DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION

ATTACHMENT 5

WCAP-15201

MCGUIRE UNIT 2

HEATUP AND COOLDOWN LIMIT CURVES

FOR NORMAL OPERATION USING CODE CASE N-640

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15201

McGuire Unit 2 Heatup and Cooldown Limit Curves for Normal Operation Using Code Case N-640

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PREFACE

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature limit curves for normal operation of the McGuire Unit 2 reactor vessel. These curves were generated based on the latest available reactor vessel information (Capsule W analysis, WCAP-14799[1] and the latest Pressure-Temperature (P-T) Limit Curves from WCAP-14868[2]).

The McGuire Unit 2 heatup and cooldown pressure-temperature limit curves at 34 EFPY have been updated based on the use of the ASME Code Case N-640[3] which allows the use of the K_{Ic} methodology.

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."[4] 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.

2 PURPOSE

The Duke Power Company contracted Westinghouse to regenerate the 34 EFPY heatup and cooldown curves documented in WCAP-14868[2] and add two new heatup rates (80 and 100°F/hr.) using K_{Ic} in place of K_{IR} for the calculation of the stress intensity factors. The heatup and cooldown curves from WCAP-14868 were generated without margins for instrumentation errors and 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[5].

The purpose of this report is to document the generation of new 34 EFPY P-T limit curves utilizing the K_{Ic} methodology[3]. The P-T curves are developed with the identical adjust reference temperature (ART) values used in WCAP-14868. This report includes all the original text and tables from WCAP-14868 with appropriate changes corresponding to K_{Ic}, along with a justification for relaxing the flange temperature requirement of Appendix G to 10CFR Part 50 based on the use of K_{Ic} methodology rather than the K_{Ia} methodology. The use of K_{Ic} and relaxation of the flange temperature requirement will add substantial pressure margin to the heatup and cooldown curves documented in WCAP-14868. This increase in allowable pressure is presented in Section 6 of this report.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

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 the ASME Appendix G to Section XI[3 & 6]. The K_{Ic} curve is given by the following equation:

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

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 Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

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

where,

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

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

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

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

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
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 $2 \le \sqrt{t} \le 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

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

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

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953\text{x}10^{-3}$ x CR x $t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753\text{x}10^{-3}$ x HU x $t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. 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).

(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:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

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

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

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:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

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, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code[13] with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just 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 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_{Ic} at the 1/4T 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 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_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} 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_{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 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.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3107 psi), which is 621 psig for McGuire Unit 2 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 120°F and the pressure limitation of 20 percent of hydrotest pressure were developed using K_{Ia} fracture toughness, in the mid 1970's.

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 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}) here to develop equivalent flange requirements.

The geometry of the closure head flange region for a typical Westinghouse four loop plant reactor vessel such as McGuire Unit 2 reactor vessel is shown in Figure 1. The stresses in this region are highest near the outside of the head. Therefore, an outside reference flaw of 25 percent of the wall thickness parallel to the dome to flange weld (i.e. in the direction of the 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 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$\sqrt{\text{in}}$., even for postulated flaws up to 50 percent of the wall thickness. For reference flaw, with the safety factor of two, the applied stress intensity factor is 85.15 ksi$\sqrt{\text{in}}$. at 25 percent of the wall thickness.

The determination of the boltup, or flange requirement, is shown in Figure 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 $T - RT_{NDT} = 100°F$, 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 $T - RT_{NDT} = 120°F$. This corresponds to a K_{ia} (with a safety factor of 2) of 98 ksi$\sqrt{\text{in}}$. Note that the use of K_{Ic} curve to determine this requirement results in a revised requirement of $T - RT_{NDT} = 45°F$, as seen in Figure 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 1°F (Per Appendix B of Ref. 11) occurs in the closure head flange of the McGuire Unit 2 reactor vessel, so the minimum allowable temperature of this region is 46°F at pressures greater than 621 psig with no margins for instrument uncertainties.

