DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION

ATTACHMENT 4

WCAP-15192

MCGUIRE UNIT 1

HEATUP AND COOLDOWN LIMIT CURVES

FOR NORMAL OPERATION

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15192

McGuire Unit 1
Heatup and Cooldown Limit Curves
For Normal Operation

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PREFACE

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EXECUTIVE SUMMARY

The purpose of this report is to generate pressure-temperature limit curves for McGuire Unit 1 for normal operation at 34 EFPY using the methodology from the 1996 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G along with Code Case N-640. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values at the ¼T and ¾T location. The limiting ¼T and ¾T ART values are summarized in Table 4-11 and were calculated using the lower shell longitudinal weld seams 3-442A and C (i.e. The limiting beltline region material when credible surveillance testing data is used). The pressure-temperature limit curves were generated for a heatup rates of 60, 80 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100°F/hr. These curves can be found in Figures 5-1 through 5-3.

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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 (Transverse to the major rolling 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/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} + ΔRT_{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 March 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_{1C} methodology rather than the K_{1A} methodology. In December of 1998 Westinghouse completed an analysis of surveillance capsule Y from the McGuire Unit 1 reactor vessel. As a part of this analysis Westinghouse generated new heatup and cooldown curves for normal operation. These heatup and cooldown curves were generated based on the 1989 version of Appendix G to the ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1 and non-credible surveillance program weld data. The curves developed in 1998 were developed without margins for instrumentation errors, 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 report is to present the calculations and development of the Duke Power Company McGuire Unit 1 pressure-temperature curves for 34 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 temperature flange requirement of Appendix G to 10 CFR Part 50 based on the use of the K_{1C} methodology rather than the K_{1A} methodology. These pressure-temperature curves are being developed for normal operation up to 34 EFPY and do not include margins for instrumentation errors. In addition, 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.

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. Appendix G to Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components"[3] and ASME Code Case N-640[4] contain 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_{1C}, for the metal temperature at that time. K_{1C} is obtained from the reference fracture toughness curve, defined in Code Case N-640 of the ASME Appendix G to Section XI. The K_{1c} curve is given by the following equation:

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

where,

K_{1C} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{1C} 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-508-1, 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 as:

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

where,

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

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

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:

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

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

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

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:

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

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[8] 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_{1C} 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 tèmperature 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 ΔT (temperature) developed during cooldown results in a higher value of K_{1C} at the ¼T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{1C} 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_{1C} for the ¼T crack during heatup is lower than the K_{1C} 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_{1C} 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 (3107 psig), which is 621 psig for the McGuire 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 such as the McGuire 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 40°F (Table 4-5 in WCAP-14994) occurs in the closure head flange of the McGuire Unit 1 reactor vessel, so the minimum allowable temperature of this region is 85°F at pressures greater than 621 psig with no margin for uncertainties.

