	26.0	32.5	1.563
(Transverse)	W	4.664	1.388	57.8	80.2	1.927
				SUM	390.7	11.036
	$CF_{A9154-1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (390.7) \div (11.036) = 35.4°F$					
Surveillance Weld	U	0.654	0.881	28.0[d]	24.7	0776
Material	V	1.538	1.119	58.6[d]	65.6	1.252
	X	2.543	1.250	28.3[d]	35.4	1.563
	W	4.664	1.388	54.4[d]	75.5	1.927
				SUM	201.2	5.518
	$CF_{Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (201.2) \div (5.518) = 36.5°F$					

Notes:

(a) F= Measured fluence from capsule W dosimetry analysis results(8) (x 10^{19} n/cm^2, E > 1.0 MeV).

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

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

(d) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio of 1.26. ($CF_{VW} \div CF_{SW}$ = 68°F ÷ 54°F = 1.26)

TABLE 4-7

Summary of the V. C. Summer Unit 1 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	**Position 2.1**
Intermediate Shell Plate A9154-1	65.0°F	35.4°F
Intermediate Shell Plate A9153-2	58.0°F	---
Lower Shell Plate C9923-1	51.0°F	---
Lower Shell Plate C9923-2	51.0°F	---
Intermediate Shell Longitudinal Weld Seams BC & BD	68.0°F	36.5°F
Lower Shell Longitudinal Weld Seams, BA & BB	68.0°F	36.5°F
Intermediate to Lower Shell Plate Circumferential Weld Seam AB	68.0°F	36.5°F

Note: See Reference 8 for the credibility evaluation of the V.C. Summer Unit 1 surveillance data.

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 V. C. Summer Unit 1 reactor vessel beltline materials for 20 EFPY and 32 EFPY.

TABLE 4-8
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 20 EPFY Heatup/Cooldown Curves

Azimuth	**1/4 T F** (n/cm^2, E > 1.0 MeV)	**1/4T FF**	**3/4T F** (n/cm^2, E >1.0 MeV)	**3/4 T FF**
Intermediate Shell Plate A9154-1	1.54×10^{19}	1.12	6.07×10^{18}	0.860
Intermediate Shell Plate A9153-2	1.54×10^{19}	1.12	6.07×10^{18}	0.860
Lower Shell Plate C9923-1	1.54×10^{19}	1.12	6.07×10^{18}	0.860
Lower Shell Plate C9923-2	1.54×10^{19}	1.12	6.07×10^{18}	0.860
Intermediate & Lower Shell Longitudinal Weld Seams BC, BD and BA, BB (45° Azimuth)	5.60×10^{18}	0.838	2.21×10^{18}	0.594
Intermediate to Lower Shell Cirumferential Weld Seam AB	1.54×10^{19}	1.12	6.07×10^{19}	0.860

TABLE 4-9
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Plate A9154-1	2.41×10^{19}	1.24	9.52×10^{18}	0.986
Intermediate Shell Plate A9153-2	2.41×10^{19}	1.24	9.52×10^{18}	0.986
Lower Shell Plate C9923-1	2.41×10^{19}	1.24	9.52×10^{18}	0.986
Lower Shell Plate C9923-2	2.41×10^{19}	1.24	9.52×10^{18}	0.986
Intermediate & Lower Shell Longitudinal Weld Seams BC, BD and BA, BB (45° Azimuth)	8.98×10^{18}	0.970	3.54×10^{18}	0.713
Intermediate to Lower Shell Cirumferential Weld Seam A5	2.41×10^{19}	1.24	9.52×10^{18}	0.986

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 (°F)	FF	IRT_{NDT} (a)	ΔRT_{NDT} (c)	Margin	ART (b)
Intermediate Shell Plate A9154-1	Position 2.1	35.4	1.12	30	39.6	17	87
Intermediate Shell Plate A9153-2	Position 1.1	58	1.12	-20	65.0	34	79
Lower Shell Plate C9923-1	Position 1.1	51	1.12	10	57.1	34	**101**
Lower Shell Plate C9923-2	Position 1.1	51	1.12	10	57.1	34	**101**
Inter. & Lower Shell Longitudinal Weld Seams BC, BD and BA, BB (45° Azimuth)	Position 2.1	36.5	0.838	-44	30.6	28	15
Intermediate to Lower Shell Circumferential Weld Seam AB	Position 2.1	36.5	1.12	-44	40.9	28	25

Notes:

(a) Initial RT_{NDT} values are measured values.