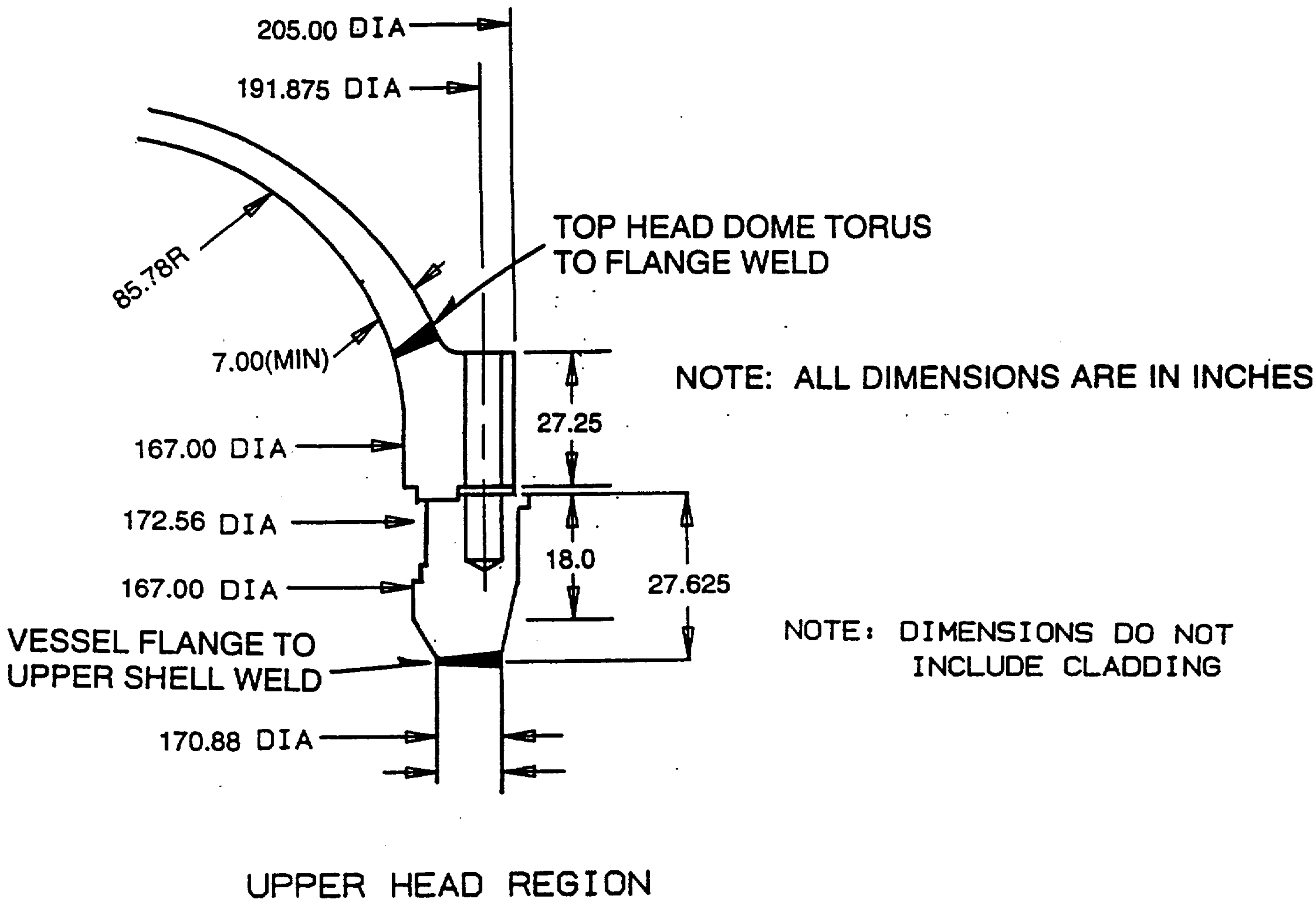

Figure 1 Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

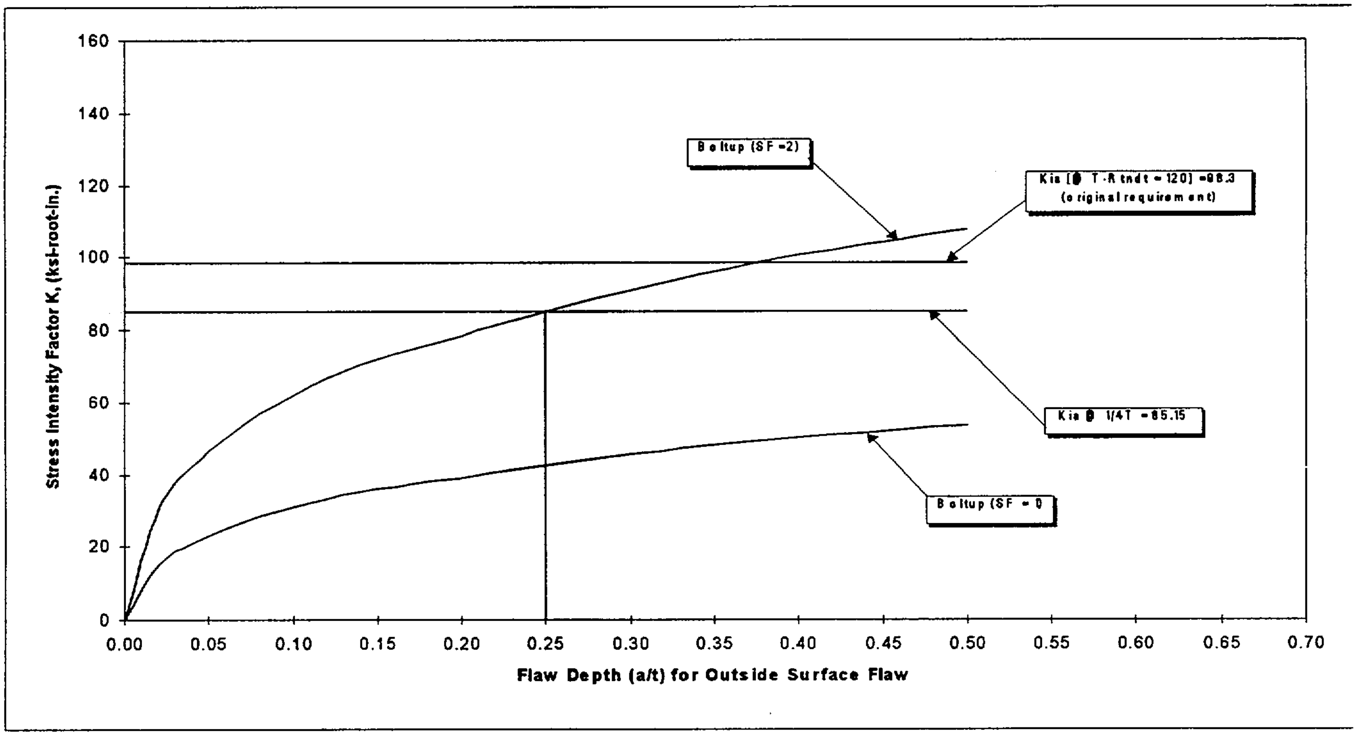

Figure 2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

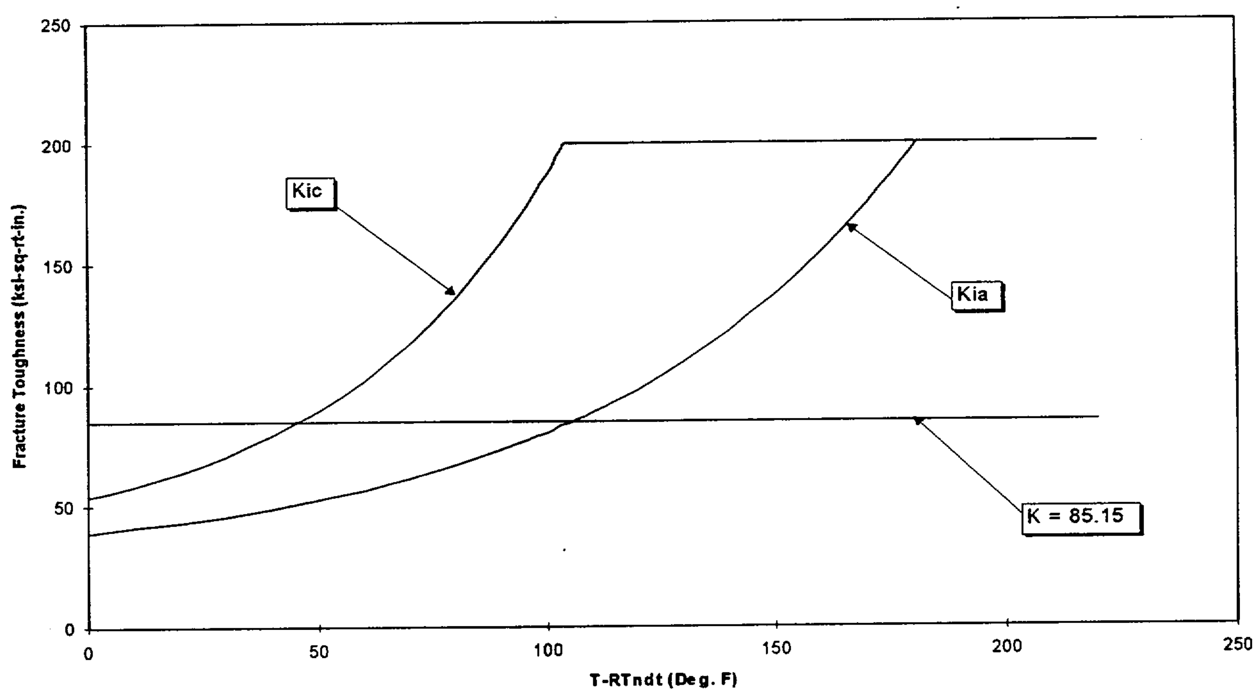

Figure 3 Determination of Boltup Requirement, Using K_{Ic}

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:

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \tag{7}$$

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[7]. 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:

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

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.

$$f_{(\text{depth } x)} = f_{\text{surface}} * e^{(-0.24x)} \tag{9}$$

where x inches (vessel beltline thickness is 8.465 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections as a part of WCAP-14799 and are also presented in a condensed version in Table 1 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Table 1 contains the calculated vessel surface fluences 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 McGuire Unit 2 reactor vessel. Additionally, the surveillance capsule fluence values are presented in Table 2.

TABLE 1
Summary of the Peak Pressure Vessel Neutron Fluence Values used for the Calculation of ΔRT Values
(n/cm^2, E > 1.0 MeV)

EFPY	Surface*	¼ T	¾ T
34	1.93 x 10^{19}	1.16 x 10^{19}	4.21 x 10^{18}

* Clad/Base Metal Interface

TABLE 2
Measured Integrated Neutron Exposure of the McGuire Unit 2 Surveillance Capsules

Capsule	Fluence
V	3.268 x 10^{18} n/cm^2 , (E > 1.0 MeV)
X	1.406 x 10^{19} n/cm^2 , (E > 1.0 MeV)
U	1.962 x 10^{19} n/cm^2 , (E > 1.0 MeV)
W	2.969 x 10^{19} n/cm^2 , (E > 1.0 MeV)
Z*	2.348 x 10^{19} n/cm^2 , (E > 1.0 MeV)
Y*	1.967 x 10^{19} n/cm^2 , (E > 1.0 MeV)

* These two capsules, Z and Y, are designated as standby capsules. These capsules were removed from the reactor vessel and the specimens placed in storage due to their high lead factors. Since the specimens were not tested they will not be used in the calculation of chemistry factors. This table was taken in its entirety from Reference 2.

Margin is calculated as, $M = 2\sqrt{\sigma_i^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, is σ_i 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 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 0.5 times the mean value of ΔRT_{NDT}.

Contained in Table 3 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials[1]. These measured shift values were obtained using CVGRAPH, Version 4.1[9], which is a hyperbolic tangent curve-fitting program.

TABLE 3*

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
Intermediate Shell Forging 05 (Axial Orientation)	V	58.64°F
	X	91.12°F
	U	84.14°F
	W	130.33°F
Intermediate Shell Forging 05 (Tangential Orientation)	V	68.97°F
	X	98.28°F
	U	91.18°F
	W	102.03°F
Surveillance Program Weld Metal	V	38.51°F
	X	35.93°F
	U	23.81°F
	W	43.76°F

* Table 3 was taken in its entirety from Reference 2.

Table 4 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 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 6. Table 5 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 6.

TABLE 4***

Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)*	Ni(%)*	Initial RT_{NDT}**
Closure Head Flange	- -	- -	1°F
Vessel Flange	- -	- -	-4°F
Intermediate Shell Forging 05	0.153	0.793	-4°F
Lower Shell Forging 04	0.15	0.88	-30°F
Circumferential Weld	0.039	0.724	-68°F

* From Duke Power response to Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity"[10].

** From Appendix B of Reference 11.

*** Table 4 was taken in its entirety from Reference 2.

TABLE 5[5]

Calculation of Chemistry Factors using McGuire Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f[1]	FF[2]	ΔRT_{NDT}[3]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell	V	0.3268	0.692	58.64	40.6	0.479
Forging 05	X	1.406	1.09	91.12	99.3	1.19
	U	1.962	1.18	84.14	99.3	1.39
(Axial)	W	2.969	1.29	130.33	168.1	1.66
Intermediate Shell	V	0.3268	0.692	68.97	47.7	0.479
Forging 05	X	1.406	1.09	98.28	107.1	1.19
	U	1.962	1.18	91.18	107.6	1.39
(Tangential)	W	2.969	1.29	102.03	131.6	1.66
				SUM	801.3	9.44
	$CF_{Forging\ 05} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (801.3) \div (9.44) = 84.9°F$					
	V	0.3268	0.692	38.51	26.6	0.479
Circumferential	X	1.406	1.09	35.93	39.2	1.19
Weld Seam[4]	U	1.962	1.18	23.81	28.1	1.39
	W	2.969	1.29	43.76	56.5	1.66
				SUM	150.49	4.72
	$CF_{S/P\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (150.4) \div (4.72) = 31.9°F$					

Notes:

(1) f = Measured fluence from capsule W dosimetry analysis results[1], (x 10^{19} n/cm^2, E > 1.0 MeV).

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

(3) ΔRT_{NDT} values are measured[1].

(4) The McGuire Unit 2 surveillance weld was fabricated using the same weld wire (Ht. # 895075) and flux type (Grau L.O. flux) as the intermediate to lower shell girth weld. Per chemistry data presented in Table 4 of Reference 10, the average copper and nickel weight percent of the McGuire Unit 2 surveillance weld metal is 0.036% and 0.736%, respectively and the overall combined average copper and nickel weight percent for weld wire (Ht. # 895075) and flux type (Grau L.O. flux) is 0.039% copper and 0.724% nickel. Hence, there is no clear evidence that the copper and nickel content of the surveillance weld differs from that of the vessel weld. In addition, the limiting beltline material of the McGuire Unit 2 reactor vessel is the lower shell forging 04 and not the low copper weld metal. Therefore, the use of the ratio procedure is not warranted and will not be applied in these calculations.

(5) Table 5 was taken in its entirety from Reference 2.

TABLE 6*

Summary of the McGuire Unit 2 Reactor Vessel Beltline Material Chemistry Factors

Material	Reg. Guide 1.99, Rev. 2 Position 1.1 CF's	Reg. Guide 1.99, Rev. 2 Position 2.1 CF's
Intermediate Shell Forging 05	117°F	84.9°F
Lower Shell Forging 04	115.8°F	---
Circumferential Weld Seam	52.7°F	31.9°F

* Table 6 was taken in its entirety from Reference 2.

Contained in Table 7 is a summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the McGuire Unit 2 reactor vessel beltline materials.

TABLE 7

Summary of the Calculated Fluence Factors Used for the Generation of the 34 EFPY Heatup and Cooldown Curves

EFPY	1/4T FF	3/4T FF
34	1.041	0.760

The adjusted reference temperature (ART) must be calculated for 34 EFPY for each beltline material at the 1/4T and 3/4T locations. In addition, ART values must be calculated per Regulatory Guide 1.99, Revision 2, Position 1.1 and 2.1

Contained in Table 8 and 9 are the calculations of the 34 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 34 EFPY

Material	RG 1.99, R2 Method	CF (°F)	FF	ΔRT_{NDT} (°F)	Margin (°F)	IRT_{NDT}(1) (°F)	ART(2) (°F)
Intermediate Shell Forging 05	Position 1.1	117.2	1.041	122.0	34	-4	152.0
(Heat 526840)	Position 2.1	84.9	1.041	88.4	17	-4	101.4
Circumferential Weld Seam W05	Position 1.1	52.7	1.041	54.9	54.9	-68	41.8
	Position 2.1	31.9	1.041	33.2	28	-68	-6.8
Lower Shell Forging 04 (Heat 411337/11)	Position 1.1	115.8	1.041	120.5	34	-30	**124.5**

Notes:
(1) Initial RT_{NDT} values measured values.
(2) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 34 EFPY

Material	RG 1.99, R2 Method	CF (°F)	FF	ΔRT_{NDT} (°F)	Margin (°F)	IRT_{NDT}(1) (°F)	ART(2) (°F)
Intermediate Shell Forging 05	Position 1.1	117.2	0.760	89.1	34	-4	119.1
(Heat 526840)	Position 2.1	84.9	0.760	64.5	17	-4	77.5
Circumferential Weld Seam W05	Position 1.1	52.7	0.760	40.1	40.1	-68	12.2
	Position 2.1	31.9	0.760	24.2	24.2	-68	-19.6
Lower Shell Forging 04 (Heat 411337/11)	Position 1.1	115.8	0.760	88.0	34	-30	**92.0**

Notes:
(1) Initial RT_{NDT} values measured values.
(2) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)

The lower shell forging 04 (Heat # 411337/11) is the limiting beltline material for all heatup and cooldown curves to be generated. Contained in Table 10 is a summary of the limiting ARTs to be used in the generation of the McGuire Unit 2 reactor vessel heatup and cooldown curves.