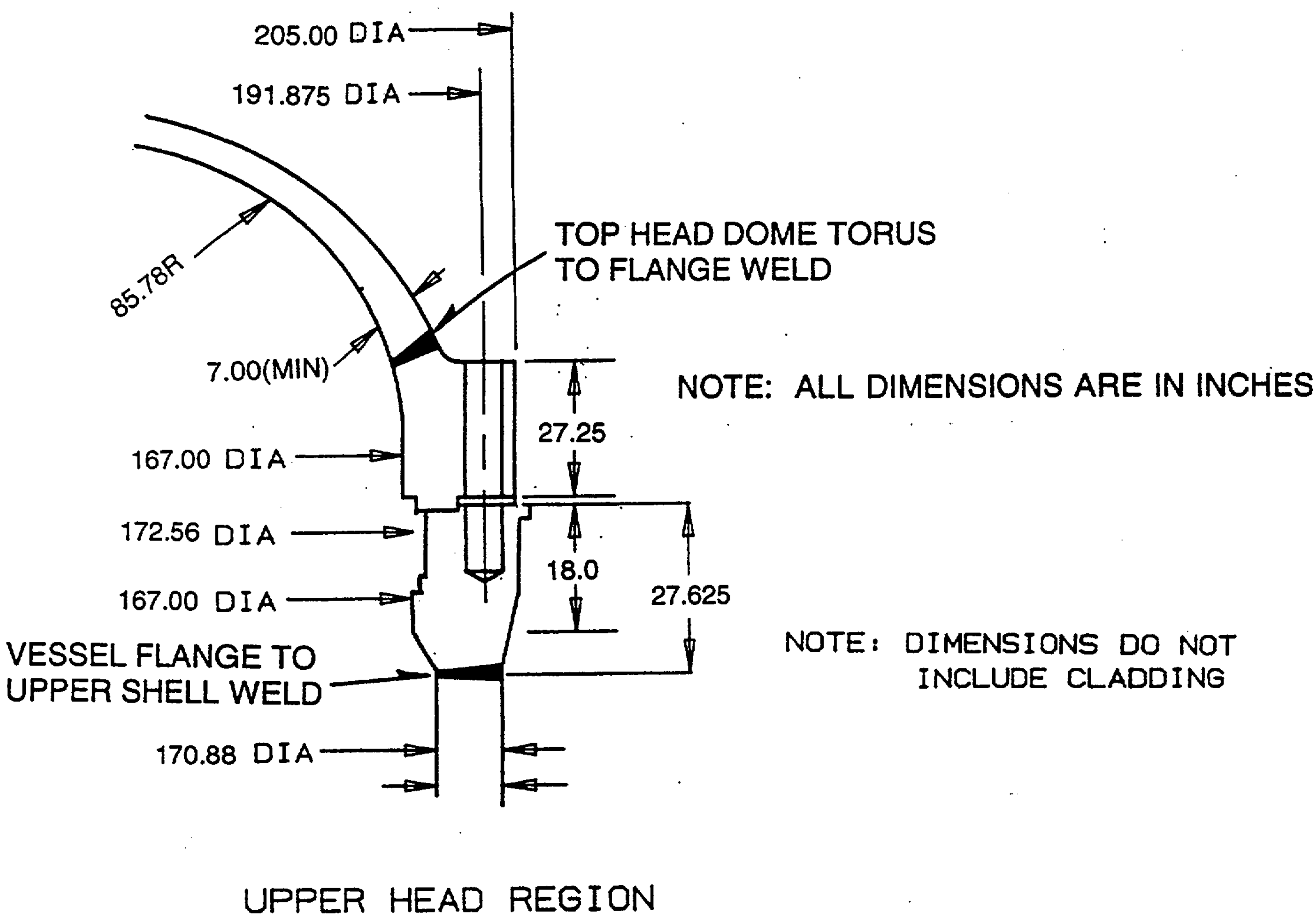

Figure 3-1 Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

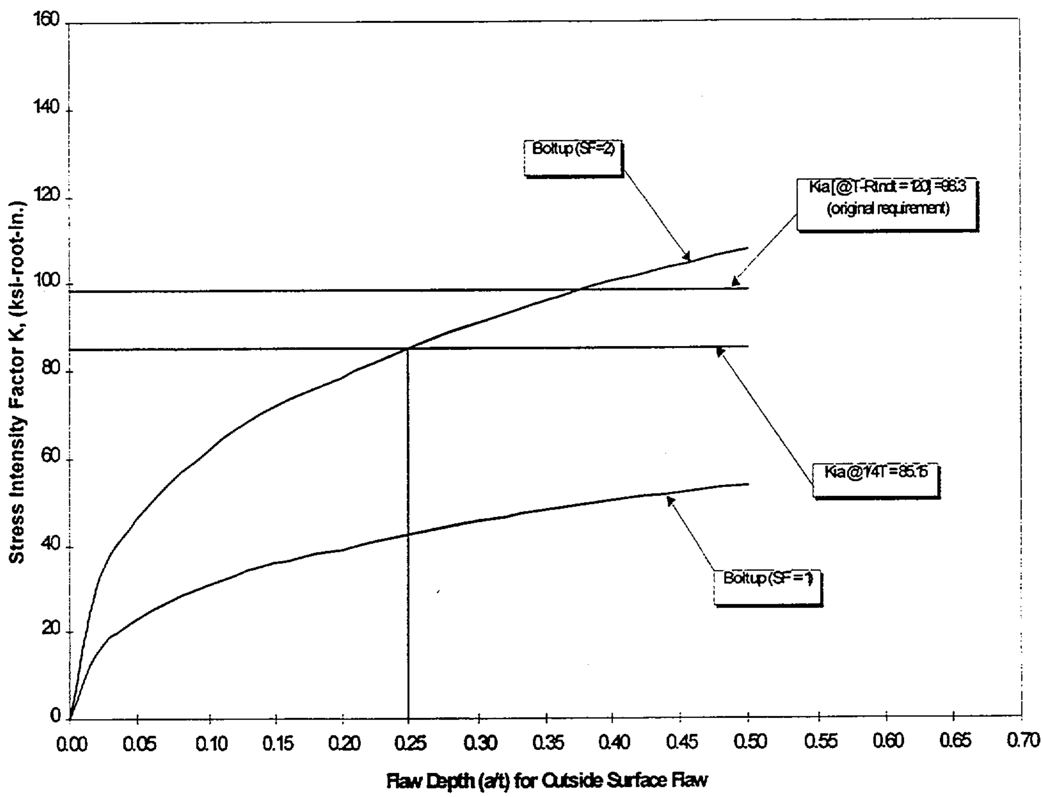

Figure 3-2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

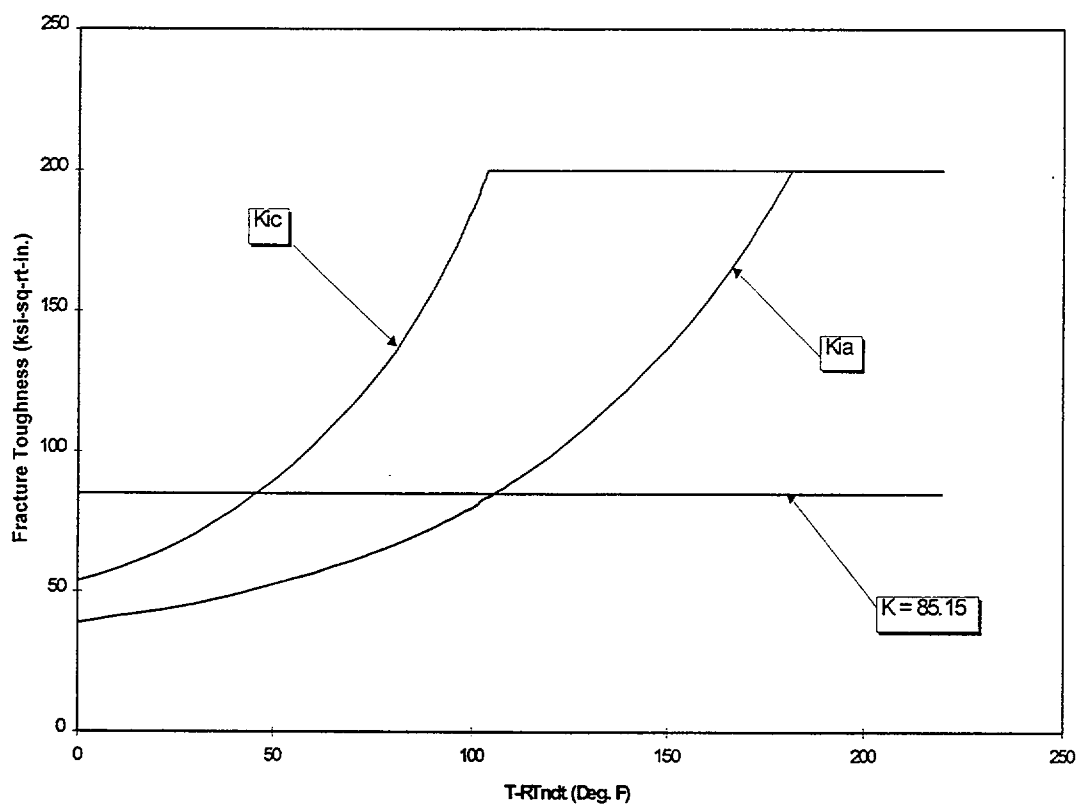

Figure 3-3 Determination of Boltup Requirement, using K_{IC}

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[5]. 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:

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

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 8.63 inches[6]) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 10 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections[7] and the results of the calculations of the peak fluence values at the vessel clad/base metal interface are presented in Table 4-1 while the capsule fluence values are presented in Table 4-2 of this report. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Table 4-1 contains the best estimate vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, ¼T and ¾T calculated fluences used to generate the ART values for all beltline materials in the McGuire Unit 1 reactor vessel.

