Duke Power
A Duke Energy Company

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(704) 875-4100

May 30, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370
Supplement to Proposed Revision to Technical Specifications (TS) Submittal dated February 14, 2001
TAC Numbers MB1261 and MB1262

By letter dated February 14, 2001, Duke Energy submitted a proposed TS change to revise TS 3.3.2, Engineered Safety Feature Actuation System Instrumentation, TS 3.4.3, Reactor Coolant System Pressure and Temperature (P/T) Limits, and TS 3.4.12, Low Temperature Overpressure Protection System. The TS change includes McGuire P/T limit curves extended to 34 Effective Full Power Years. Westinghouse identified an error in the computer code used to generate the proposed P/T curves. By letter dated April 4, 2001, Duke Energy notified the NRC of the error and requested that the NRC suspended review of the TS change submittal.

Duke Energy requests that the NRC resume review of the subject submittal incorporating the following changes.

1/ In Attachment 1 of the submittal, replace Pages 3.3.2-11, 3.3.2-12, B 3.3.2-16, B 3.3.2-17, B 3.3.2-18, Figures 3.4.3-1, 3.4.3-2, 3.4.3-3 and 3.4.3-4 with the corresponding pages as included in Attachment 1 of this supplement.

2/ In Attachment 2 of the submittal, replace Figures 3.4.3-1, 3.4.3-2, 3.4.3-3 and 3.4.3-4 with the corresponding figures as included in Attachment 2 of this supplement.

3/ In Attachment 4 of the submittal, replace Westinghouse Report WCAP-15192 with WCAP-15192 Revision 1 as included in Attachment 3 of this supplement.

4/ In Attachment 5 of the submittal, replace Westinghouse Report WCAP-15201 with WCAP-15201 Revision 1 as included in Attachment 4 of this supplement.

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Revision 1 of the WCAP reports corrects the error detected in the computer program used to generate the heatup and cooldown curves. This correction resulted in changes to the heatup curves (Figures 3.4.3-1 through 3.4.3-4) in the submittal. The cooldown curves were not affected. The revised heatup curves are slightly more restrictive but do not affect Duke Energy's low temperature overpressure protection analysis. The revised WCAPs only affect the heatup curves. Pages 3.3.2-11, 3.3.2-12, B 3.3.2-16, B 3.3.2-17, and B 3.3.2-18 in Attachment 1 of the submittal are replaced with corresponding pages to reflect the latest approved version of the McGuire Technical Specifications and corresponding Bases. The remaining attachments of the original submittal including the technical justification, no significant hazards consideration evaluation and environmental impact assessment remain valid for NRC review and approval.

If there are any questions regarding this submittal, please contact P. T. Vu at (704) 875-4302.

BWPeele

B. L. Peele

BLP/PTV/s

Attachments

xc:

L. A. Reyes
NRC Regional Administrator, Region II

S. M. Schaeffer
NRC Resident Inspector

R. E. Martin
NRC Project Manager

R. M. Frye
Director, N.C. Division of Radiation Protection

AFFIDAVIT

B. L. Peele, being duly sworn, states that he is acting Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the McGuire Nuclear Station Facility Operating Licenses Numbers NPF-9 and NPF-17 and Technical Specifications; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

B L Peele

B. L. Peele, acting Vice President

Subscribed and sworn to me: 5/30/01
Date

Notary Public: Deborah G. Thrap
Deborah G. Thrap

My Commission Expires: 4/6/2002
Date

SEAL

bxc:

L. Vaughn
M. Wilder
L. Jones
L. Burba
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A. Garrett
D. Smith (MNS-MSE)
W. Brady
T. Yadon
K. Crane
NSRB
ELL
Master File 1.3.2.9

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MCGUIRE NUCLEAR STATION

ATTACHMENT 1

MARKED-UP TECHNICAL SPECIFICATIONS PAGES

For Information Only

Table 3.3.2-1 (page 2 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Containment Isolation (continued)						
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	1 per train, 2 trains	B	SR 3.3.2.7	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
(3) Containment Pressure - High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8	≤ 3.0 psig	2.9 psig
4. Steam Line Isolation						
a. Manual Initiation						
(1) System	1,2(b),3(b)	2 trains	F	SR 3.3.2.7	NA	NA
(2) Individual	1,2(b),3(b)	1 per line	G	SR 3.3.2.7	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(b),3(b)	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
c. Containment Pressure - High High	1,2(b), 3(b)	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≤ 3.0 psig	2.9 psig
d. Steam Line Pressure						
(1) Low	1,2(b), 3(a)(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≥ 755 psig	775 psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.
(b) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 3 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
(2) Negative Rate - High	$3^{(b)(c)}$	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	$\leq 120^{(d)}$ psi	$100^{(d)}$ psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	$1,2^{(e)}$	2 trains	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - High High (P-14)	$1,2^{(e)}$	3 per SG	J	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.8 SR 3.3.2.9	$\leq$ 85.6%	83.9%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. T_{avg}-Low	$1,2^{(e)}$	1 per loop	J	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8	$\geq$ 551°F	553°F
coincident with Reactor Trip, P-4	Refer to Function 8.a (Reactor Trip, P-4) for all initiation functions and requirements.					
e. Doghouse Water Level-High High	$1,2^{(e)}$	2 per train per Doghouse	L,M	SR 3.3.2.1 SR 3.3.2.7	$\leq$13 inches	12 inches
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - Low Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	$\geq$ 15%	16.7%

(continued)

Steam Line Isolation on

(b) Except when all MSIVs are closed and de-activated.
(c) Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Safety Injection Steam Line Pressure-Low is not blocked.
(d) Time constant utilized in the rate/lag controller is $\geq$ 50 seconds.
(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