TABLE 10
Summary of the Limiting ART Values Used in the Generation of the McGuire Unit 2 Heatup/Cooldown Curves

EFPY	1/4T Limiting ART	3/4T Limiting ART
34	124.5°F	92.0°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[13] discussed in Sections 3.0 and 4.0 of this report. The pressure difference between the wide-range pressure transmitter and the limiting beltline region has not been accounted for in the pressure-temperature limit curves generated for normal operation.

Figures 4 and 5 present the heatup curves without margins for possible instrumentation errors using heatup rates of 60, 80 and 100°F/hr applicable for the first 34 EFPY. Figure 6 presents the cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 34 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 through 3. 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 Figures 4 and 5. 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 Code Case N-640[3] (approved in February of 1999) as follows:

$$1.5\ K_{Im} < K_{Ic}$$

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 5. 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 3.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperature for the in service hydrostatic leak tests for the McGuire Unit 2 reactor vessel at 34 EFPY is 185°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 4 through 6 define all of the above limits for ensuring prevention of nonductile failure for the McGuire Unit 2 reactor vessel.

The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 4 through 6 are presented in Tables 11 and 12. As seen by comparing these results to that from Tables 5-3 and 5-4 of WCAP-14868, there is a minimum increase in pressure of 169 psig (@ lowest temperature) when K_{Ic} is used in the calculation of heatup and cooldown limit curves.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 124.5°F
3/4T, 92.0°F

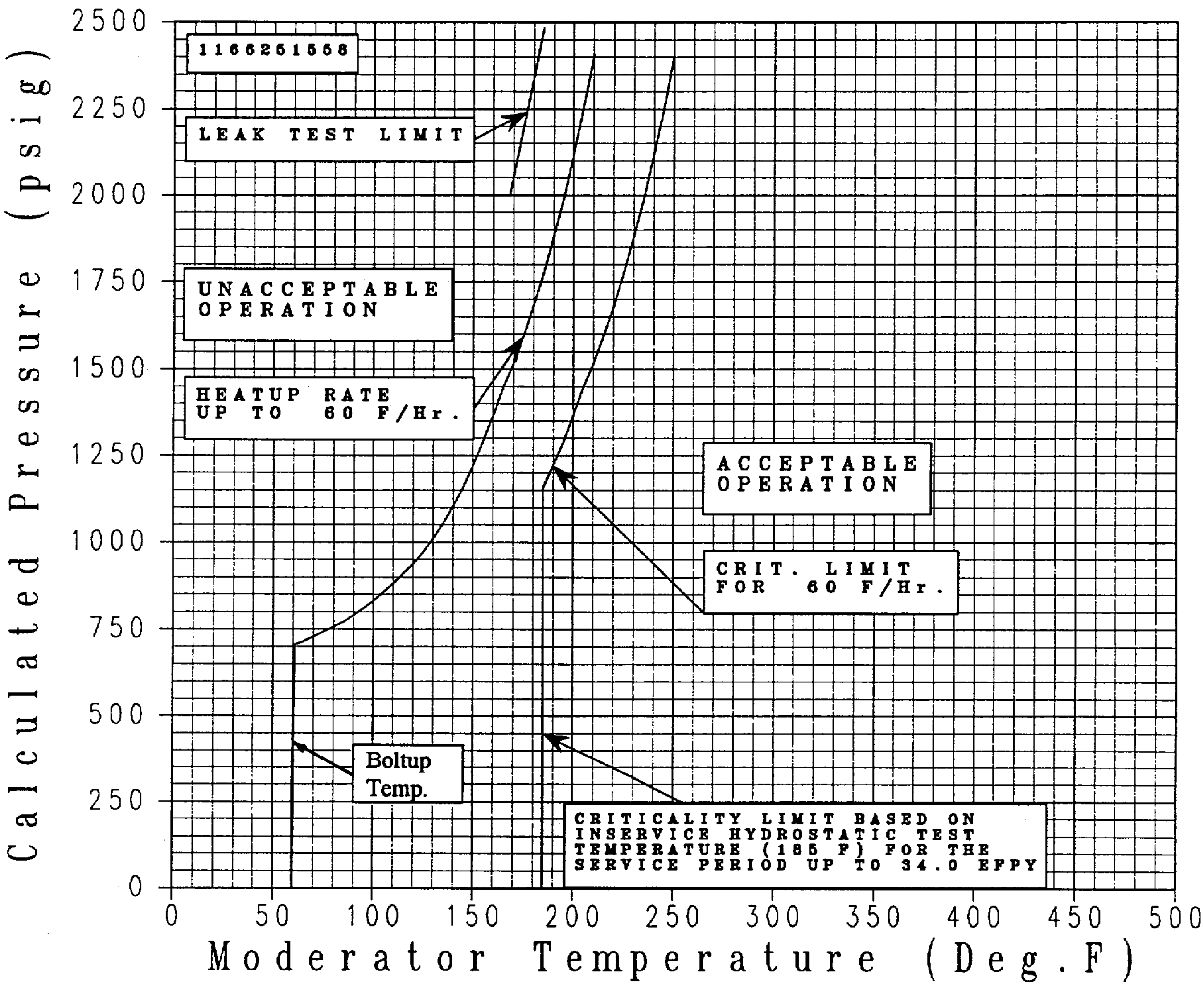

Figure 4 **McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates up 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors)**

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 124.5°F
3/4T, 92.0°F

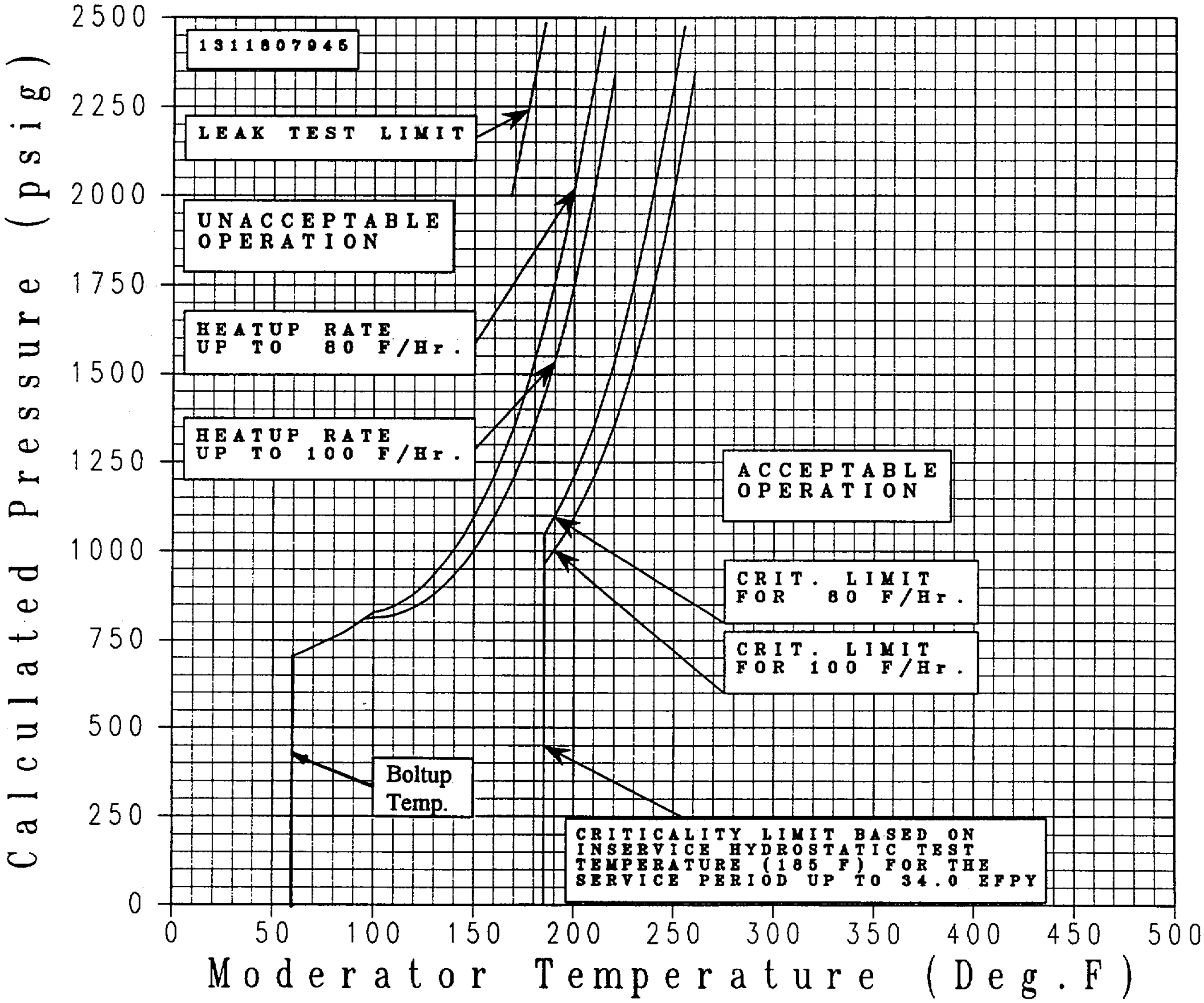

Figure 5 **McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 and 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors)**