TABLE 4-1
Summary of the Peak Pressure Vessel Neutron Fluence Values
at the Clad/Base Metal Interface at 34 EFPY
(n/cm^2, E > 1.0 MeV)

0°	15°	30°	45°
1.23E+19	1.80E+19	1.74E+19	1.96E+19

TABLE 4-2
Measured Integrated Neutron Exposure of the McGuire Unit 1
Surveillance Capsules Tested to Date

Capsule	Fluence
U	4.447×10^{18} n/cm^2 , (E > 1.0 MeV)
X	1.424×10^{19} n/cm^2 , (E > 1.0 MeV)
V	1.940×10^{19} n/cm^2 , (E > 1.0 MeV)
Z	2.166×10^{19} n/cm^2 , (E > 1.0 MeV)
Y	2.933×10^{19} n/cm^2 , (E > 1.0 MeV)

TABLE 4-3*
Summary of the Peak Pressure Vessel Neutron Fluence Values
at 34 EFPY used for the Calculation of ART Values
(n/cm^2, E > 1.0 MeV)

Material	Surface	¼ T	¾ T
Intermediate Shell Plate B5012-1	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Intermediate Shell Plate B5012-2	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Intermediate Shell Plate B5012-3	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Lower Shell Plate B5013-1	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Lower Shell Plate B5013-2	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Lower Shell Plate B5013-3	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	1.23×10^{19}	0.733×10^{19}	0.260×10^{19}
Intermediate Shell Longitudinal Weld Seam 2-442B (120° Azimuth)	1.74×10^{19}	1.04×10^{19}	0.368×10^{19}
Intermediate Shell Longitudinal Weld Seam 2-442C (240° Azimuth)	1.74×10^{19}	1.04×10^{19}	0.368×10^{19}
Intermediate to Lower Shell Cirumferential Weld Seam 9-442	1.96×10^{19}	1.17×10^{19}	0.415×10^{19}
Lower Shell Longitudinal Weld Seams 3-442A & C (60° & 300° Azimuth)	1.74×10^{19}	1.04×10^{19}	0.368×10^{19}
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	1.23×10^{19}	0.733×10^{19}	0.260×10^{19}

* This data was taken from Table 4-2 in WCAP-14994.

Margin is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, σ_i, is 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is used. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates when surveillance capsule data is not used and 8.5°F for plates when surveillance capsule data is used. For welds, σ_Δ is 28°F when surveillance capsule data is not used and 14°F when surveillance capsule data is used. In addition, σ_Δ need not exceed one-half the mean value of ΔRT_{NDT}.

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials[7]. These measured shift values were obtained using CVGRAPH, Version 4.1[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 B5012-1 (Longitudinal Orientation)	U	30.95°F
	X	33.51°F
	V	81.01°F
	Y	93.10°F
Intermediate Shell Plate B5012-1 (Transverse Orientation)	U	48.44°F
	X	60.69°F
	V	74.60°F
	Y	108.58°F
Surveillance Program Weld Metal	U	161.32°F
	X	170.69°F
	V	180.15°F
	Y	190.42°F
Heat Affected Zone	U	86.04°F
	X	115.11°F
	V	139.74°F
	Y	154.38°F

Notes:
(a) Calculated using measured Charpy data and plotted using CVGRAPH[9]
(b) This data was taken from Table 5-10 of WCAP-14993.

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

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[(a)]
Closure Head Flange B5002	---	0.75	40°F
Vessel Flange B4701	---	0.73	29°F
Intermediate Shell Plate B5012-1	0.11	0.61	34°F
Intermediate Shell Plate B5012-2	0.14	0.61	0°F
Intermediate Shell Plate B5012-3	0.11	0.66	-13°F
Lower Shell Plate B5013-1	0.14	0.58	0°F
Lower Shell Plate B5013-2	0.10	0.51	30°F
Lower Shell Plate B5013-3	0.10	0.55	15°F
Intermediate Shell Longitudinal Welds, 2-442A, B & C[(b)]	0.199	0.846	-50°F
Lower Shell Longitudinal Welds, 3-442A, B & C[(b)]	0.213	0.867	-50°F
Circumferential Weld 9-442[(b)]	0.051	0.096	-70°F
McGuire 1: Surveillance Program Weld Metal	0.198	0.874[(c)]	---
Diablo Canyon 2: Surveillance Program Weld Metal	0.219	0.867	---

Notes:

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

(b) The intermediate shell longitudinal weld seams 2-442A, B and C were fabricated with weld wire heat numbers 20291 and 12008, Flux Type 1092 Lot Number 3854. The intermediate to lower shell circumferential weld seam 9-442 was fabricated with weld wire heat number 83640, Flux Type 0091 Lot Number 3490. The lower shell longitudinal weld seams 3-442A, B and C were fabricated with weld wire heat number 21935 and 12008, Flux Type 1092 Lot Number 3889. The McGuire Unit 1 surveillance weld metal was made with the same weld heat as the intermediate shell longitudinal weld seams, while the Diablo Canyon Unit 2 surveillance weld metal was made with the same weld wire heat as the lower shell longitudinal weld seams (Justification in WCAP-13949, ref. 6). Per Regulatory Guide 1.99, Revision 2, "weight percent copper" and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld."