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

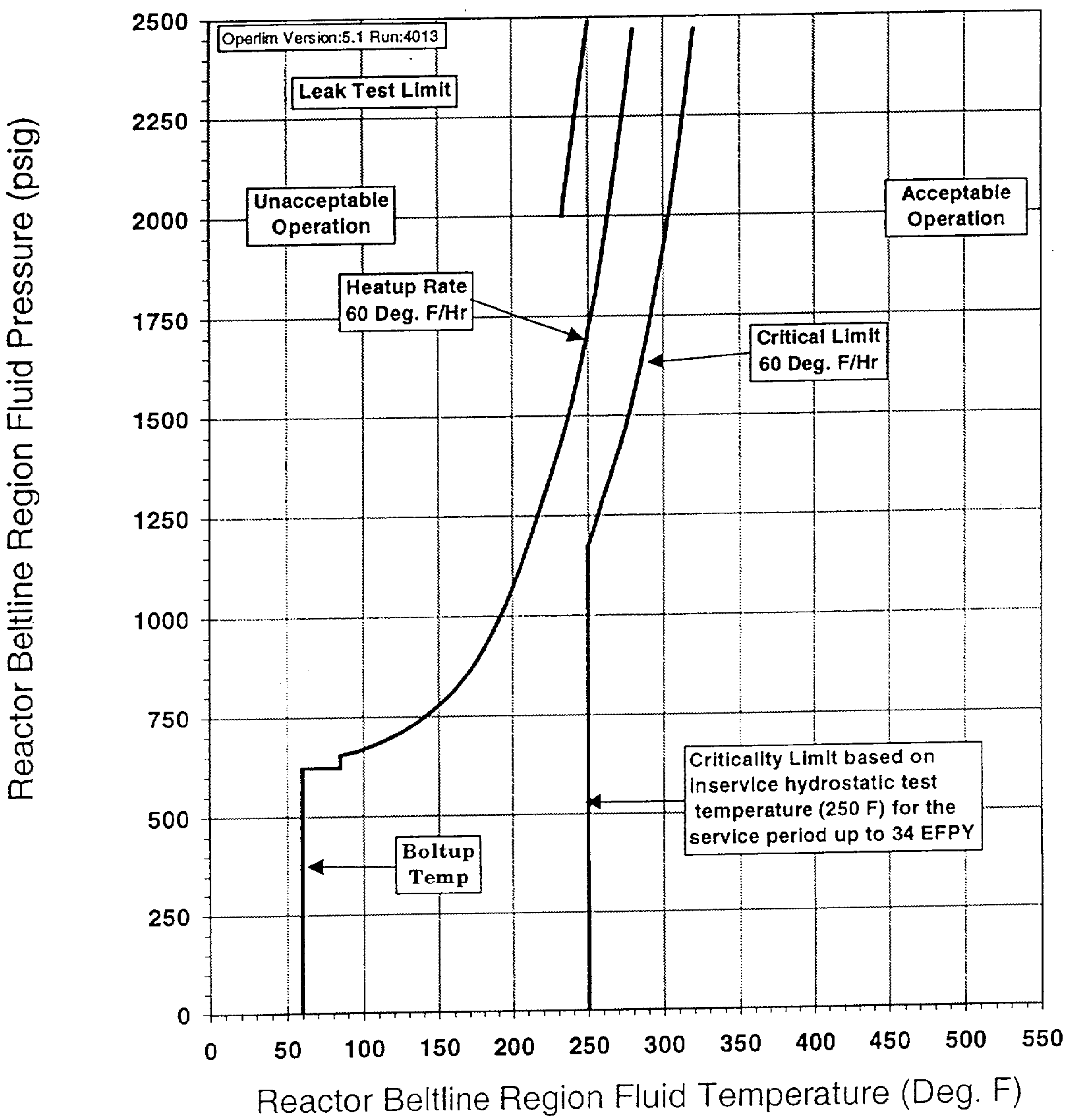

Figure 3.4.3-1

McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60° F/hr) Applicable to 34 EFPY (Without Margins 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