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 124.5°F
3/4T, 92.0°F

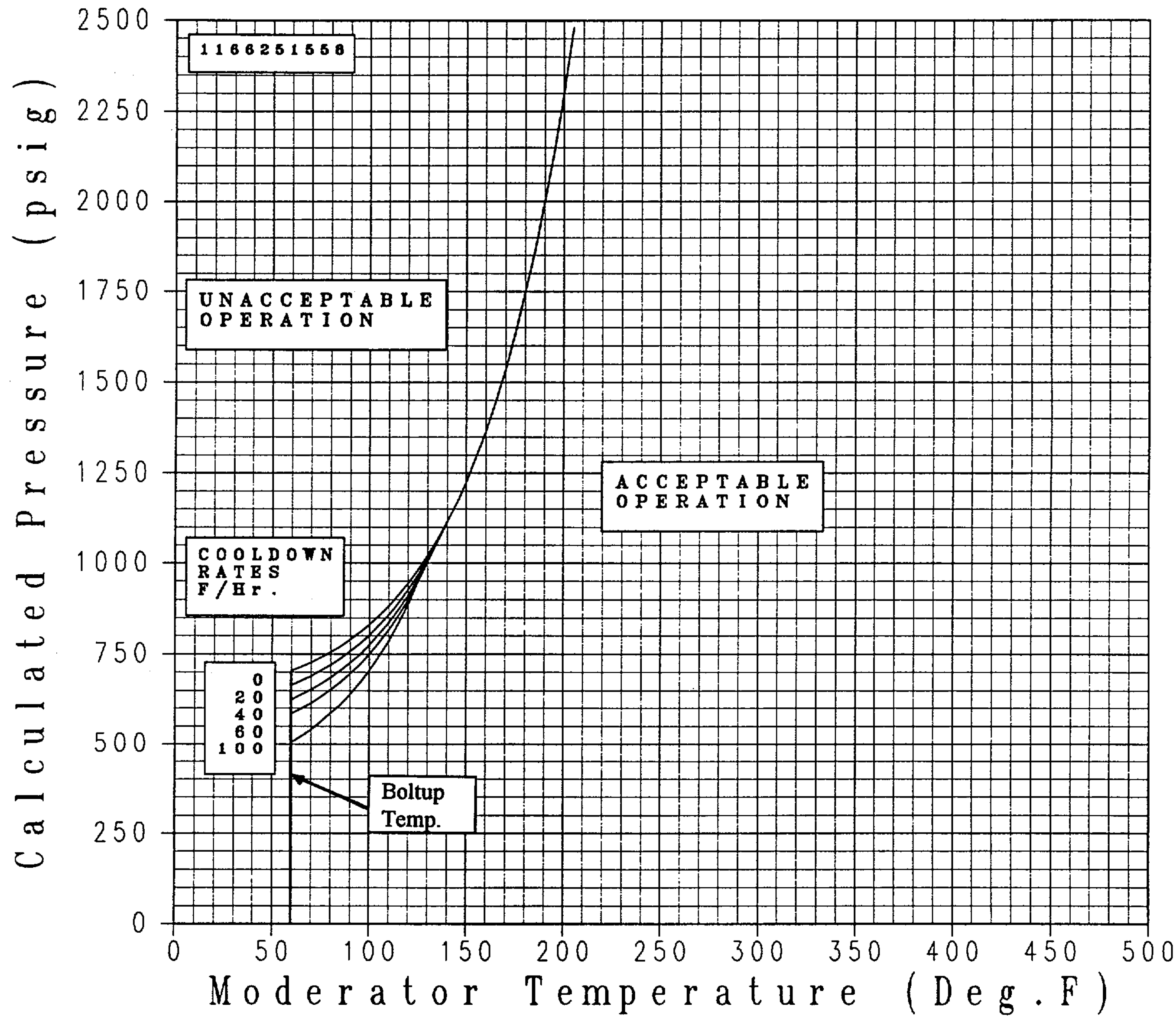

Figure 5 **McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors)**

TABLE 11
34 EFPY Heatup Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Heatup		Curves									
60 Heatup		60 Critical Limit		80 Heatup		80 Critical Limit		100 Heatup		100 Critical Limit	
T	P	T	P	T	P	T	P	T	P	T	P
60	0	103	0	60	0	103	0	60	0	103	0
60	705	185	716	60	705	185	716	60	705	185	716
65	716	185	772	65	716	185	772	65	716	185	772
85	772	185	790	85	772	185	790	85	772	185	790
90	790	185	810	90	790	185	810	90	790	185	810
95	810	185	832	95	810	185	829	95	810	185	815
100	832	185	856	100	829	185	836	100	815	185	816
105	856	185	883	105	836	185	847	105	816	185	821
110	883	185	910	110	847	185	862	110	821	185	830
115	910	185	938	115	862	185	882	115	830	185	843
120	938	185	972	120	882	185	906	120	843	185	860
125	972	185	1010	125	906	185	934	125	860	185	880
130	1010	185	1054	130	934	185	967	130	880	185	905
135	1054	185	1103	135	967	185	1005	135	905	185	933
140	1103	185	1159	140	1005	185	1048	140	933	185	967
145	1159	190	1221	145	1048	190	1097	145	967	190	1005
150	1221	195	1290	150	1097	195	1151	150	1005	195	1048
155	1290	200	1366	155	1151	200	1213	155	1048	200	1097
160	1366	205	1447	160	1213	205	1281	160	1097	205	1151
165	1447	210	1518	165	1281	210	1357	165	1151	210	1213
170	1518	215	1595	170	1357	215	1441	170	1213	215	1281
175	1595	220	1680	175	1441	220	1535	175	1281	220	1357
180	1680	225	1774	180	1535	225	1639	180	1357	225	1441
185	1774	230	1877	185	1639	230	1754	185	1441	230	1535
190	1877	235	1991	190	1754	235	1881	190	1535	235	1638
195	1991	240	2117	195	1881	240	2021	195	1638	240	1753
200	2117	245	2256	200	2021	245	2177	200	1753	245	1880
205	2256	250	2409	205	2177	250	2322	205	1880	250	2020
210	2409			210	2322	255	2476	210	2020	255	2175
				215	2476			215	2175	260	2346
								220	2346		

Leak Test Limit	
T	P
168	2000
185	2485

TABLE 12
34 EFPY Cooldown Curve Data Points Using 1996 App. G
(without Uncertainties for Instrumentation Errors)

Cooldown Curves				Configuration # 1166251558					
Steady State		20F		40F		60F		100F	
T	P	T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0	60	0
60	705	60	666	60	626	60	586	60	507
65	716	65	678	65	639	65	600	65	524
70	728	70	691	70	653	70	616	70	542
75	741	75	705	75	669	75	633	75	563
80	756	80	721	80	686	80	652	80	586
85	772	85	738	85	705	85	673	85	611
90	790	90	758	90	726	90	696	90	639
95	810	95	780	95	750	95	722	95	671
100	832	100	803	100	776	100	751	100	706
105	856	105	830	105	805	105	782	105	745
110	883	110	859	110	837	110	818	110	788
115	913	115	892	115	873	115	857	115	835
120	945	120	927	120	912	120	900	120	888
125	981	125	967	125	956	125	948	125	947
130	1021	130	1011	130	1004	130	1001	130	1012
135	1066	135	1059	135	1057	135	1060		
140	1114	140	1113						
145	1168								
150	1228								
155	1294								
160	1366								
165	1447								
170	1536								
175	1634								
180	1742								
185	1862								
190	1995								
195	2141								
200	2303								
205	2482								

6 REFERENCES

1. WCAP-14799, "Analysis of Capsule W from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program," Ed Terek, et al., Dated March 1997.

2. WCAP-14868, "McGuire Unit 2 Heatup and Cooldown Limit Curves For Normal Operation", Ed Terek, Dated April, 1997.

3. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", 9/18/98.

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

5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

6. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure.", Dated 1989 & December 1995.

7. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."

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. CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.0, developed by ATI Consulting, March 1995.

10. Duke Power Response to NRC Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity" for McGuire Unit 2.

11. WCAP-13516, "Analysis of Capsule U from the Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program", J.M. Chicots, et. al., Dated October 1992.

12. WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," W. S. Hazelton, et al., April 1975.

13. EDRE-EMT-787, "Release of Program OPERLIM Version 4.3 for Production use under UNIX Operating System HP-UX 9.01", October 9, 1998.

14. WCAP-15117, "Analysis of Capsule V and the Dosimeters from Capsules U and X from the Duke Power Company Catawba Unit 1 Reactor Vessel Radiation surveillance Program", E. Terek, et.al., October 1998.

APPENDIX A

CREDIBILITY EVALUATION OF THE MCGUIRE UNIT 2 SURVEILLANCE MATERIAL

INTRODUCTION:

Regulatory Guide 1.99, Revision 2, describes general procedures acceptable to the NRC staff for calculating the effects of neutron radiation embrittlement of the low-alloy steels currently used for light-water-cooled reactor vessels. Position C.2 of Regulatory Guide 1.99, Revision 2, describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor vessel beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question.

To date there has been six surveillance capsules removed from the McGuire Unit 2 reactor vessel (Capsule Z & Y were analyzed for dosimetry only). To use these surveillance data sets, they must be shown to be credible. In accordance with the discussion of Regulatory Guide 1.99, Revision 2, there are five requirements that must be met for the surveillance data to be judged credible.

The purpose of this evaluation is to apply the credibility requirements of Regulatory Guide 1.99, Revision 2, to the McGuire Unit 2 reactor vessel surveillance data and determine if the McGuire Unit 2 surveillance data is credible. It should be noted here that the surveillance capsule weld data for McGuire Unit 2 was used in the credibility analysis for Catawba Unit 1[15] and was determined to be credible. Therefore no further evaluation is required on the weld metal.

EVALUATION:

Criterion 1: Materials in the capsules should be those judged most likely to be controlling with regard to radiation embrittlement.

The beltline region of the reactor vessel is defined in Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements", as follows:

> "the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

The McGuire Unit 2 reactor vessel consists of the following beltline region materials:

- Intermediate shell forging 05 (Heat # 526840)
- Lower shell forging 04 (Heat # 411337/11)
- The intermediate shell to lower shell girth weld (Heat number 895075, Flux Type Grau L.O.)

The McGuire Unit 2 surveillance program utilizes tangential and axial test specimens from intermediate shell forging 05.

At the time when the surveillance program material was selected it was believed that copper and phosphorus were the elements most important to embrittlement of reactor vessel steels. Intermediate shell forging 05 had the highest initial RT_{NDT}'s and the lowest initial USE of the two forgings in the beltline region. In addition, Forging 05 had the highest weight percent of copper and phosphorus. Hence, intermediate shell forging 05 was chosen for the surveillance program.

Based on the above discussion, the McGuire Unit 2 surveillance forging material meets the this criteria.

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

Plots of Charpy energy versus temperature for the unirradiated and irradiated condition are presented in Appendix C of Reference 1.

> Based on engineering judgement, the scatter in the data presented in these plots is small enough to permit the determination of the 30 ft-lb temperature and the upper shelf energy of the McGuire Unit 1 surveillance materials unambiguously. Hence, the McGuire Unit 2 surveillance program meets this criterion.

Criterion 3: When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The functional form of the least squares method as described in Regulatory Position 2.1 will be utilized to determine a best-fit line for this data and to determine if the scatter of these ΔRT_{NDT} values about this line is less than 17°F for the forging.

Following is the calculation of the best fit line as described in Regulatory Position 2.1 of Regulatory Guide 1.99, Revision 2.

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 is presented in Table D-1.

TABLE D-1
Best Fit Evaluation for McGuire Unit 2 Surveillance Forging Material

Base Material	**CF[b] (°F)**	**FF[b]**	**ΔRT_{NDT} [b] (30 ft-lb) (°F)**	**Best Fit[a] ΔRT_{NDT} (°F)**	**Scatter of ΔRT_{NDT} (°F)**	**< 17°F (Base Metal)**
Intermediate Shell	84.9	0.692	58.64	58.8	0.2	Yes
Forging 05	84.9	1.09	91.12	92.5	1.4	Yes
	84.9	1.18	84.14	100.2	16.1	Yes
(Axial)	84.9	1.29	130.33	109.5	-20.8	No
Intermediate Shell	84.9	0.692	68.97	58.8	-10.2	Yes
Forging 05	84.9	1.09	98.28	92.5	-5.8	Yes
	84.9	1.18	91.18	100.2	9.0	Yes
(Tangential)	84.9	1.29	102.03	109.5	7.5	Yes

NOTES:

(a) Best Fit Line Per Equation 2 of Reg. Guide 1.99 Rev. 2 Position 1.1.

(b) See Table 5 in this Report for CF, FF and measured ΔRT_{NDT}

The scatter of ΔRT_{NDT} values about the functional form of a best-fit line drawn as described in Regulatory Position 2.1 (Table D-1) is not less than 17°F for only one of eight points. Per the NRC Industry Meeting of February 12th and 13th 1998, the NRC provided guidance for this situation, where it was stated that when only one of six (or more) points was outside the scatter band, it would be consider credible data. Thus, this criteria is met for the surveillance forging data of the McGuire Unit 2 surveillance program.