(c) Value is the average of five measurements (see Tables 4-1 and 4-2 in WCAP-14993).

TABLE 4-6

Calculation of Chemistry Factors using McGuire Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f[1]	FF[2]	ΔRT_{NDT}[3]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell Plate	U	0.4447	0.775	30.95	23.99	0.60
B5012-1	X	1.424	1.098	33.51	36.79	1.21
	V	1.940	1.181	81.01	95.67	1.39
(Longitudinal)	Y	2.933	1.285	93.10	119.63	1.65
Intermediate Shell Plate	U	0.4447	0.775	48.44	37.54	0.60
B5012-1	X	1.424	1.098	60.69	66.64	1.21
	V	1.940	1.181	74.60	88.10	1.39
(Transverse)	Y	2.933	1.285	108.58	139.53	1.65
				SUM	607.89	9.70
	$CF_{B5012\text{-}1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (607.89) \div (9.70) = 62.7°F$					
Intermediate Shell	U	0.4447	0.775	159.71[4]	123.77	0.60
Longitudinal Welds	X	1.424	1.098	168.98[4]	185.54	1.21
2-442A, B & C	V	1.940	1.181	178.35[4]	210.63	1.39
	Y	2.933	1.285	188.52[4]	242.25	1.65
				SUM	762.19	4.85
	$CF_{Weld\ 2\text{-}442} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (762.19) \div (4.85) = 157.15°F$					
Intermediate Shell	U	0.357	0.716	172.10[4&5]	123.22	0.51
Longitudinal Welds	X	0.866	0.960	202.20[4&5]	194.11	0.92
3-442A, B & C	Y	1.32	1.077	210.40[4&5]	226.60	1.16
(Using Diablo Canyon 2				SUM	543.93	2.59
Surveillance Data)	$CF_{Weld\ 3\text{-}442} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (543.93) \div (2.59) = 210.01°F$					

Notes:

(1) f = Measured fluence from capsule Y dosimetry analysis results[7], (x 10^{19} n/cm², E > 1.0 MeV).

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

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

(4) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 0.99.

(5) Fluences and pre-adjusted ΔRT_{NDT} values were updated in reference 12.

TABLE 4-7
Summary of the McGuire 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[(a)]	Position 2.1[(a)]
Intermediate Shell Plate B5012-1	74.2°F	62.7°F
Intermediate Shell Plate B5012-2	100.3°F	---
Intermediate Shell Plate B5012-3	74.9°F	---
Lower Shell Plate B5013-1	99.1°F	---
Lower Shell Plate B5013-2	65.0°F	---
Lower Shell Plate B5013-3	65.0°F	---
Intermediate Shell Longitudinal Welds 2-442A, B & C[(c)]	201.3°F	157.2°F
Lower Shell Longitudinal Welds 3-442A, B & C[(d)]	208.2°F	210.0°F[(b)]
Circumferential Weld 9-442	37.5°F	---
McGuire 1 Surveillance Program Weld Metal[(c)]	204.2°F	---
Diablo Canyon 2 Surveillance Program Weld Metal[(d)]	211.2°F	---

Notes:

(a) Regulatory Guide 1.99, Revision 2, Position 1.1 (i.e. using best estimate chemistry and tables) or Position 2.1 (i.e. using surveillance program results) methodology.
(b) This was determined using surveillance capsule data from Diablo Canyon Unit 2 (Table 4-6). Justification can be found in Reference 6.
(c) The McGuire Unit 1 surveillance capsule weld material fabricated with the same weld wire heat as the intermediate shell longitudinal weld seams 2-442A, B, C (Heat # 20291/12008).
(d) The Diablo Canyon Unit 2 surveillance capsule weld material was fabricated with the same weld wire heat as the lower shell longitudinal weld seams 3-442A, B, C of the McGuire Unit 1 (Heat # 21935/12008).

Contained in Table 4-8 is a summary of the fluence factor (FF) values used in the calculation of adjusted reference temperatures for the McGuire Unit 1 reactor vessel beltline materials for 34 EFPY.

TABLE 4-8
Summary of the Calculated Fluence Factors used for the Generation of the 34 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm2, E > 1.0 MeV)	¼ T FF[(a)]	¾ T f (n/cm2, E > 1.0 MeV)	¾ T FF[(b)]
Intermediate Shell Plate B5012-1	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Intermediate Shell Plate B5012-2	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Intermediate Shell Plate B5012-3	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Lower Shell Plate B5013-1	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Lower Shell Plate B5013-2	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Lower Shell Plate B5013-3	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	0.733×10^{19}	0.91	0.260×10^{19}	0.63
Intermediate Shell Longitudinal Weld Seams 2-442B & C (120° & 240° Azimuth)	1.04×10^{19}	1.01	0.368×10^{19}	0.72
Intermediate to Lower Shell Cirumferential Weld Seam 9-442	1.17×10^{19}	1.04	0.415×10^{19}	0.76
Lower Shell Longitudinal Weld Seams 3-442A & C (60° & 300° Azimuth)	1.04×10^{19}	1.01	0.368×10^{19}	0.72
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	0.733×10^{19}	0.91	0.260×10^{19}	0.63

Notes:

(a) Fluence Factor at the 1/4T vessel thickness location.
(b) Fluence Factor at the 3/4T vessel thickness location.
(c) The data in this table was obtained from Table 4-10 of WCAP-14994.

Contained in Tables 4-9 and 4-10 are the calculations of the ART values used for the generation of the 34 EFPY pressure-temperature curves.