(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 (°F)**	**FF**	**IRT_{NDT}[a]**	**ΔRT_{NDT}[c]**	**Margin**	**ART[b]**
Intermediate Shell Plate A9154-1	Position 2.1	35.4	0.860	30	30.4	17	77
Intermediate Shell Plate A9153-2	Position 1.1	58	0.860	-20	50.0	34	64
Lower Shell Plate C9923-1	Position 1.1	51	0.860	10	43.9	34	**88**
Lower Shell Plate C9923-2	Position 1.1	51	0.860	10	43.9	34	**88**
Inter. & Lower Shell Longitudinal Weld Seams BC, BD and BA, BB (45° Azimuth)	Position 2.1	36.5	0.594	-44	21.6	21.6	-1
Intermediate to Lower Shell Circumferential Weld Seam AB	Position 2.1	36.5	0.860	-44	31.4	28	15

Notes:-

(a) Initial RT_{NDT} values are measured values.

(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 A9154-1	Position 2.1	35.4	1.24	30	43.9	17	91
Intermediate Shell Plate A9153-2	Position 1.1	58	1.24	-20	71.9	34	86
Lower Shell Plate C9923-1	Position 1.1	51	1.24	10	63.2	34	**107**
Lower Shell Plate C9923-2	Position 1.1	51	1.24	10	63.2	34	**107**
Inter. & Lower Shell Longitudinal Weld Seams BC, BD and BA, BB (45° Azimuth)	Position 2.1	36.5	0.970	-44	35.4	28	19
Intermediate to Lower Shell Circumferential Weld Seam AB	Position 2.1	36.5	1.24	-44	45.3	28	29

Notes:

(a) Initial RT_{NDT} values are measured values.

(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 A9154-1	Position 2.1	35.4	0.986	30	34.9	17	82
Intermediate Shell Plate A9153-2	Position 1.1	58	0.986	-20	57.2	34	71
Lower Shell Plate C9923-1	Position 1.1	51	0.986	10	50.3	34	**94**
Lower Shell Plate C9923-2	Position 1.1	51	0.986	10	50.3	34	**94**
Inter. & Lower Shell Longitudinal Weld Seams BC, BD and BA, BB (45° Azimuth)	Position 2.1	36.5	0.713	-44	26.0	26	8
Intermediate to Lower Shell Circumferential Weld Seam AB	Position 2.1	36.5	0.986	-44	36.0	28	20

Notes:

(a) Initial RT_{NDT} values are measured values.

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

(c) ΔRT_{NDT} = CF * FF

The lower shell plates C9923-1 and C9923-2 are the limiting beltline materials 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 V. C. Summer Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-14
Summary of the Limiting ART Values Used in the Generation of the V. C. Summer Unit 1 Heatup/Cooldown Curves

EFPY	1/4T Limiting ART	3/4T Limiting ART
20	101°F	88°F
32	107°F	94°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.

Figures 5-1 and 5-3 present the heatup curves with margins of 10°F and 60 psig for possible instrumentation errors for heatup rates of 50 and 100°F/hr. The curves are applicable for 20 EFPY and 32 EFPY respectively, for the V. C. Summer Unit 1 reactor vessel. Additionally, Figures 5-2 and 5-4 present the cooldown curves without margins for possible instrumentation errors for cooldown rates of 0, 25, 50 and 100°F/hr. These curves are also applicable for 20 EFPY and 32 EFPY, respectively, for the V. C. Summer Unit 1 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:

$$1.5K_{Im} < K_{Ic} \quad (12)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,
$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RT_{NDT})]}$,
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.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of nonductile failure for the V. C. Summer Unit 1 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 PLATES C9923-1, -2