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

The capsule specimens are located in the reactor between the core barrel and the vessel wall and are positioned opposite the center of the core. The test capsules are in baskets attached to the neutron pads. The location of the specimens with respect to the reactor vessel beltline provides assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25°F. Hence, this criteria is met.

Criterion 5: The surveillance data for the correlation monitor material in the capsule should fall within the scatter band of the data base for that material.

The McGuire Unit 2 Surveillance program does not contain correlation monitor material. Therefore, this criterion is not applicable to the McGuire Unit 2 surveillance program.

CONCLUSION:

Based on the preceding positive responses to all five criteria of Regulatory Guide 1.99, Revision 2, Section B, the McGuire Unit 2 forging surveillance data is credible.

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION

ATTACHMENT 6

ASME CODE CASE N-640 AND TECHNICAL BASIS

BC 98-379
ISI 94-004
Dec. '98

CASES OF ASME BOILER AND PRESSURE VESSEL CODE

Approval Date: February 26, 1999
See Numeric Index for expiration and any reaffirmation dates.

Case N-640
Alternative Reference Fracture Toughness for Development of P-T Limit Curves
Section XI, Division 1

Inquiry: May the reference fracture toughness curve K_{IC}, as found in Appendix A of Section XI, be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves?

Reply: It is the opinion of the Committee that the reference fracture toughness K_{IC} of Fig. A-4200-1 of Appendix A may be used in lieu of Fig. G-2210-1 in Appendix G for the development of P-T Limit Curves. When this Case is employed LTOP Systems shall limit the maximum pressure in the vessel to 100% of the pressure allowed by the the P-T Limit Curves.

TECHNICAL BASIS FOR REVISED P-T LIMIT CURVE METHODOLOGY

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Abstract

The startup and shutdown process for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate nine numbers of safety margins; one of which is a lower bound fracture toughness curve.

There are two lower bound fracture toughness curves available in Section XI, K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged.

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could, in fact, reduce overall plant safety. By opening up the operating window relative to the pump seal requirements, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin which exists with the revised methodology is very large, whether considered deterministically or from the standpoint of risk.

Changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures and/or fuel problems will decrease.

Introduction

The startup and shutdown process, as well as press testing, for an operating nuclear plant is controlled by pressure-temperature limit curves, which are developed based on fracture mechanics analysis. These limits are developed in Appendix G of Section XI, and incorporate four specific safety margins:

1. Large flaw, ¼ thickness
2. Safety factor = 2 on pressure stress for startup and shutdown
3. Lower bound fracture toughness
4. Upper bound adjusted reference temperature (RT_{NDT})

There are two lower bound fracture toughness curves available in Section XI, K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. The only change involved in this action is to change the fracture toughness curve used for development of P-T limit curves from K_{IA} to K_{IC}. The other margins involved with the process remain unchanged. There are a number of reasons why the limiting toughness in the Appendix G pressure-temperature limits should be changed from K_{IA} to K_{IC}.

Use of K_{IC} is More Technically Correct

The heat-up and cool-down process is a very slow one, with the fastest rate allowed being 100° per hour. The rate of change of pressure and temperature is often constant, so the stress is essentially constant in this case. Both the heat-up and cool-down and the pressure testing are essentially static processes. In fact, all operating transients (levels A, B, C and D) correspond to static loadings, with regard to fracture toughness.

The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{IA} should be used for the reactor pressure vessel is when a crack is running. This might happen during a PTS transient event, but not during heatup or cooldown. Therefore, use of the static toughness K_{IC} lower bound toughness would be more technically correct for development of P-T limit curves.

Use of Historically Large Margin No Longer Necessary

In 1974, when the Appendix G methodology was first codified, the use of K_{IA} (K_{IR} in the terminology of the time) to provide additional margin was thought to be necessary to cover uncertainties and a number of postulated but unquantified effects. Almost 25 years later, significantly more is known about these uncertainties and effects.

Flaw Size

With regard to flaw indications in reactor vessels, there have been no indications found at the inside surface of any operating reactor in the core region which exceed the acceptance standards of Section XI, in the entire 28 year history of Section XI. This is a particularly impressive conclusion when considering that core region inspections have been required to concentrate on the inner surface and near inner surface region since the implementation of Regulatory Guide 1.150. Flaws have been found, but all have been qualified as buried, or embedded.

There are a number of reasons why no surface flaws exist, and these are related to the fabrication and inspection practices for vessels. For the base metal and full penetration welds, a full volumetric examination and surface exam is required before cladding is applied, and these exams are repeated after cladding.

Further confirmation of the lack of any surface indications has recently been obtained by the destructive examination of portions of several commercial reactor vessels, for example the Midland vessel and the PVRUF vessel.

Fracture Toughness

Since the original formulation of the K_{IA} and K_{IC} curves, in 1972, the fracture toughness database has increased by more than an order of magnitude, and both K_{IA} and K_{IC} remain lower bound curves, as shown for example in Figure 1 for K_{IC}[1] compared to Figure 2, which is the original database[2].

In can be seen from Figure 1 that there are a few data points which fall just below the curve. Consideration of these points, as well as the (over 1500) points above the curve, leads to the conclusion that the K_{IC} curve is a lower bound for a large percentage of the data.

Local Brittle Zones

A third argument for the use of K_{IA} in the original version of Appendix G was based upon the concern that there could be a small, local brittle zone in the weld or heat-affected-zone of the base material that could pop-in and produce a dynamically moving cleavage crack. Therefore, the toughness property used to assess the moving crack should be related to dynamic or crack arrest conditions, especially for a ferritic pressure vessel steel showing distinct temperature and loading-rate (strain-rate) dependence. The dynamic crack should arrest at a ¼-T size, and any re-initiation should consider the effects of a minimum toughness associated with dynamic loading. This argument provided a rationale for assuming a ¼-T postulated flaw size and a lower bound fracture toughness curve considering dynamic and crack arrest loading. The K_{IR} curve in Appendix G of Section III, and the equivalent K_{IA} curve in Appendix A and Appendix G of Section XI provide this lower bound curve for high-rate loading (above any realistic rates in reactor pressure vessels during any accident condition) and crack arrest conditions. This argument, of course, relies upon the existence of a local brittle zone.

After over 30 years of research on reactor pressure vessel steels fabricated under tight controls, micro-cleavage pop-in has not been found to be significant. This means that researchers have not produced catastrophic failure of a vessel, component, or even a fracture toughness test specimen in the transition temperature regime. The quality of quenched, tempered, and stress-relieved nuclear reactor pressure vessel steels, that typically have a lower bainitic microstructure, is such that there may not be any local brittle zones that can be identified. Testing of some test specimens at ORNL has shown some evidence of early pop-ins for some simulated production weld metals, but the level of fracture toughness for these possible early initiations is within the data scatter for other ASTM-defined fracture toughness values (K_{IC} and/or K_{JC}). Therefore, it is time to remove the conservatism associated with this postulated condition and use the ASME Code lower bound K_{IC} curve directly to assess fracture initiation. This is especially true when the unneeded margin may in fact reduce overall plant safety.

Overall Plant Safety is Improved

The primary reason for making this change is to reduce the excess conservatism in the current Appendix G approach that could in fact reduce overall plant safety. Considering the impact of the change on other systems (such as pumps) and also on personnel exposure, a strong argument can be made that the proposed change will increase plant safety and reduce personnel exposure for both PWRs and BWRs.

Impact on PWRs:

By opening up the operating window relative to the pump seal requirements, as shown schematically in Figure 3, the chances of damaging the seals and initiating a small LOCA, a potential pressurized thermal shock (PTS) initiator, are reduced. Moreover, excessive shielding to provide an acceptable operating window with the current requirements can result in higher fuel peaking and less margin to fuel damage during an accident condition.

The proposed change also reduces the need for lock-out of the HPSI systems, which improves personnel and plant safety and reduces the potential for a radio-active release. Finally, challenges to the plant LTOP system and potential problems with reseating the valves would also be reduced.

Impact on BWRs:

The primary impact on the BWR will be a reduction in the pressure test temperature. BWRs use pump heat to reach the required pressure test temperatures. Several BWR plants are required to perform the pressure test at temperatures over 212°F under the current Appendix G criteria. The high test temperature poses several concerns: (i) pump cavitation and seal degradation, (ii) primary containment isolation is required and ECCS/safety systems have to be operational at temperatures in excess of 212°F, (iii) leak detection is difficult and more dangerous since the resulting leakage is steam and poses safety hazards of burns and exposure to personnel. The reduced test temperature eliminates these safety issues without reducing overall fracture margin.

Reactor Vessel Fracture Likelihood is Very Low

It has long been known that the P-T limit curve methodology is very conservative[3,4]. Changing the reference toughness to K_{Ic} will maintain a very high margin, as illustrated in Figure 4, for a pressurized water reactor. This figure shows a series of P-T curves developed for the same plant, but with different assumptions concerning flaw size, safety margin and fracture toughness.

The results shown in Figure 4 were obtained for a sample problem which was solved by several members of the Section XI working group on Operating Plant Criteria, for both PWR and BWR plants. The sample problem requires development of an operating P-T cooldown curve or the pressure test for an irradiated vessel. Two P-T curves were required, one using K_{IA} and the second using K_{IC}. In both cases the quarter thickness flaw was used, along with the appropriate safety factor on pressure.

To determine the margins (pressure ratios) that are included in these curves, a reference P-T curve was developed, using a best estimate (mean) K_{Ic} curve, and no safety factor on stress, along with a flaw depth of one inch. Typical results are shown in Table 1. Comparing the reference or best estimate curve with the two P-T curves calculated using code requirements, we see that there is a large margin on the allowable pressure, whether one uses K_{IA} or K_{Ic} limits in Appendix G.

For PWRs, another important contribution to the margin, which cannot be quantified, is the low temperature overpressure protection system (LTOP) which is operational in the low temperature range. The margins increase significantly for higher temperatures, as seen in Figure 4. .

Impact of the Change on P-T Curves

To show the effect that the proposed change would produce, a series of P-T limit curves were produced for a typical plant. These curves were produced using identical input information, with one curve using K_{IA} and the other using the proposed new approach, with K_{Ic}. Since the limiting conditions for the PWR (cool-down) and the BWR (pressure test) are different, separate evaluations were performed for PWRs and BWRs.

The results are shown in Figure 5 for a typical PWR cool-down transient.

Summary and Conclusions

Technology developed over the last 25 years has provided a strong basis for revising the ASME Section XI pressure-temperature limit curve methodology. The safety margin that exists with the revised methodology is still very large.

From the standpoint of risk, changing the methodology will result in an increase in the safety of operating plants, as the likelihood of pump seal failures, need for HPSI systems lock-out, LTOP system challenges and/or fuel margin problems, and personnel hazards and exposure will all decrease.

References

1. VanderSluys, W.A. and Yoon, K.K., "Transition Temperature Range Fracture Toughness in Ferritic Steels and Reference Temperature of ASTM", prepared for PVRC and BWOG, BAW 2318, Framatome Technologies, April 1998.

2. Marston, T.U., "Flaw Evaluation Procedures, Background and Application of ASME Section XI, Appendix A", EPRI Special Report NP-719-SR, August 1978.