TABLE 4-9
Calculation of the ART Values for the 1/4T Location @ 34 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}(a)	ΔRT_{NDT}(c)	Margin	ART(b)
Intermediate Shell Plate B5012-1	Position 1.1	74.2	1.04	34	77.2	34	145
	Position 2.1	62.7	1.04	34	65.2	17(e)	116
Intermediate Shell Plate B5012-2	Position 1.1	100.3	1.04	0	104.3	34	138
Intermediate Shell Plate B5012 -3	Position 1.1	74.9	1.04	-13	77.9	34	99
Lower Shell Plate B5013-1	Position 1.1	99.1	1.04	0	103.1	34	137
Lower Shell Plate B5013-2	Position 1.1	65.0	1.04	30	67.6	34	132
Lower Shell Plate B5013 -3	Position 1.1	65.0	1.04	15	67.6	34	117
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	Position 1.1	201.3	0.91	-50	183.2	56	189
	Position 2.1	157.2	0.91	-50	143.1	28(e)	121
Intermediate Shell Longitudinal Weld Seam 2-442B & C (120° & 240° Azimuth)	Position 1.1	201.3	1.01	-50	203.3	56	209
	Position 2.1	157.2	1.01	-50	158.8	28(e)	137
Intermediate to Lower Shell Circumferential Weld Seam 9-442	Position 1.1	37.5	1.04	-70	39.0	39.0	8
Lower Shell Longitudinal Weld Seams 3-442A & C (60°& 300° Azimuth)	Position 1.1	208.2	1.01	-50	210.3	56	216
	Position 2.1	210.0(d)	1.01	-50	212.1	28	190
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	Position 1.1	208.2	0.91	-50	189.5	56	196
	Position 2.1	210.0(d)	0.91	-50	191.1	28	169

Notes:

(a) Initial RT_{NDT} values are measured values (see Table 4-5).
(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(c) ΔRT_{NDT} = CF * FF
(d) Based on Diablo Canyon Unit 2 surveillance capsule data (See Reference 7).
(e) The McGuire Surveillance Data is credible (See Reference 13).
(f) The data contained in this table was obtained from Table 4-14 in WCAP-14994 and Reference 13.

TABLE 4-10
Calculation of the ART Values for the 3/4T Location @ 34 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT} (a)	ΔRT_{NDT} (c)	Margin	ART (b)
Intermediate Shell Plate B5012-1	Position 1.1	74.2	0.76	34	56.4	34	124
	Position 2.1	62.7	0.76	34	47.7	17 (e)	99
Intermediate Shell Plate B5012-2	Position 1.1	100.3	0.76	0	76.2	34	110
Intermediate Shell Plate B5012 -3	Position 1.1	74.9	0.76	-13	56.9	34	78
Lower Shell Plate B5013-1	Position 1.1	99.1	0.76	0	75.3	34	109
Lower Shell Plate B5013-2	Position 1.1	65.0	0.76	30	49.4	34	113
Lower Shell Plate B5013 -3	Position 1.1	65.0	0.76	15	49.4	34	98
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	Position 1.1	201.3	0.63	-50	126.8	56	133
	Position 2.1	157.2	0.63	-50	99.0	28 (e)	77
Intermediate Shell Longitudinal Weld Seam 2-442B & C (120° & 240° Azimuth)	Position 1.1	201.3	0.72	-50	144.9	56	151
	Position 2.1	157.2	0.72	-50	113.2	28 (e)	91
Intermediate to Lower Shell Circumferential Weld Seam 9-442	Position 1.1	37.5	0.76	-70	28.5	28.5	-13
Lower Shell Longitudinal Weld Seams 3-442A & C (60°& 300° Azimuth)	Position 1.1	208.2	0.72	-50	149.9	56	156
	Position 2.1	210.0 (d)	0.72	-50	151.2	28	129
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	Position 1.1	208.2	0.63	-50	131.2	56	137
	Position 2.1	210.0 (d)	0.63	-50	132.3	28	110

Notes:

(a) Initial RT_{NDT} values are measured values (see Table 4-5).
(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(c) ΔRT_{NDT} = CF * FF
(d) Based on Diablo Canyon Unit 2 surveillance capsule data (See Reference 7).
(e) The McGuire Surveillance Data is credible (See Reference 13).
(f) The data contained in this table was obtained from Table 4-14 in WCAP-14994 and Reference 13.

The Lower Shell Longitudinal Welds (Seams 3-442A and C) are the limiting beltline material for all heatup and cooldown curves to be generated. Contained in Table 4-11 is a summary of the limiting ARTs to be used in the generation of the McGuire Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-11
Summary of the Limiting ART Values Used in the
Generation of the McGuire Unit 1 Heatup/Cooldown Curves

EFPY	1/4T Limiting ART	3/4T Limiting ART
34	190°F	129°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 Sections 3 and 4 of this report.

Figures 5-1 and 5-2 present the heatup curves with no margins for possible instrumentation errors for heatup rates of 60, 80 and 100°F/hr. This curve is applicable for up to 34 EFPY of operation of the McGuire Unit 1 reactor vessel. Additionally, Figure 5-3 presents the cooldown curves with no margins for possible instrumentation errors for cooldown rates of 0, 20, 40, 60, and 100°F/hr. These curves are also applicable for up to 34 EFPY of operation of the McGuire 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 5-3. 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 5-2. 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 ASME Code Case N-640 (approved Feb. 26, 1999) as follows:

$$1.5K_{Im} < K_{IC} \tag{12}$$

where,

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

$K_{IC} = 33.2 + 20.734\ e^{[0.02\,(T - RTNDT)]}$,

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 and provides 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 5-3 define all of the above limits for ensuring prevention of nonductile failure for the McGuire Unit 1 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-3 are presented in Tables 5-1 through 5-3, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD SEAMS 3-442A & C
LIMITING ART VALUES AT 34 EFPY: 1/4T, 190°F
3/4T, 129°F