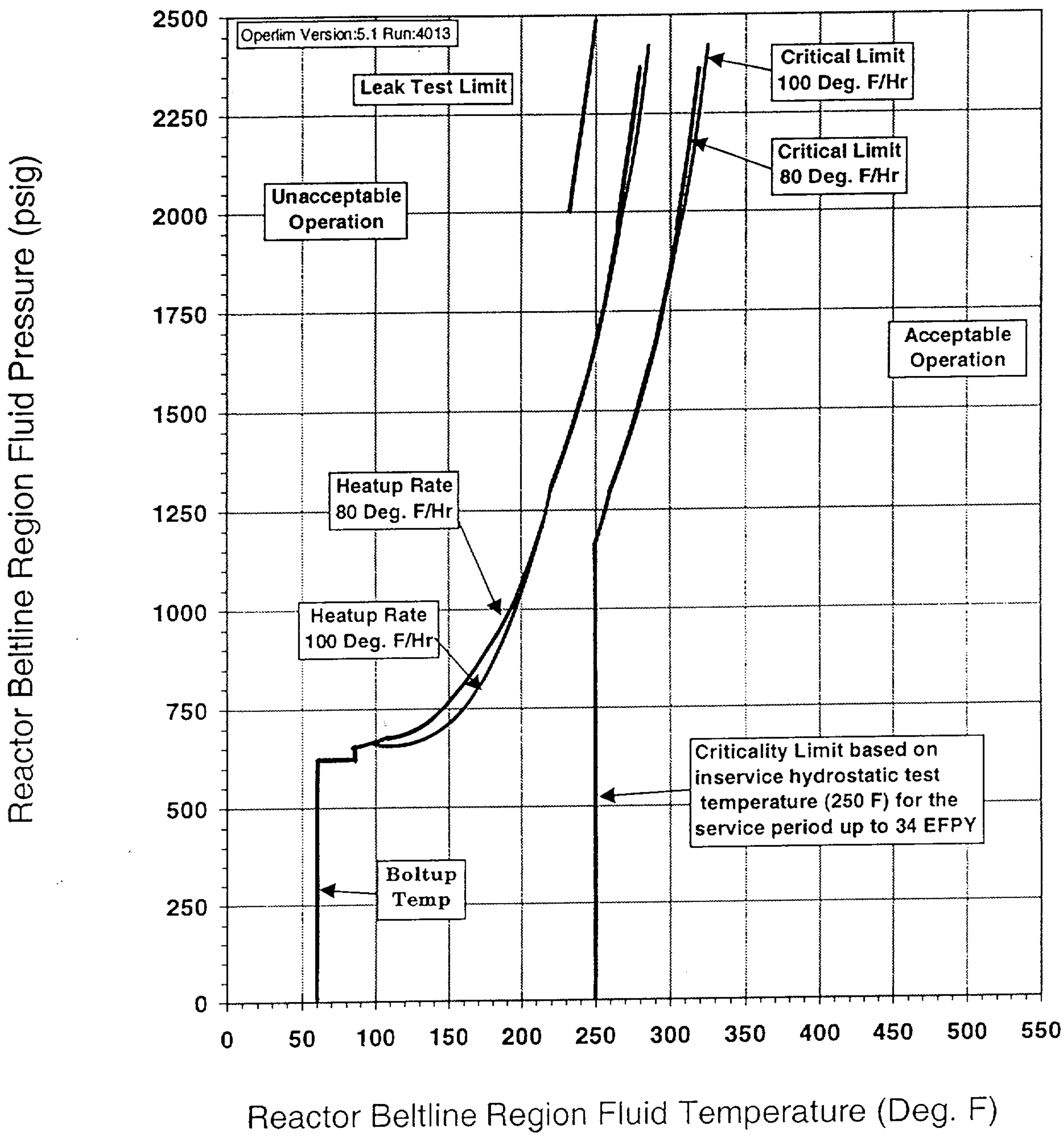

Figure 3.4.3-2

McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 and 100° F/hr) Applicable to 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

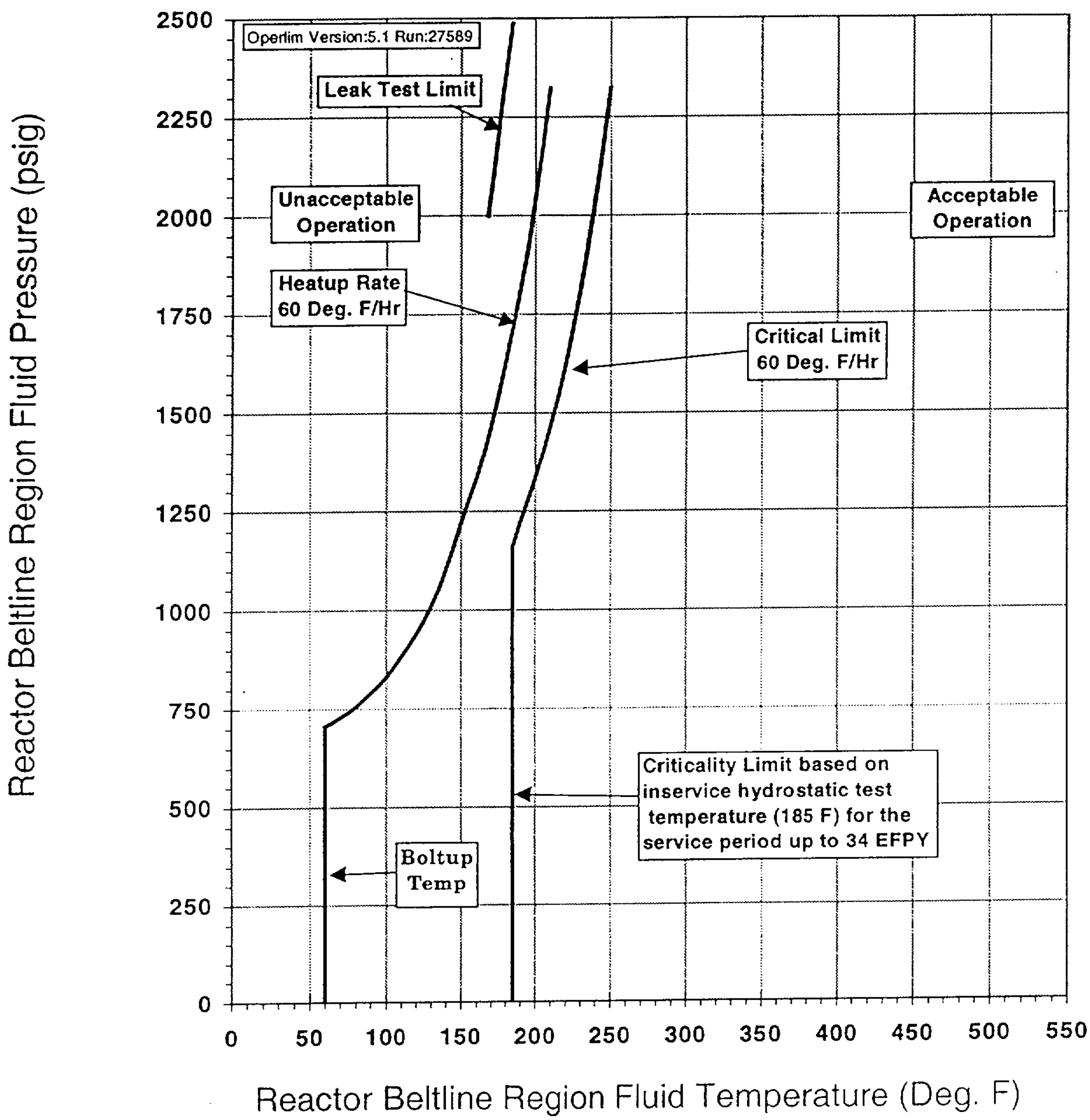

Figure 3.4.3-3

McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60° F/hr) Applicable to 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