3. Chirigos, J.N. and Meyer, T.A., "Influence of Material Property Variations on the Assessment of Structural Integrity of Nuclear Components", ASTM Journal of Testing and Evaluation, Vol. 6, No. 5, Sept. 1978, pp 289-295.

4. White Paper on Reactor Vessel Integrity Requirements for Level A and B conditions, prepared by Section XI Task Group on R.V. Integrity Requirements, EPRI TR-100251, January 1993.

Table 1

Summary of Allowable Pressures for

20 Degree/hour Cooldown of Axial Flaw

at 70 Degrees F and RT_{PTS} of 270 F

(Typical PWR Plant)

Type of Evaluation	Allowable Pressure* (psi)	Pressure Ratio
Appendix G with t/4 flaw and K_{Ia} Limit	420	1.00
Appendix G with t/4 flaw and K_{Ic} Limit	530	1.26
Reference 1 inch flaw for pressure, thermal, residual and cladding loads	1520	3.61
Reference 1 inch flaw for pressure, thermal and residual loads	1845	4.38
Reference 1 inch flaw for pressure and thermal loading only	2305	5.48

* Note: Comparable values of allowable pressure were calculated by the ASME Section XI Operating Plant Working Group Members from Westinghouse, Framatome Technologies and Oak Ridge National Laboratory

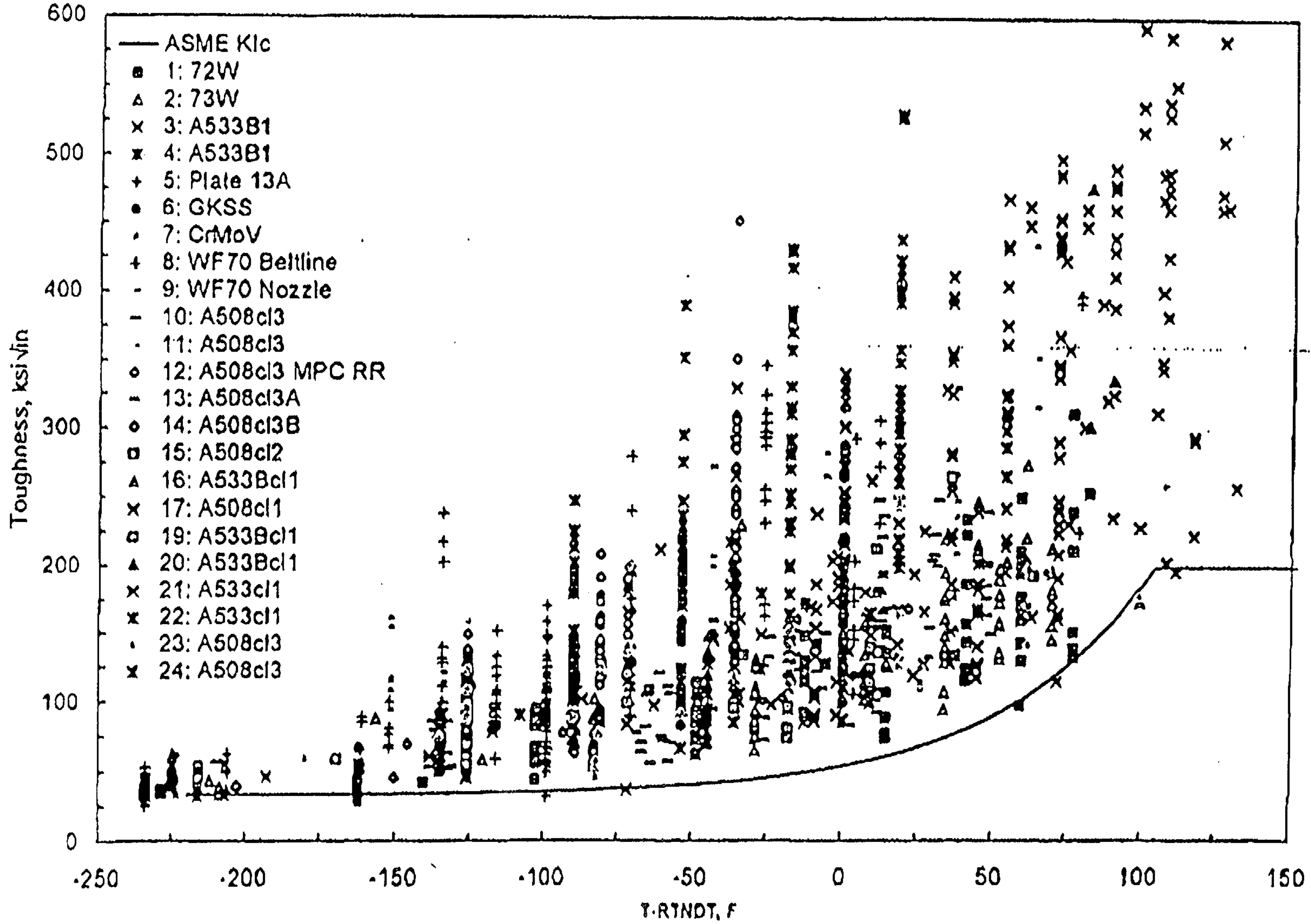

Figure1. Static Fracture Toughness Data (K_{Ic}) Now Available, Compared to K_{Ic} [1]

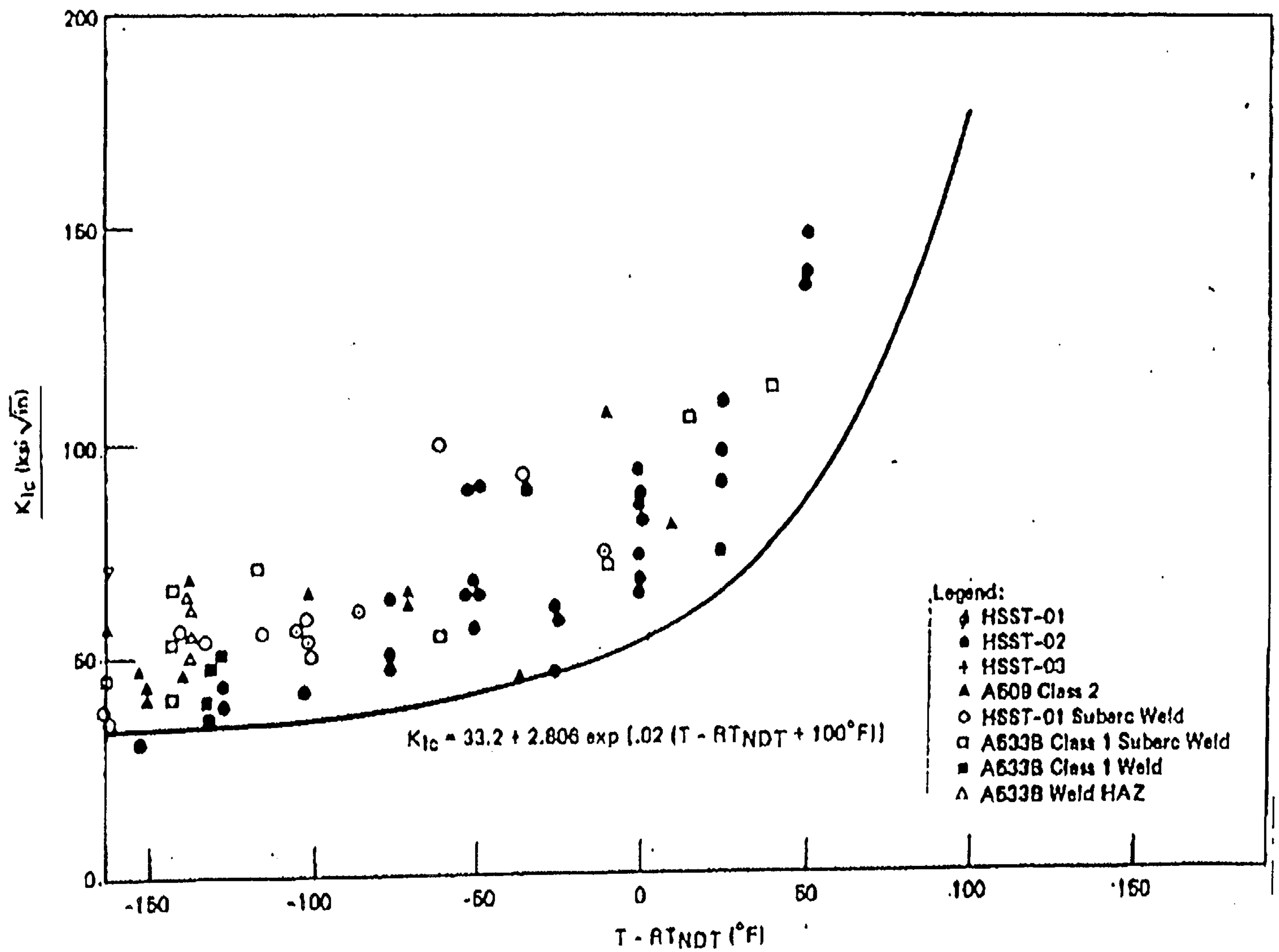

Figure 2. Original K_{Ic} Reference Toughness Curve, with Supporting Data [2]

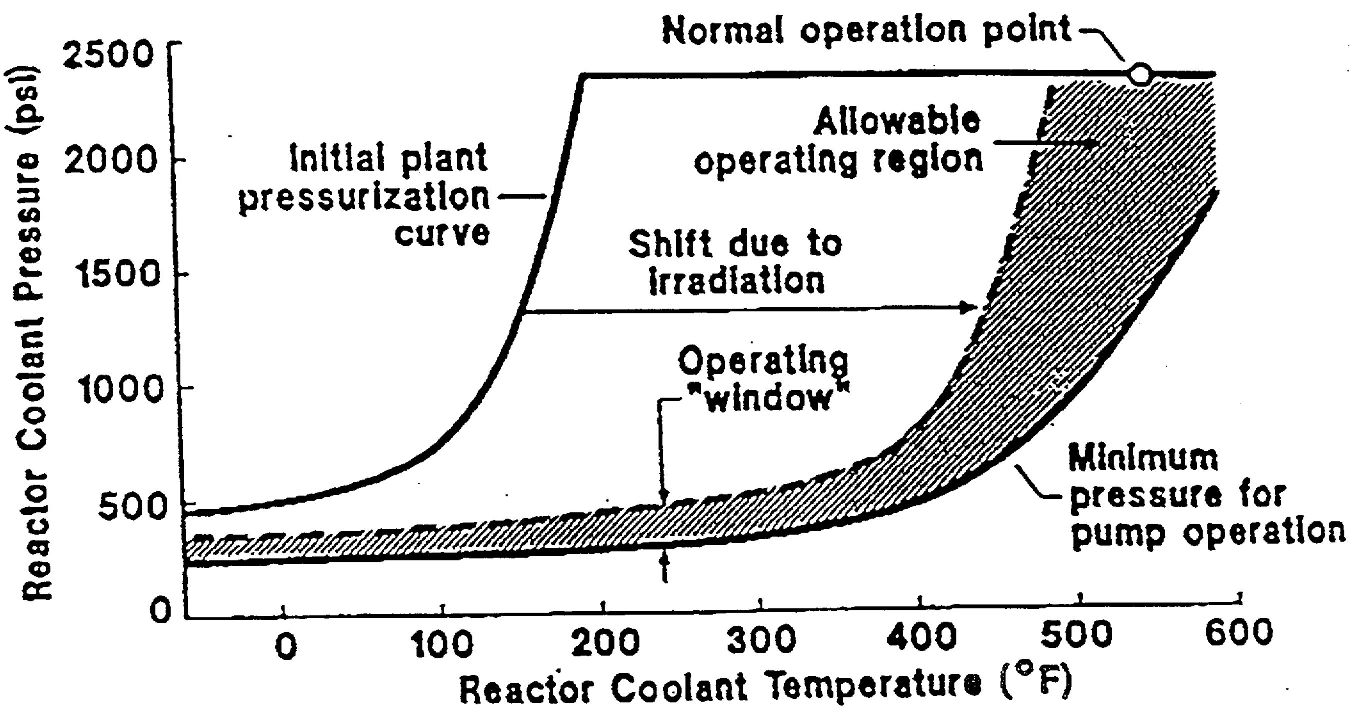

Figure 3. Operating Window From P-T Limit Curves [4]