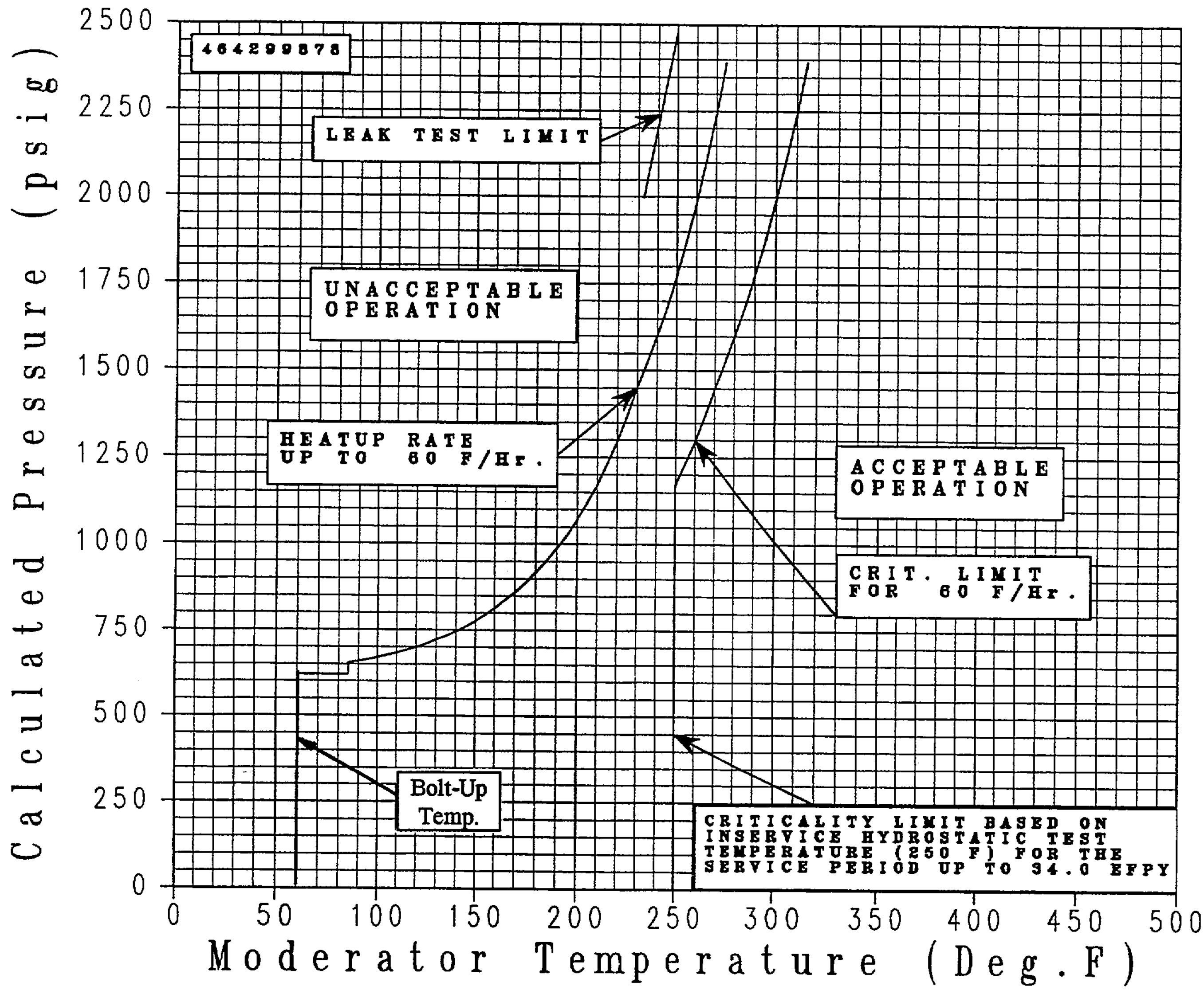

FIGURE 5-1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable to 34 EFPY (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD SEAMS 3-442A & C
LIMITING ART VALUES AT 34 EFPY: 1/4T, 190°F
3/4T, 129°F