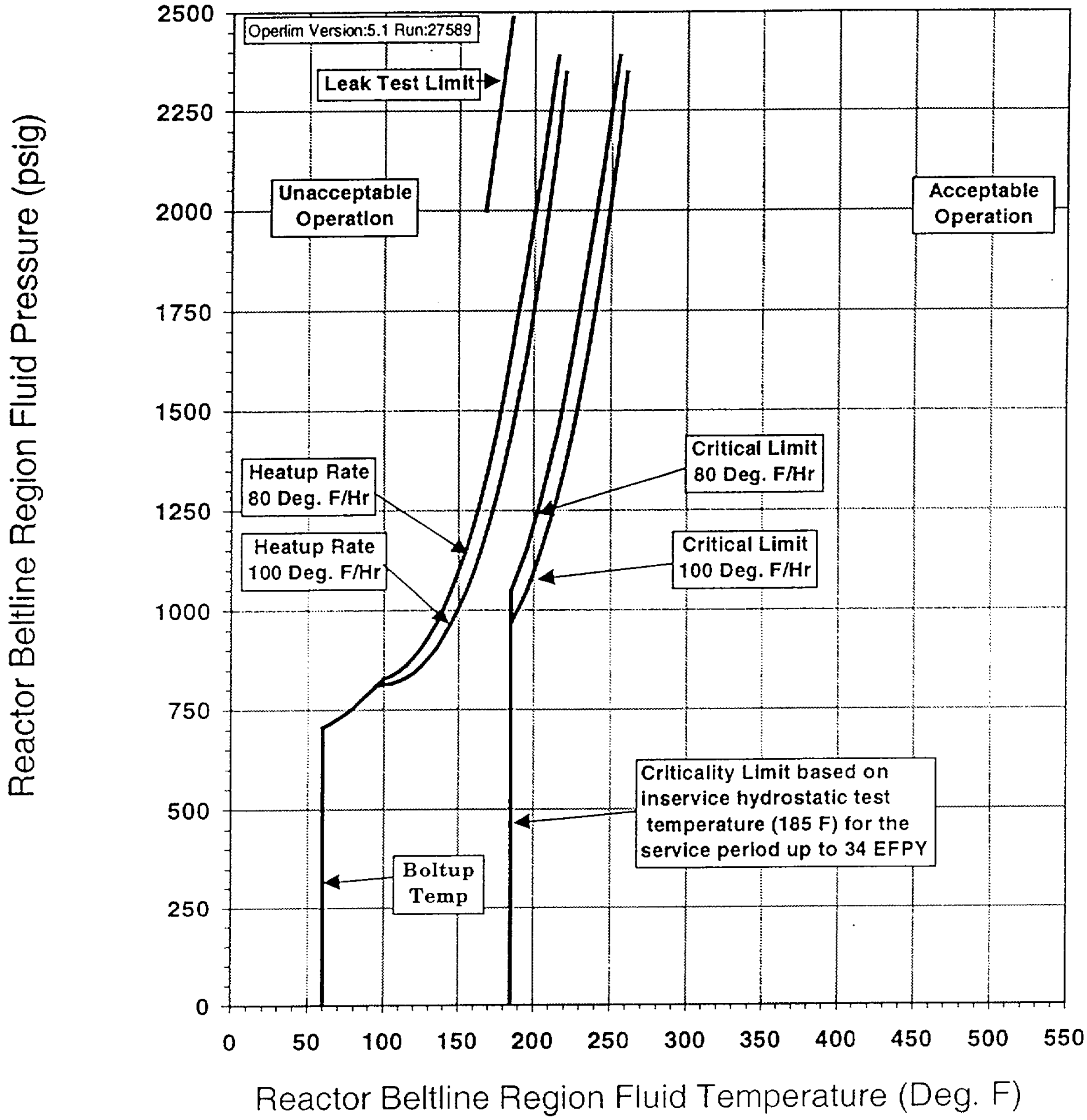

Figure 3.4.3-4

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

For Information Only

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation-Containment Pressure-High High

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain three unfaulted SGs as a heat sink for the reactor, and to limit the mass and energy release to containment. The Containment Pressure - High High function is described in ESFAS Function 2.C.

Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High High setpoint.

d. Steam Line Isolation-Steam Line Pressure

(1) Steam Line Pressure-Low

Steam Line Pressure-Low provides closure of the MSIVs in the event of an SLB to maintain three unfaulted SGs as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure - Low

Steam Line Pressure-Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure-High High. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure-Negative Rate-High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) <u>Steam Line Pressure-Negative Rate-High</u>

Steam Line Pressure-Negative Rate-High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure-Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate-High signal is automatically enabled. Steam Line Pressure-Negative Rate-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

Steam Line Pressure-Negative Rate-High must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure-Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the primary and secondary sides to have an SLB or other accident that would result in a release of significant enough quantities of energy to cause a cooldown of the RCS.

5. Turbine Trip and Feedwater Isolation

The primary functions of the Turbine Trip and Feedwater Isolation signals are to prevent damage to the turbine due to water in the steam lines, and to stop the excessive flow of feedwater into the SGs. These Functions are necessary to mitigate the effects of a high water level in the SGs, which could result in carryover of water into the steam lines and excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

The function is actuated when the level in any SG exceeds the high high setpoint, and performs the following functions:

- Trips the main turbine;
- Trips the MFW pumps; and
- Initiates feedwater isolation (shuts the MFW control valves, bypass feedwater control valves, feedwater isolation valves, and the MFW to AFW nozzle bypass valves).

Turbine Trip and Feedwater Isolation signals are both actuated by SG Water Level-High High, or by an SI signal. The RTS also initiates a turbine trip signal whenever a reactor trip (P-4) is generated. A Feedwater Isolation signal is also generated on a high water level in the reactor building doghouses. In the event of SI, the unit is taken off line and the turbine generator must be tripped. The MFW System is also taken out of operation and the AFW System is automatically started. The SI signal was discussed previously.