LIMITING ART VALUES AT 20 EFPY: 1/4T, 101°F

3/4T, 88°F

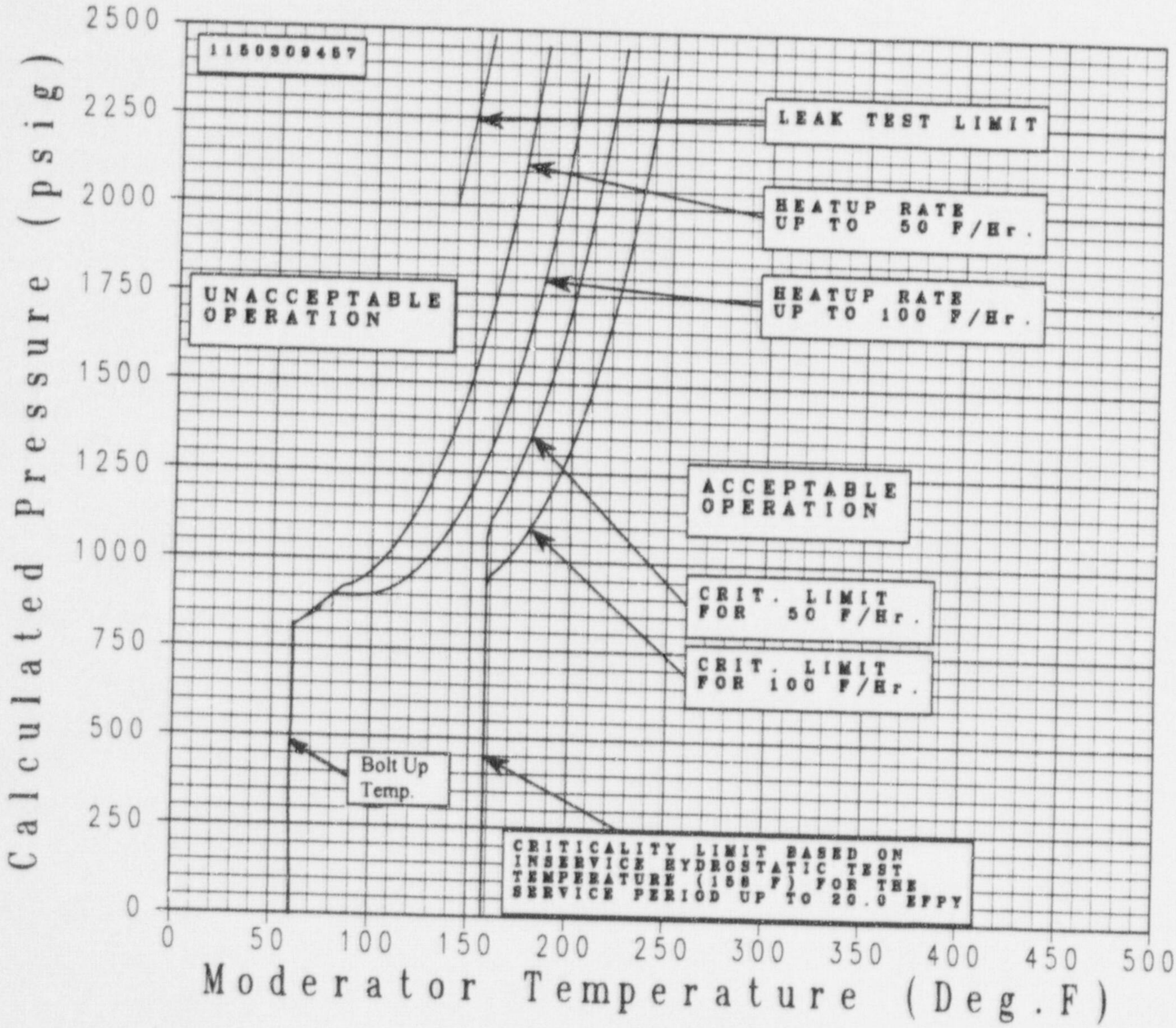

FIGURE 5-1 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 50 and 100°F/hr) Applicable to 20 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1, -2
LIMITING ART VALUES AT 20 EFPY: 1/4T, 101°F
3/4T, 88°F