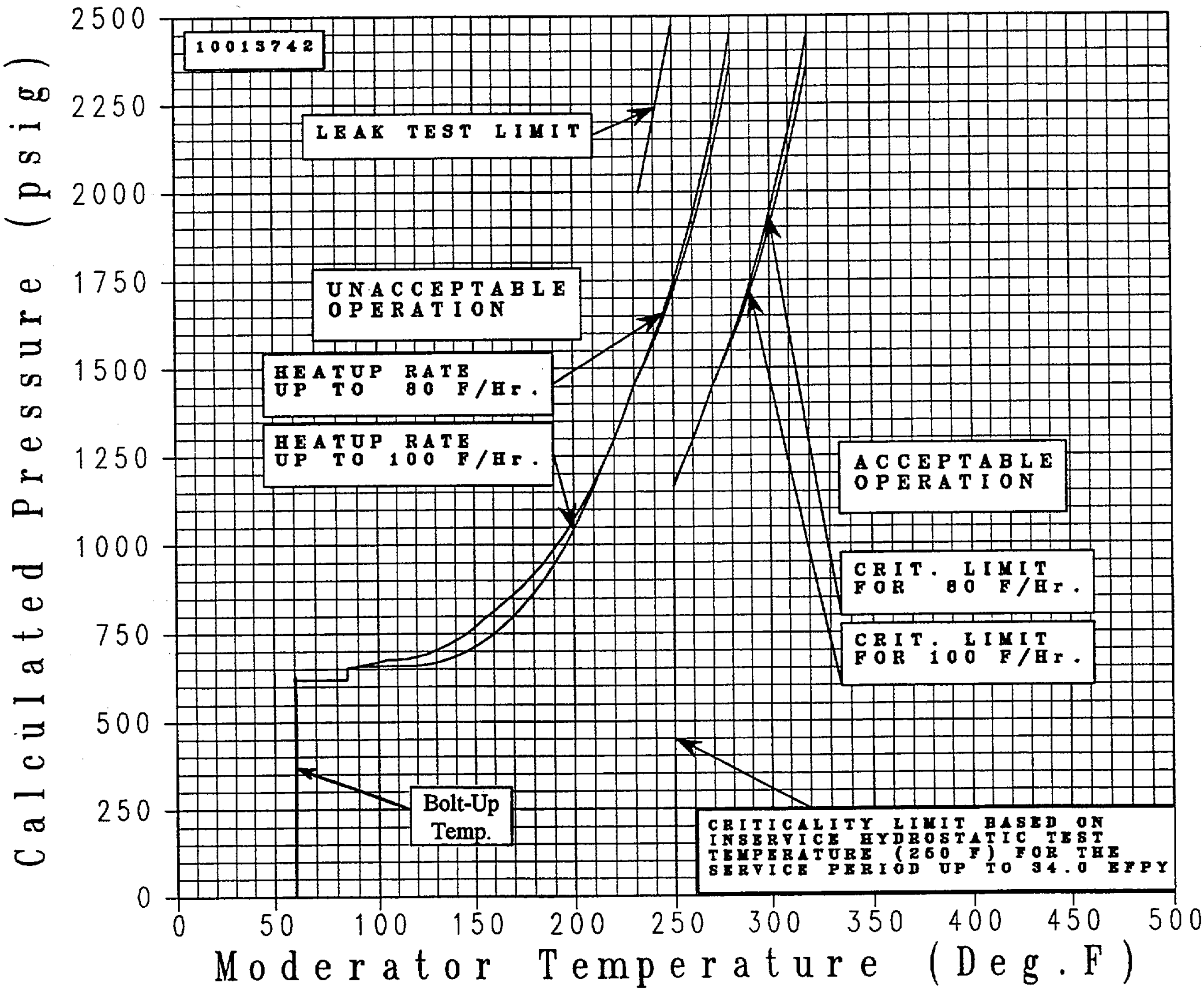

FIGURE 5-2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 and 100°F/hr) Applicable to 34 EFPY (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD SEAMS 3-442A & C
LIMITING ART VALUES AT 34 EFPY: 1/4T, 190°F
3/4T, 129°F

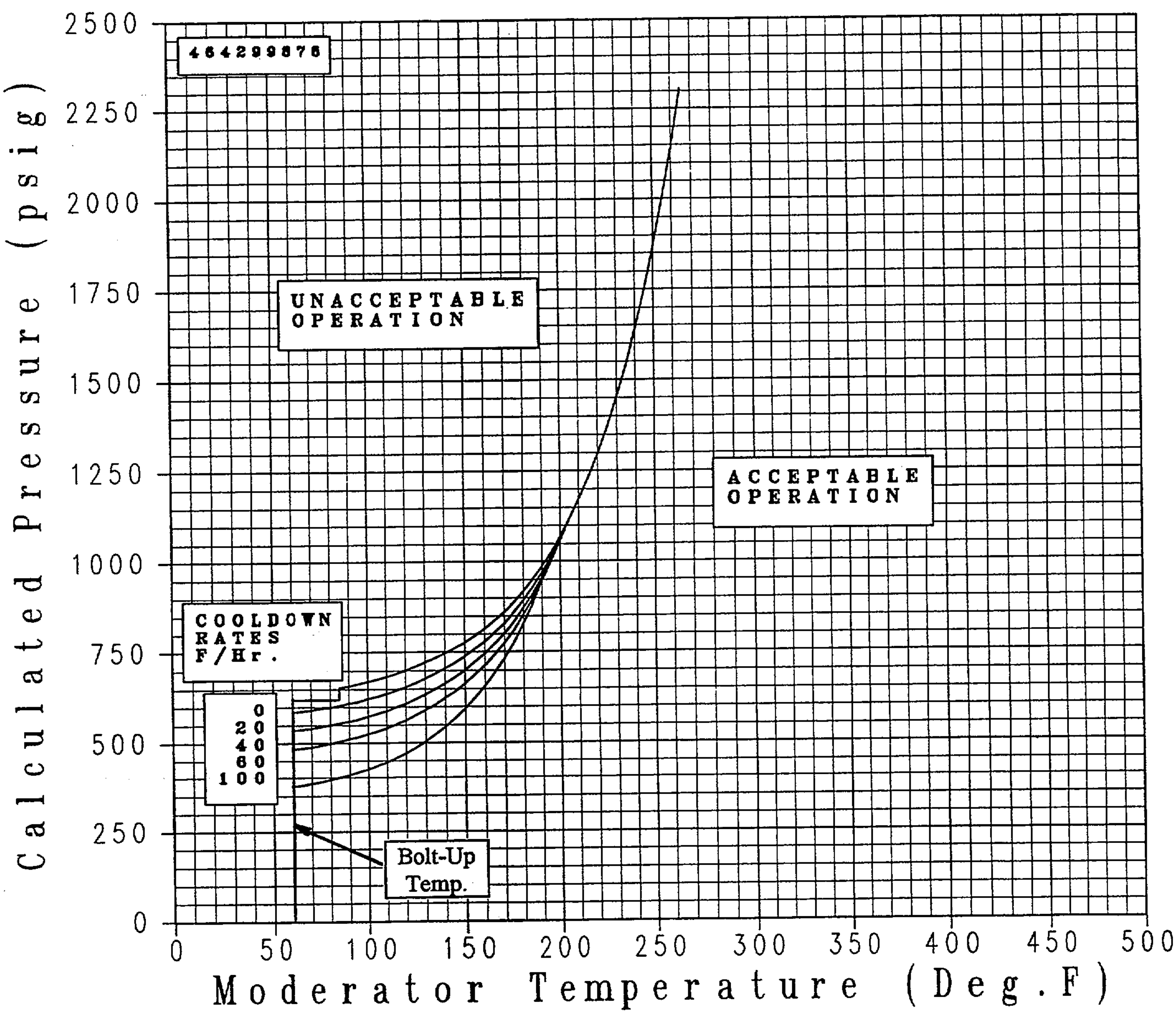

FIGURE 5-3 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable to 34 EFPY (Without Margins for Instrumentation Errors)

TABLE 5-1
McGuire Unit 1 Heatup Data at 34 EFPY (60°F/hr)
Without Margins for Instrumentation Errors