a. Turbine Trip and Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

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ATTACHMENT 2

NEW TECHNICAL SPECIFICATION PAGES

<u>MATERIAL PROPERTY BASIS</u>

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

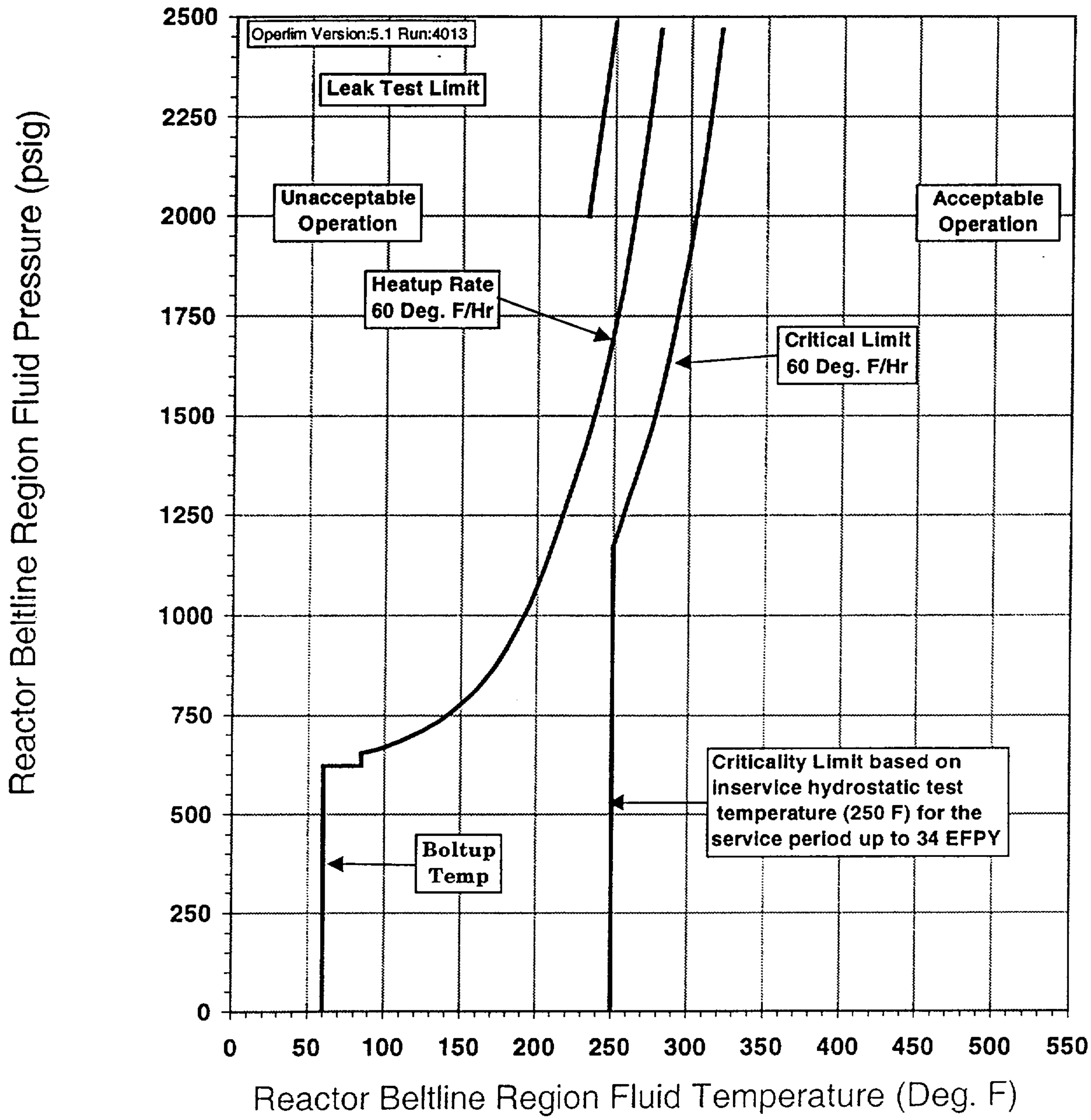

Figure 3.4.3-1

McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60° F/hr) Applicable to 34 EFPY (Without Margins 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

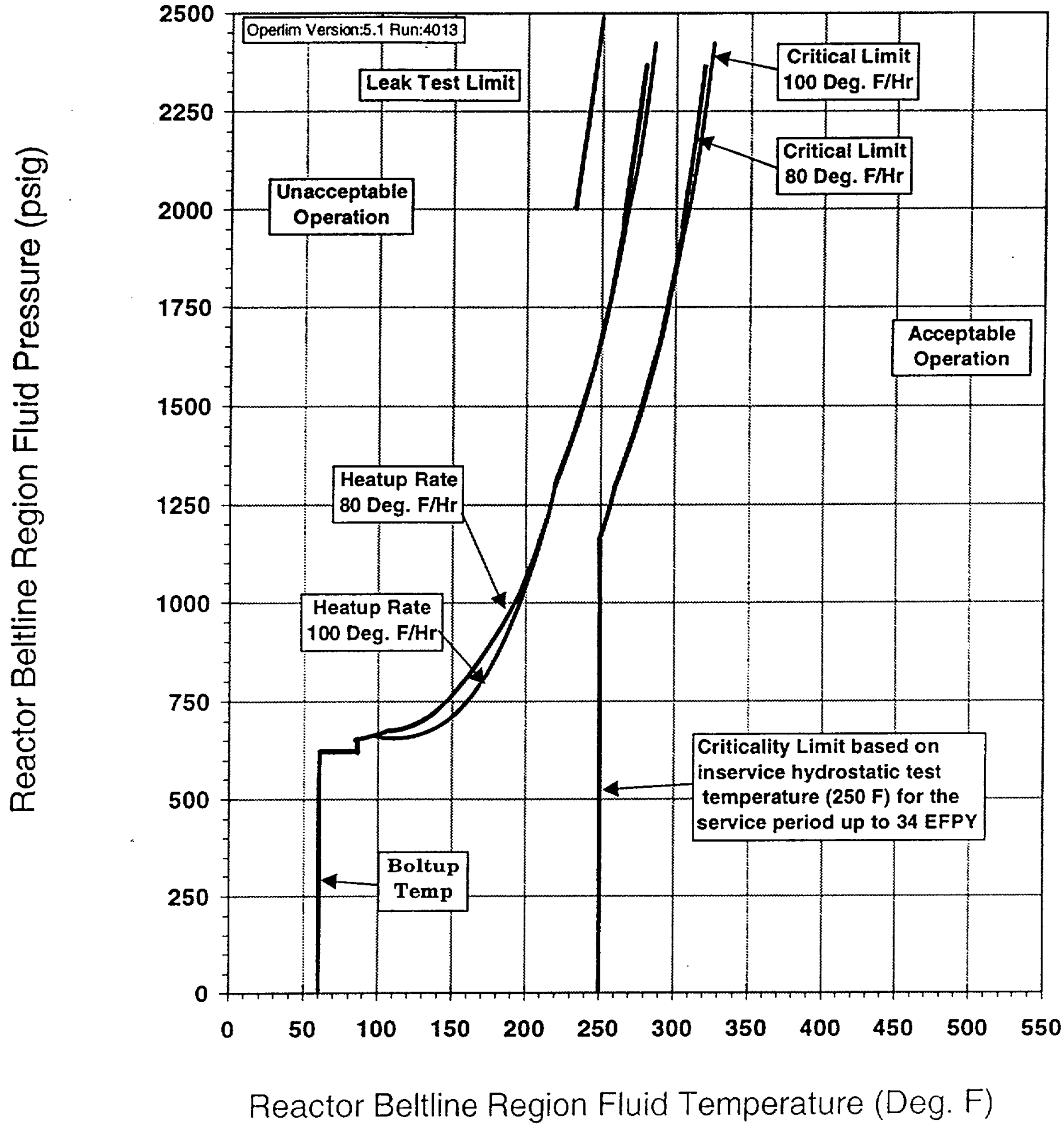

Figure 3.4.3-2

McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 and 100° F/hr) Applicable to 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