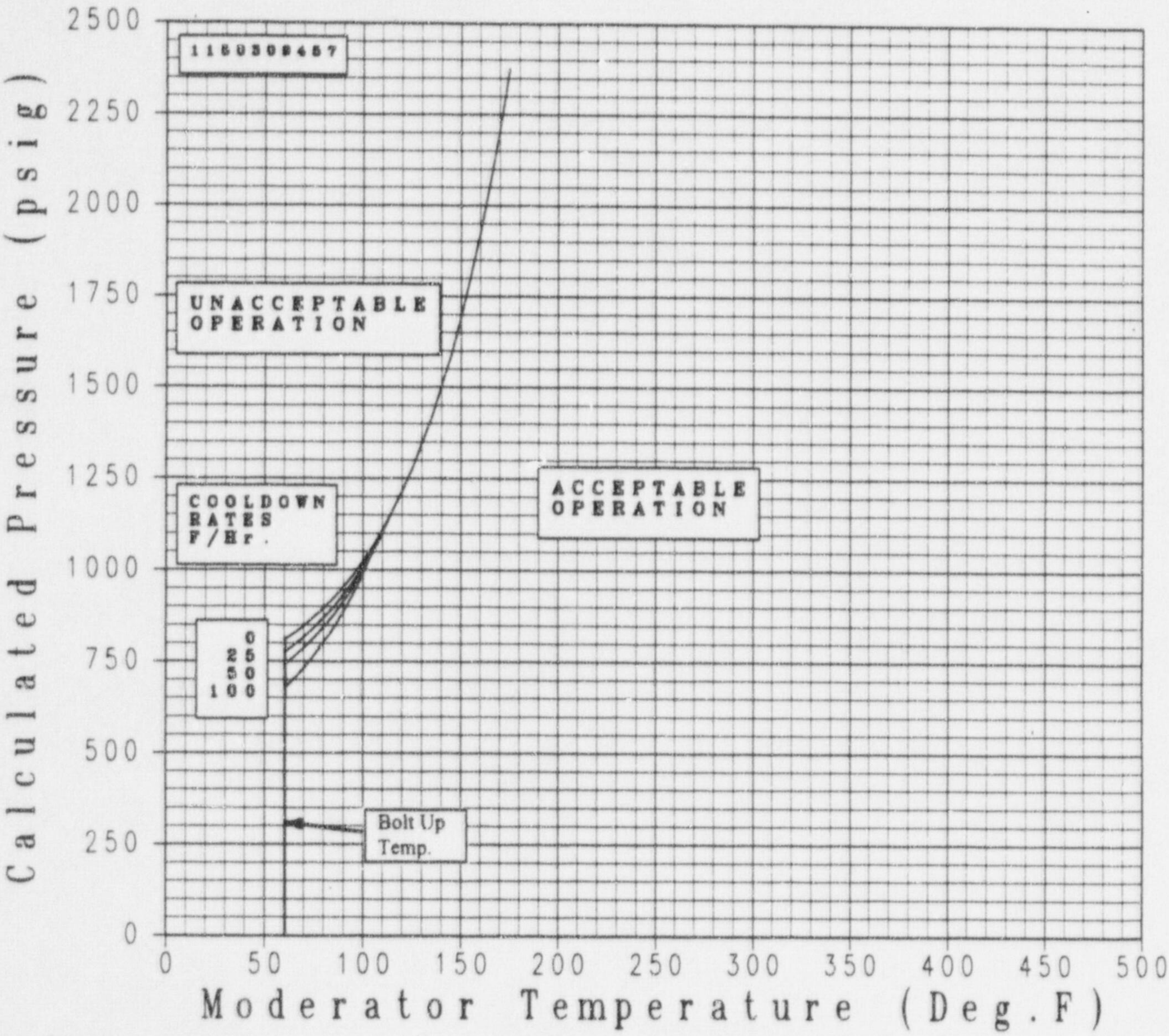

FIGURE 5-2 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 20 EFPY (Without Margins for Instrumentation Errors)

TABLE 5-1

V. C. Summer Unit 1 Heatup Data at 20 EFPY

(Without Margins for Instrumentation Errors)

Configuration # 11503[illegible]57									
50 °F/hr		Critical. Limit		100 °F/hr		Critical. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	158	0	60	0	158	0	140	2000
60	809	158	828	60	809	158	828	158	2485
65	828	158	921	65	828	158	906		
70	848	158	930	70	848	158	899		
75	870	158	945	75	870	158	897		
80	895	158	967	80	895	158	900		
85	921	158	995	85	897	158	908		
90	930	158	1029	90	897	158	920		
95	945	158	1068	95	897	158	937		
100	967	160	1113	100	900	160	959		
105	995	165	1163	105	908	165	985		
110	1029	170	1220	110	920	170	1016		
115	1068	175	1284	115	937	175	1053		
120	1113	180	1355	120	959	180	1094		
125	1163	185	1434	125	985	185	1142		
130	1220	190	1521	130	1016	190	1195		
135	1284	195	1618	135	1053	195	1255		
140	1355	200	1725	140	1094	200	1322		
145	1434	205	1843	145	1142	205	1397		
150	1521	210	1974	150	1195	210	1481		
155	1618	215	2119	155	1255	215	1574		
160	1725	220	2279	160	1322	220	1676		
165	1843	225	2455	165	1397	225	1791		
170	1974			170	1481	230	1917		
175	2119			175	1574	235	2056		
180	2279			180	1676	240	2211		
185	2455			185	1791	245	2381		
				190	1917				
				195	2056				
				200	2211				
				205	2381				