60 °F/hr		Critical. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	250	0	233	2000
60	621	250	638	250	2485
65	621	250	654		
70	621	250	658		
75	621	250	664		
80	621	250	670		
85	621	250	676		
85	654	250	684		
90	658	250	692		
95	664	250	701		
100	670	250	711		
105	676	250	721		
110	684	250	733		
115	692	250	747		
120	701	250	761		
125	711	250	778		
130	721	250	795		
135	733	250	815		
140	747	250	837		
145	761	250	861		
150	778	250	888		
155	795	250	918		
160	815	250	950		
165	837	250	986		
170	861	250	1026		
175	888	250	1070		
180	918	250	1119		
185	950	250	1173		
190	986	255	1232		
195	1026	260	1298		
200	1070	265	1371		
205	1119	270	1451		
210	1173	275	1525		
215	1232	280	1600		
220	1298	285	1682		
225	1371	290	1774		

TABLE 5-1 (Continued)
McGuire Unit 1 Heatup Data at 34 EFPY (60°F/hr)
Without Margins for Instrumentation Errors

60 °F/hr		Critical. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
230	1451	295	1875		
235	1525	300	1986		
240	1600	305	2108		
245	1682	310	2244		
250	1774	315	2393		
255	1875				
260	1986				
265	2108				
270	2244				
275	2393				

TABLE 5-2
McGuire Unit 1 Heatup Data at 34 EFPY (80 and 100°F/hr) Without Margins for Instrumentation Errors

80 °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	250	0	60	0	250	0	233	2000
60	621	250	638	60	621	250	638	250	2485
65	621	250	654	65	621	250	654		
70	621	250	658	70	621	250	658		
75	621	250	664	75	621	250	664		
80	621	250	670	80	621	250	670		
85	621	250	676	85	621	250	664		
85	654	250	677	85	654	250	660		
90	658	250	679	90	658	250	659		
95	664	250	685	95	659	250	660		
100	670	250	692	100	659	250	663		
105	676	250	702	105	659	250	668		
110	677	250	715	110	659	250	676		
115	679	250	730	115	659	250	686		
120	685	250	748	120	660	250	698		
125	692	250	769	125	663	250	713		
130	702	250	793	130	668	250	731		
135	715	250	815	135	676	250	751		
140	730	250	837	140	686	250	774		
145	748	250	861	145	698	250	801		
150	769	250	888	150	713	250	830		
155	793	250	918	155	731	250	864		
160	815	250	950	160	751	250	902		
165	837	250	986	165	774	250	944		
170	861	250	1026	170	801	250	990		
175	888	250	1070	175	830	250	1042		
180	918	250	1119	180	864	250	1100		
185	950	250	1173	185	902	250	1164		
190	986	255	1232	190	944	255	1232		
195	1026	260	1298	195	990	260	1298		
200	1070	265	1371	200	1042	265	1371		
205	1119	270	1451	205	1100	270	1451		
210	1173	275	1515	210	1164	275	1512		
215	1232	280	1584	215	1232	280	1574		
220	1298	285	1659	220	1298	285	1643		
225	1371	290	1742	225	1371	290	1719		

TABLE 5-2 (Continued)
McGuire Unit 1 Heatup Data at 34 EFPY (80 and 100°F/hr)
Without Margins for Instrumentation Errors

80 °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)
230	1451	295	1833	230	1451	295	1802		
235	1515	300	1934	235	1512	300	1894		
240	1584	305	2045	240	1574	305	1995		
245	1659	310	2168	245	1643	310	2106		
250	1742	315	2303	250	1719	315	2229		
255	1833	320	2452	255	1802	320	2364		
260	1934			260	1894				
265	2045			265	1995				
270	2168			270	2106				
275	2303			275	2229				
280	2452			280	2364				

TABLE 5-3
McGuire Unit 1 Cooldown Data at 34 EFPY
Without Margins for Instrumentation Errors

Steady State		20 °F/hr		40 °F/hr		60 °F/hr		100 °F/hr	
Temp. (°F)	Press. (psig)	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	0
60	621	60	586	60	535	60	484	60	377
65	621	65	589	65	539	65	487	65	381
70	621	70	592	70	542	70	491	70	385
75	621	75	596	75	546	75	495	75	390
80	621	80	600	80	550	80	500	80	395
85	621	85	605	85	555	85	505	85	401
85	654	90	610	90	561	90	511	90	408
90	658	95	616	95	567	95	517	95	416
95	664	100	622	100	574	100	525	100	425
100	670	105	629	105	581	105	533	105	435
105	676	110	637	110	590	110	542	110	446
110	684	115	646	115	599	115	552	115	458
115	692	120	655	120	610	120	564	120	472
120	701	125	666	125	621	125	576	125	487
125	711	130	678	130	634	130	590	130	504
130	721	135	691	135	648	135	606	135	524
135	733	140	705	140	664	140	624	140	545
140	747	145	721	145	682	145	643	145	569
145	761	150	739	150	701	150	664	150	595
150	778	155	759	155	723	155	688	155	625
155	795	160	780	160	747	160	715	160	657
160	815	165	804	165	773	165	744	165	694
165	837	170	831	170	803	170	777	170	734
170	861	175	860	175	835	175	813	175	779
175	888	180	893	180	871	180	853	180	829
180	918	185	929	185	911	185	897	185	884
185	950	190	969	190	955	190	946	190	946
190	986	195	1013	195	1004	195	1001	195	1014
195	1026	200	1062	200	1058	200	1061		
200	1070	205	1116	205	1118				
205	1119								
210	1173								
215	1232								
220	1298								
225	1371								
230	1451								

TABLE 5-3 (Continued)
McGuire Unit 1 Cooldown Data at 34 EFPY
Without Margins for Instrumentation Errors

Steady State		20 °F/hr		40 °F/hr		60 °F/hr		100 °F/hr	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
235	1540								
240	1638								
245	1746								
250	1866								
255	1999								
260	2145								
265	2307								

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.

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