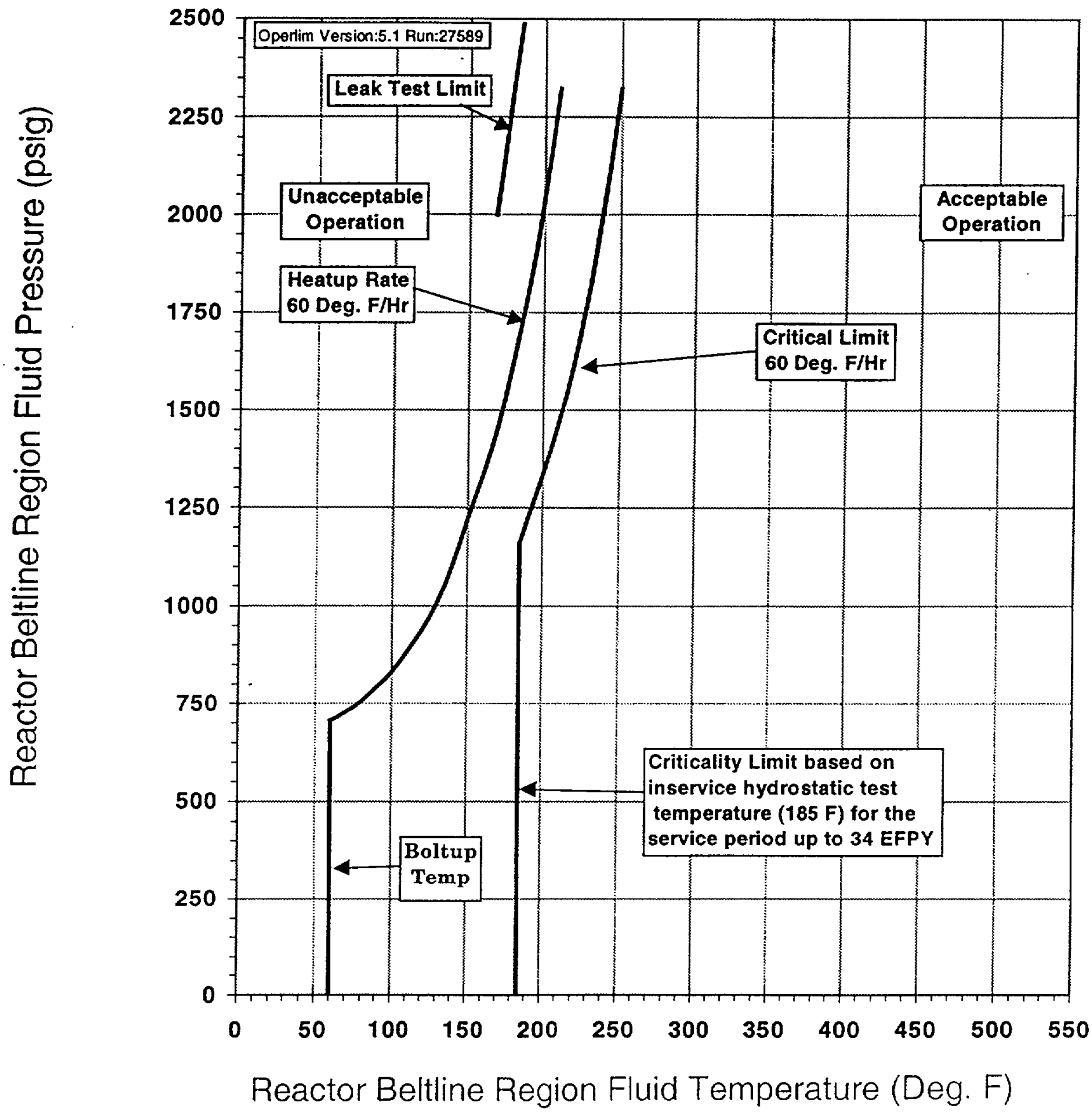

Figure 3.4.3-3

McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60° F/hr) Applicable to 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