TABLE 5-2

V. C. Summer Unit 1 Cooldown Data at 20 EFPY

(Without Margins for Instrumentation Errors)

Configuration #	1150309457						
Steady State		25 °F/hr		50 °F/hr		100 °F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	809	60	774	60	740	60	675
65	828	65	794	65	762	65	702
70	848	70	817	70	787	70	731
75	870	75	841	75	814	75	764
80	895	80	868	80	843	80	801
85	922	85	898	85	876	85	841
90	953	90	932	90	913	90	886
95	986	95	968	95	953	95	935
100	1023	100	1009	100	998	100	990
105	1064	105	1054	105	1048	105	1051
110	1109	110	1104	110	1103		
115	1159	115	1159				
120	1214						
125	1275						
130	1343						
135	1417						
140	1499						
145	1590						
150	1691						
155	1802						
160	1925						
165	2060						
170	2210						
175	2376						

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1,-2

LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F

3/4T, 94°F

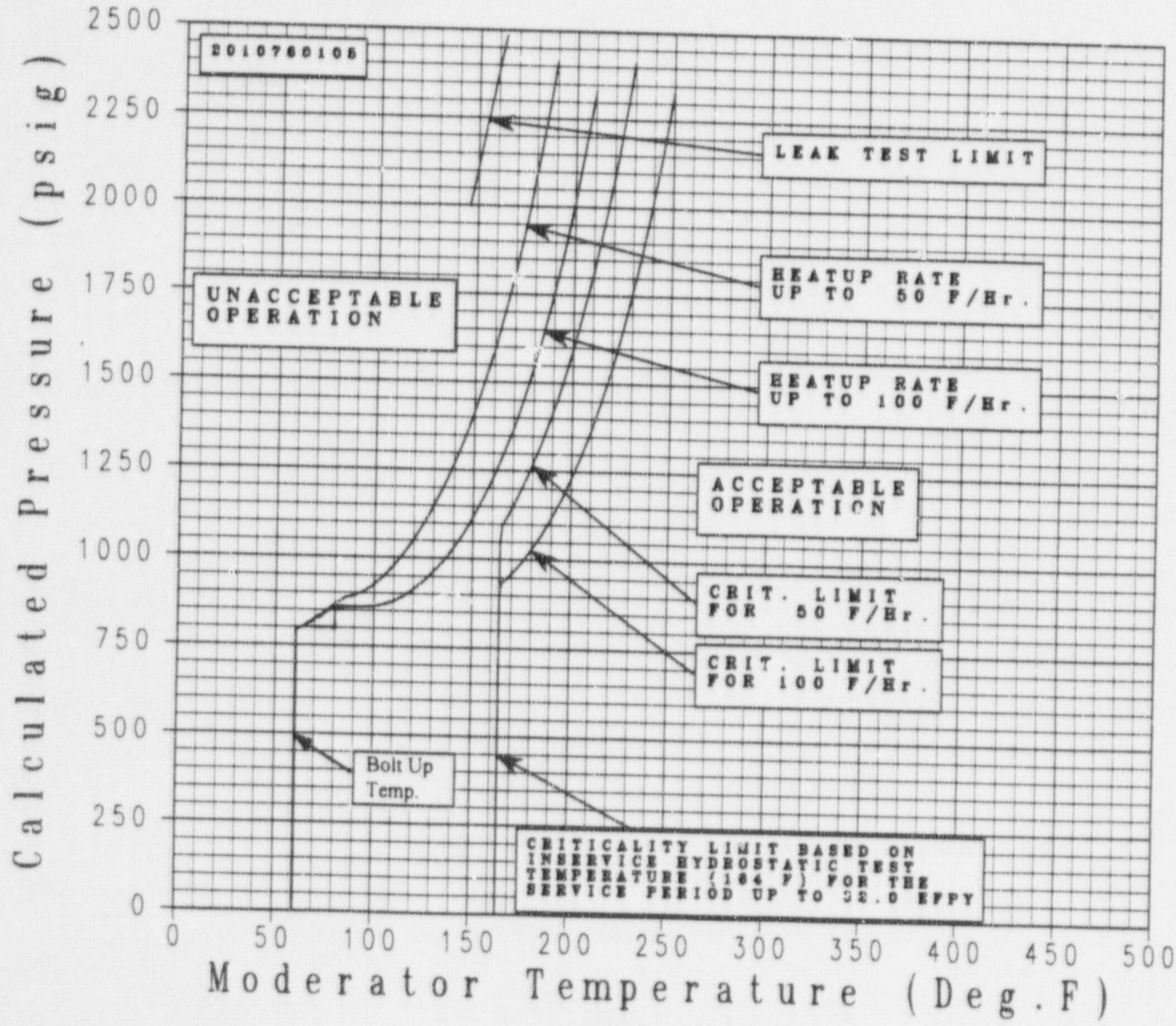

FIGURE 5-3 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 50 and 100°F/hr) Applicable to 32 EFPY (Without Margins for instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1,-2

LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F

3/4T, 94°F

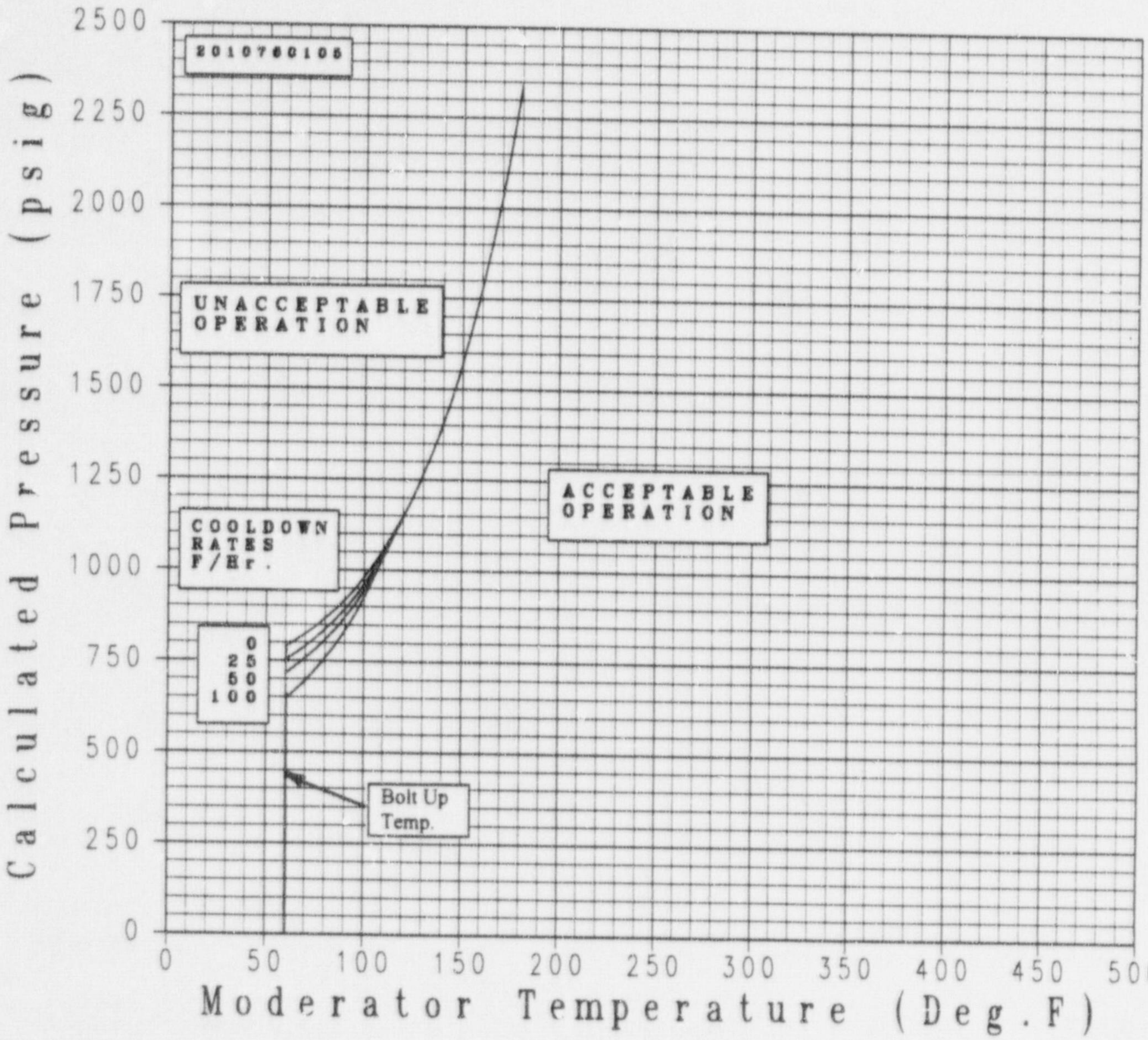

FIGURE 5-4 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY (Without Margins for Instrumentation Errors)

TABLE 5-3

V. C. Summer Unit 1 Heatup Data at 32 EFPY

(Without Margins for Instrumentation Errors)

Configuration # 2010760105									
50 °F/hr		Critical. Limit		100 °F/hr		Critical. limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	164	0	60	0	164	0	146	2000
60	789	164	806	60	789	164	806	164	2485
65	806	164	887	65	806	164	873		
70	824	164	893	70	824	164	864		
75	844	164	906	75	844	164	861		
80	866	164	925	80	861	164	863		
85	887	164	949	85	861	164	868		
90	893	164	978	90	861	164	878		
95	906	164	1013	95	861	164	893		
100	925	164	1052	100	863	164	911		
105	949	165	1097	105	868	165	934		
110	978	170	1147	110	878	170	961		
115	1013	175	1204	115	893	175	992		
120	1052	180	1267	120	911	180	1029		
125	1097	185	1336	125	934	185	1070		
130	1147	190	1414	130	961	190	1117		
135	1204	195	1500	135	992	195	1170		
140	1267	200	1595	140	1029	200	1230		
145	1336	205	1700	145	1070	205	1296		
150	1414	210	1816	150	1117	210	1370		
155	1500	215	1944	155	1170	215	1452		
160	1595	220	2085	160	1230	220	1543		
165	1700	225	2242	165	1296	225	1644		
170	1816	230	2415	170	1370	230	1755		
175	1944			175	1452	235	1879		
180	2085			180	1543	240	2016		
185	2242			185	1644	245	2167		
190	2415			190	1755	250	2334		
				195	1879				
				200	2016				
				205	2167				
				210	2334				

TABLE 5-4

V. C. Summer Unit 1 Cooldown Data at 32 EFPY