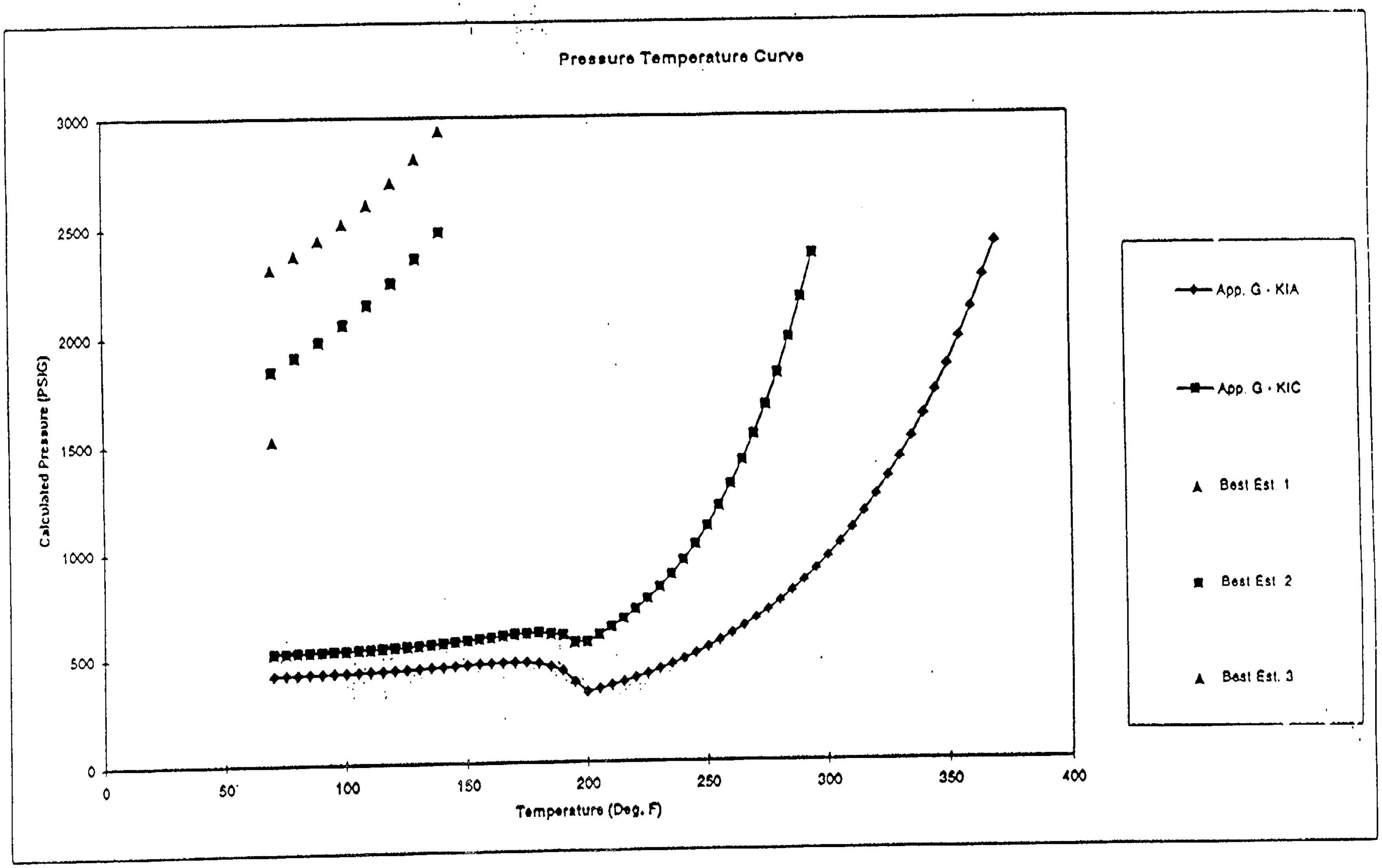

Figure 4. P-T Limit Curves Illustrating Deterministic Safety Factors

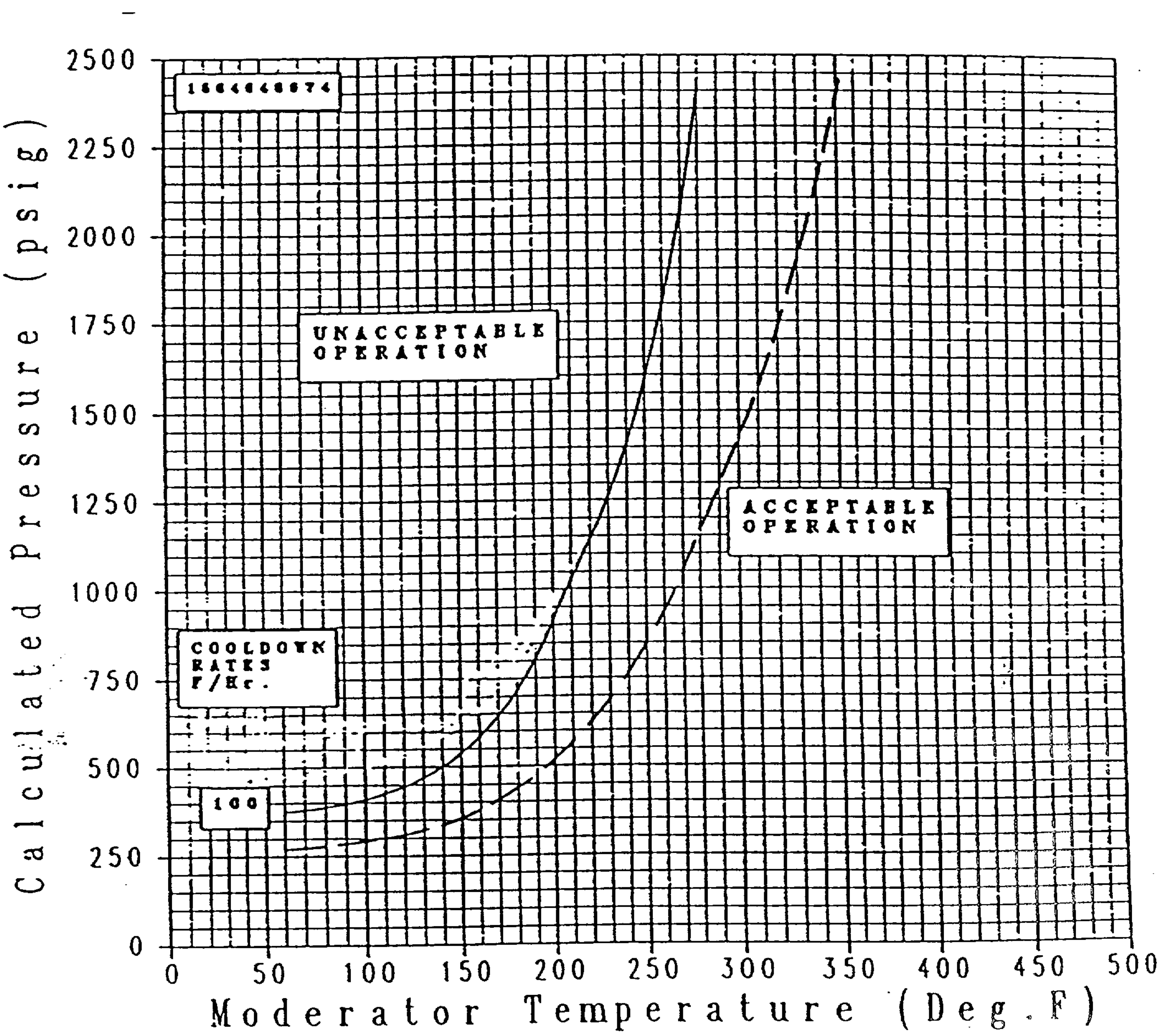

Figure 5. Comparison of Cool-Down Curves for the Existing and Proposed Methods
[Dashed Curve = Existing (K_{IA}) and Solid Curve = Proposed (K_{IC})]

Appendix A

Section XI P-T Limit Curve Sample Problems

Introduction

This series of sample problems was developed to allow comparison calculations to be carried out to support the proposed change from K-IA to K-IC in Appendix G of Section XI. These problems were developed in a meeting held on July 7, 1998, between the NRC staff, Westinghouse, ORNL, and Framatome Technologies. Later, a variation on the sample problems was developed for application to BWRs.

The sample problems involve a tightly specified reference case, with two variations, and then two P-T Limit curve calculations whose input is also tightly specified, one using K-IA and the second using K-IC. The goal of the problems is to determine the margin on pressure which exists using the K-IA approach, and the margin which exists with the proposed K-IC approach.

The problem input variables are contained in the attached tables. The problem statement is given below. As will be seen there are two problem types, the first being a best-estimate, or reference problem, and the second being standard P-T limit curves determined using code-type assumptions, with safety factors.

Reference Cases (Best Estimate)

Determine a best estimate P-T Cooldown Curve for a typical reactor vessel, over the entire temperature range of operation, starting at 70F. For BWR plants, also calculate a hydrotest pressure versus temperature curve. The problem input is defined in Table 1. This problem is meant to be a best estimate curve with no specific safety factors, and best estimate values for each of the variables. Only pressure and thermal stresses are used for case R1. Although these stresses are the only ones presently considered in P-T limit curve calculations, other stresses can exist in the vessel, and two other cases were constructed to obtain additional information on these issues. These other two cases treat stresses which are at issue regardless of which toughness is used for the calculations, but are provided for information.

Reference case R2. This case is similar to case R1, but the weld residual stresses are added for a longitudinal weld in the reactor vessel.

Reference case R3. This case is similar to case R2, but now the clad residual stresses are added. Since the clad residual stresses are negligible at higher temperatures, this calculation is only performed at room temperature, or 70F.