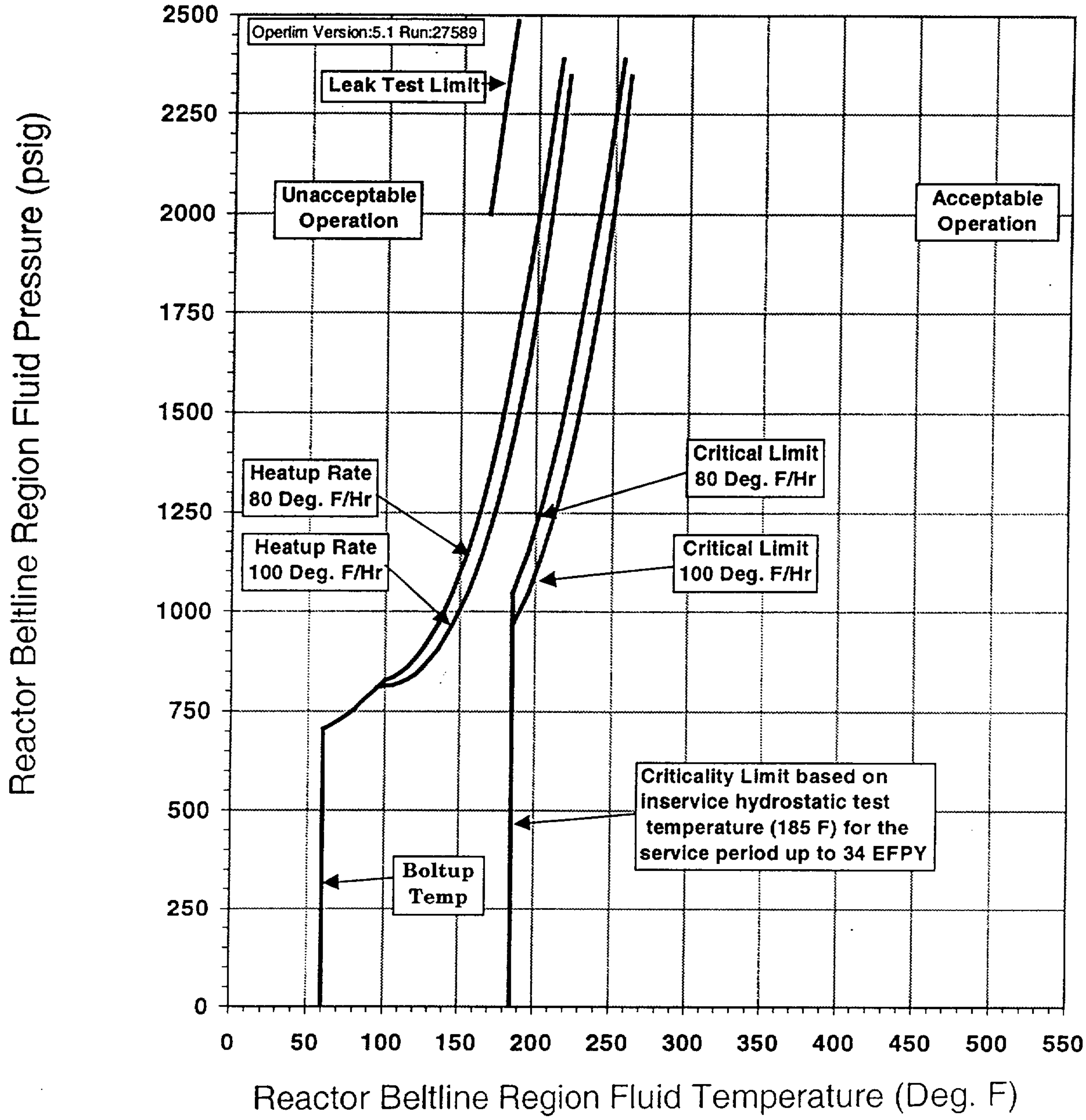

Figure 3.4.3-4

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

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION

ATTACHMENT 3

WCAP-15192 REVISION 1

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15192, Revision 1

McGuire Unit 1
Heatup and Cooldown Limit Curves
For Normal Operation

J. H. Ledger

April 2001

Prepared by the Westinghouse Electric Company LLC
for the Duke Power Company

Approved: [signature] FOR
C. H. Boyd, Manager
Engineering and Materials Technology

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

PREFACE

This report has been technically reviewed and verified by:

T. J. Laubham ______

Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15192 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15192 Rev. 0.

Note that only the heatup curves and associated data point tables have changed. The cooldown curves and data points remain valid and were not changed.

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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LIST OF FIGURES

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.

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 temperature at the tip of the assumed flaw. During cooldown, the ¼T vessel location is at a higher temperature than the fluid adjacent to the vessel inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the 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.