(Without Margins for Instrumentation Errors)

Configuration # 2010760105							
Steady State		25 °F/hr		50 °F/hr		100 °F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0
60	789	60	753	60	717	60	646
65	806	65	770	65	736	65	670
70	824	70	790	70	757	70	696
75	844	75	812	75	781	75	725
80	866	80	836	80	808	80	757
85	890	85	862	85	837	85	792
90	917	90	892	90	869	90	832
95	946	95	925	95	905	95	876
100	979	100	961	100	945	100	925
105	1015	105	1000	105	989	105	979
110	1056	110	1044	110	1037	110	1039
115	1100	115	1093	115	1091		
120	1149	120	1147				
125	1203						
130	1263						
135	1329						
140	1402						
145	1482						
150	1571						
155	1670						
160	1779						
165	1899						
170	2032						
175	2179						
180	2341						

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 1996 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure", December 1996.

4 ASME Code Case N-640, "Alternative Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February.

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

7 WCAP-12867, "Analysis of Capsule X from the South Carolina Electric & Gas Company Virgil C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program", J. M. Chicots, et al., March 1991.

8 WCAP-15101, "Analysis of Capsule W from the South Carolina Electric & Gas V. C. Summer Unit 1 Reactor Vessel Radiation Surveillance Program", T. J. Laubham, et al., September 1998.

9 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

10 WCAP-15102, Rev. 0, "V. C. Summer Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", S. Spragg, et. al., September 1998.