The stress intensity factor results for the reference cases may not always result in the maximum value at the deepest point of the flaw, so care should be taken to check this. If the maximum value is not at the deepest point, the calculated ratio of K / K-IC should be calculated around the periphery, and reported. The resulting allowable pressure would then be determined from the governing result at each time step. The calculation method could use either Section XI Appendix A, or the ORNL method, as documented in Table A-1.

P-T Curve Cases

Case 1 is a classic P-T Curve calculation done according to the existing rules in Section XI Appendix G, using the K-IA curve and the code specified safety factors. The input values are provided in Table A-2, for both PWR and BWR plants.

Case 2 is the same as case 1, except that the fracture toughness curve K-IC is used. This is the proposed Code change.

In each case a full P-T limit curve should be calculated, but there is no need to calculate leak test temperature, bolt-up temperature, or any other parameters. For BWR plants, a hydrotest pressure versus temperature curve is also required.

Guidelines for presentation of Results

The results of each of these curves should be presented in tabular form, as well as graphically. The scale on the graph should be as close as practical to the example provided.

TABLE A-1: REFERENCE CASE VARIABLES

Reference Case 1

Vessel Geometry: Thickness = 9.0 inch (PWR) or 6.0 inches (BWR)
Inside Radius = 90 inch (PWR) or 125 inches (BWR)
Clad Thickness = 0.25 inch

Flaw: Semi-elliptic Surface Flaw, Longitudinal Orientation
Depth = 1.0 inch
Length = 6 x Depth

Toughness: Mean K_{IC}, from report ORNL/NRC/LTR/93-15, July 12, 93
$K_{IC} = 36.36 + 51.59 \exp [0.0115 (T-RT_{NDT})]$

Loading: 100F/Hr cooldown from 550F to 200F — Film coefficient :
20F/Hr cooldown from 200F to 70F — h = 1000 B/hr-ft-F

Stress Intensity Factor Expression: Section XI, Appendix A, or ORNL Influence
............Coefficients, from ORNL/NRC/LTR-93-33 Rev. 1, Sept. 30, 95

Irradiation Effects: RT_{NDT} = 236°F @ inside surface
= 220°F @ depth = 1.0 in.
= 200°F @ depth = T/4
= 133°F @ depth = 3T/4

Requirement: Calculate allowable pressure as a function of coolant temperature and for BWR plants, calculate hydrotest pressure as a function of coolant temperature.

Reference Case 2

Same as Reference Case 2, but for the loadings, add a weld residual stress distribution.

	Location (a/t)	Stress(ksi)	Location (a/t)	Stress(ksi)
Inner Surface	0.000	6.50	0.045	5.47
	0.067	4.87	0.101	3.95
	0.134	2.88	0.168	1.64
	0.226	-0.79	0.285	-3.06
	0.343	-4.35	0.402	-4.31
	0.460	-3.51	0.510	-2.57
	0.572	-1.70	0.619	-1.05
	0.667	-0.46	0.739	0.35
	0.786	0.87	0.834	1.41
	0.881	1.96	0.929	2.55
	0.976	3.20	1.000	3.54

Reference Case 3

Same as Reference Case 2, but add clad residual stress distribution, and calculate allowable pressure only at 70°F.

For the clad residual stress distribution, choose either distribution 1 or distribution 2, from the attached figures. Figure A-1 was calculated from the ORNL Favor Code, and Figure A-2 was taken from a technical paper which presents results of residual stresses measured on nozzle drop-out materials.

TABLE A-2: P-T Calculation Cases

Calculation Case 1

Vessel Geometry: Thickness - 9.0 inch (PWR), 6.0 inches (BWR)
Inside Radius = 90 inch (PWR), 125 inches (BWR)
Clad Thickness = 0.25 inch

Flaw: Semi-elleptic Surface Flaw, Longitudinal Orientation
Depth - 1.0 inch
Length = 6 x Depth

Toughness: K_{IA}

Loading: 100F/hr cooldown, 550 to 200 F
20F/hr cooldown, 200 to 70F

Stress Intensity Factor Expression: Latest Section XI App G expression (from ....ORNL/NRC/LTR-93-33, Rev. 1)

Irradiation Effects: ART = 236F @ inside surface
= 220F @ depth = 1.0 inch
= 200F @ depth = T/4
= 133F @ depth = 3T/4

Requirement: Calculate allowable pressure as a function of temperature, and for BWRs calculate hydrotest pressure as a function of temperature.

Calculation Case 2

Same parameters as Case 1, but Toughness = K_{IC}

From ORNL Favor Coe, per Terry Dickson, 7/9/98

From ORNL Favor Code, per Terry Dickson, 7/9/98

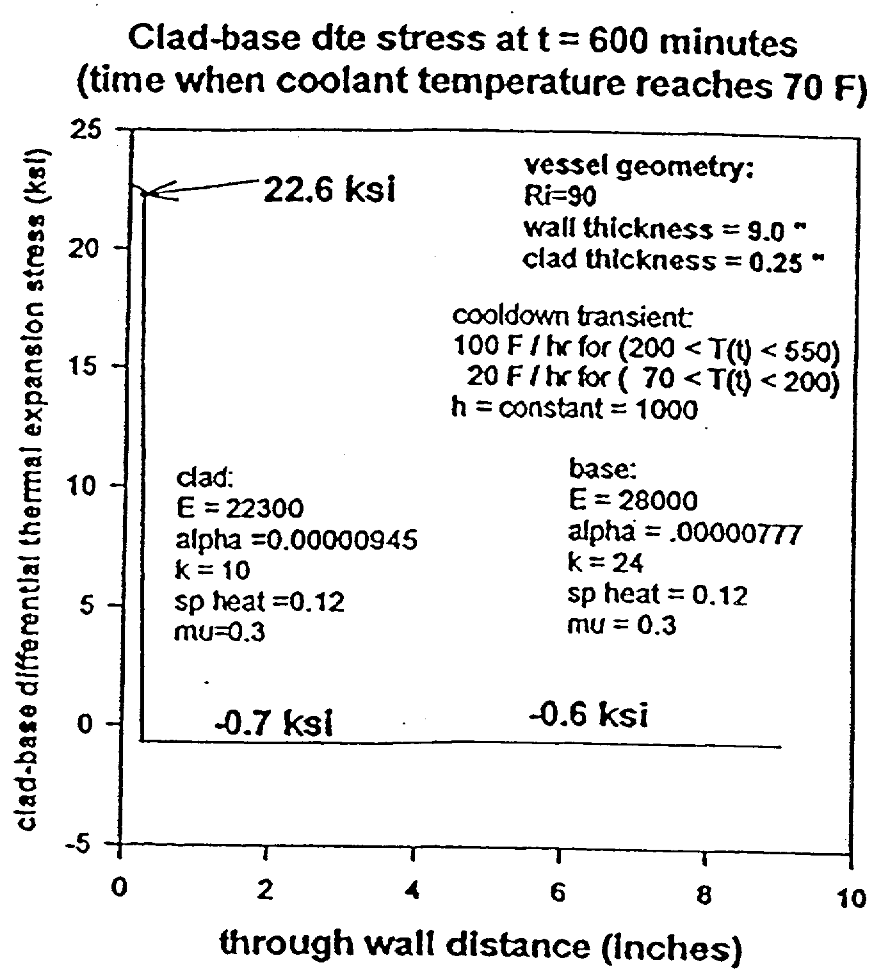

Figure A-1: Clad-base dte stress at t = 600 minutes
(time when coolant temperature reaches 70 F)

From "Effects of Cladding on Fracture Analysis," by W. H. Bamford and A. J. Bush, to be published at ASME PVP Conference, July 1998

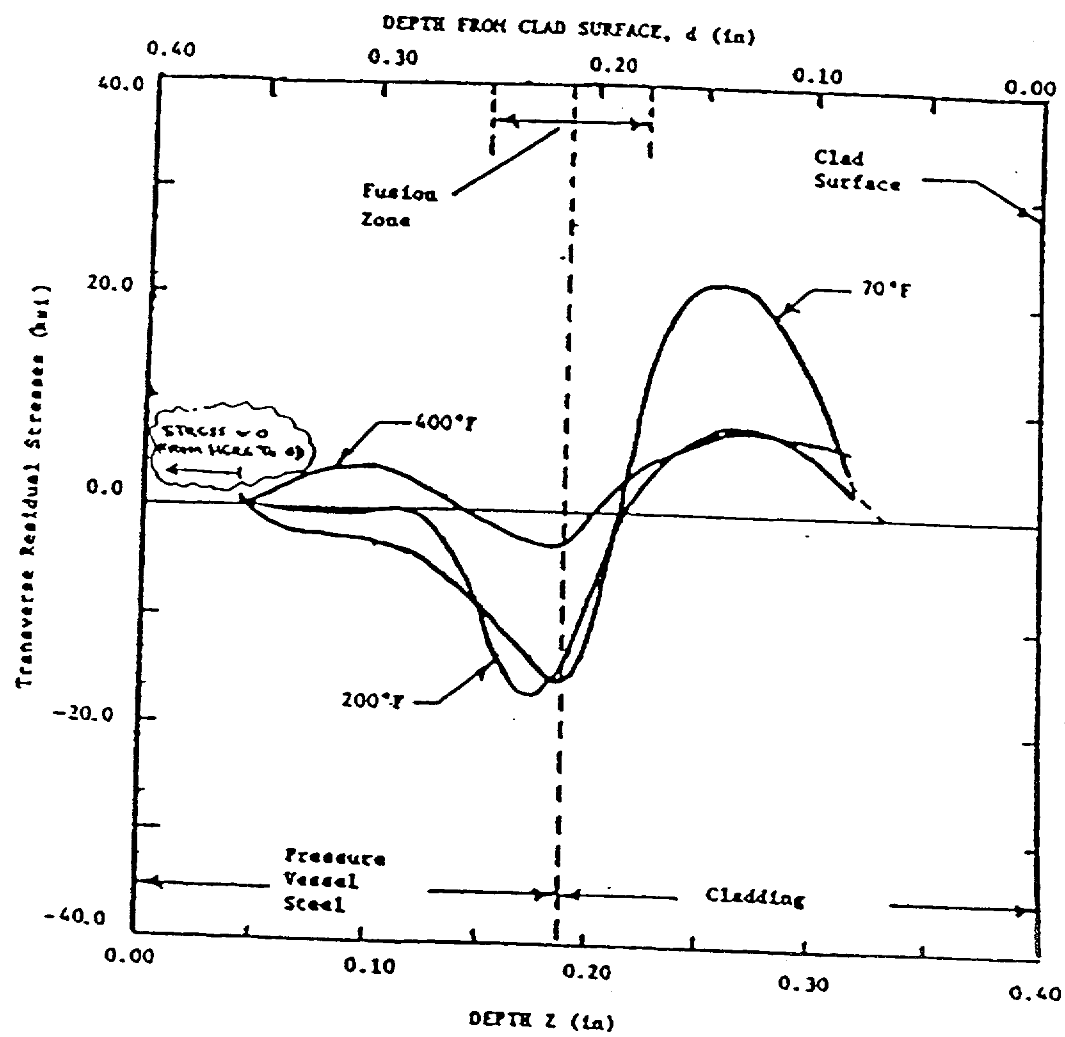

Figure A-2: Residual Stresses Transverse to Direction of Welding