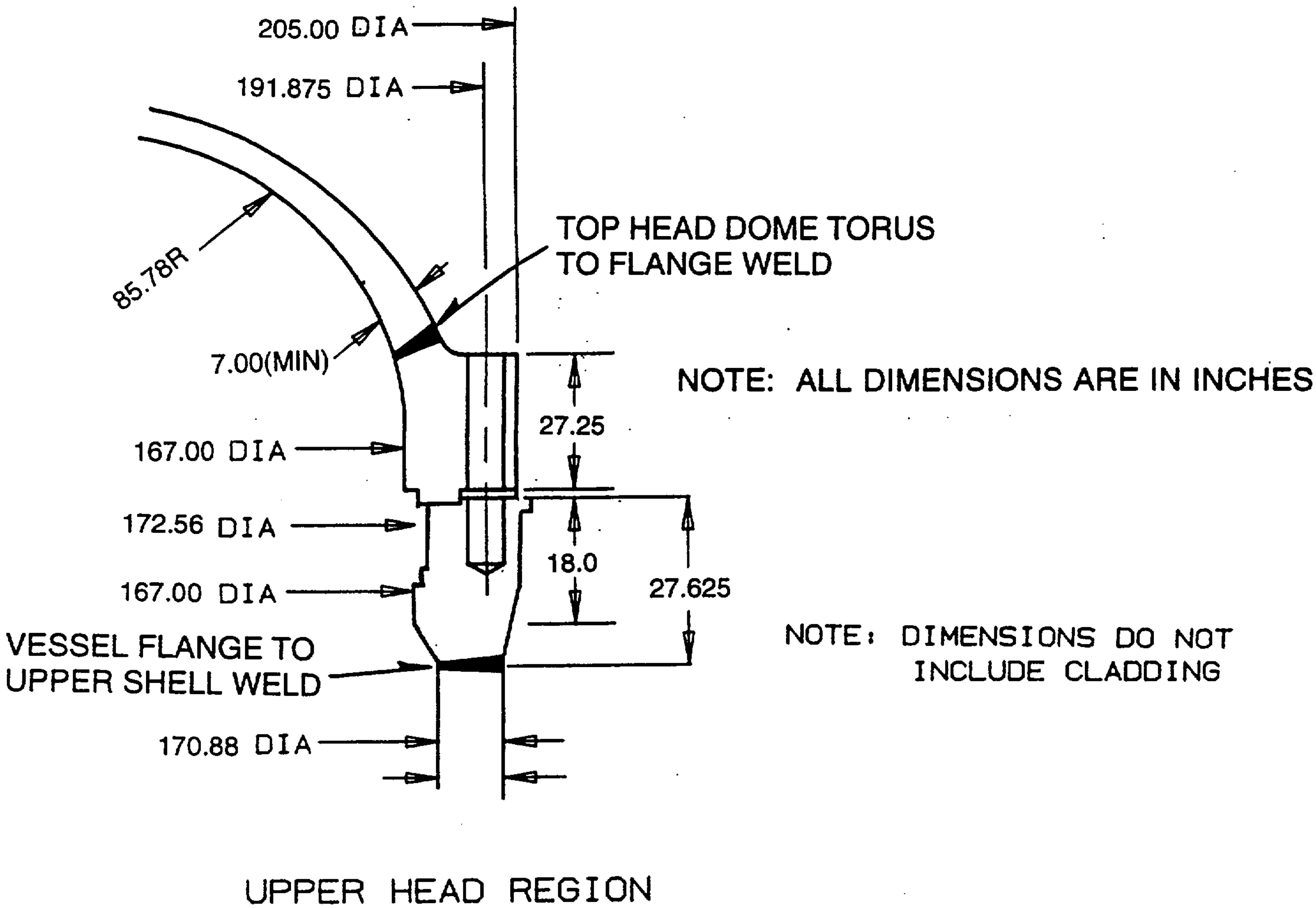

Figure 3-1 Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

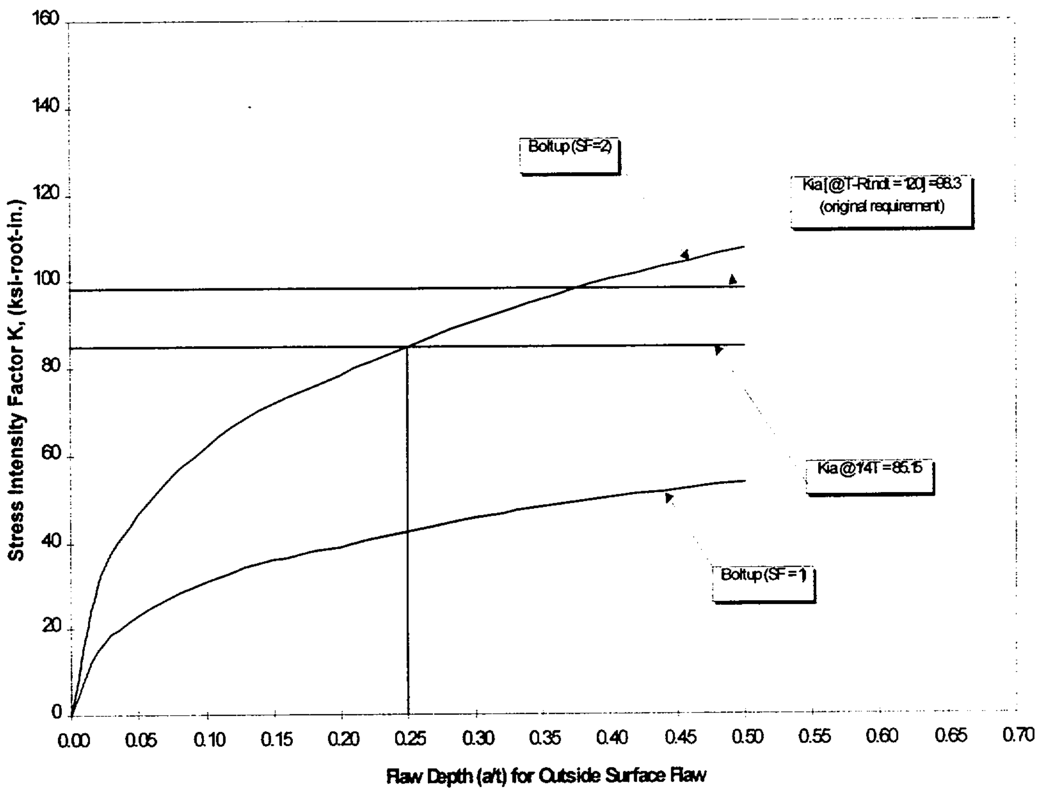

Figure 3-2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

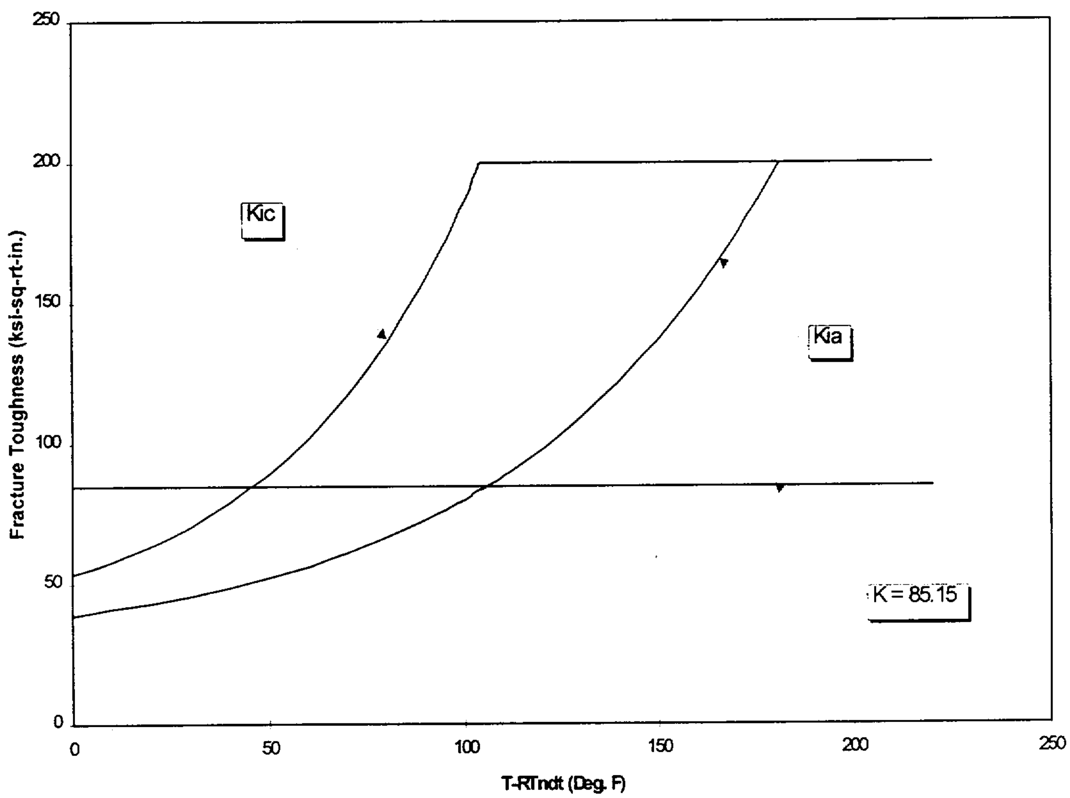

Figure 3-3 Determination of Boltup Requirement, using K_{IC}

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
	$CF_{B5012-1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (607.89) \div (9.70) = 62.7°F$					
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/cm^2, 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: ¼T, 190°F
¾T, 129°F

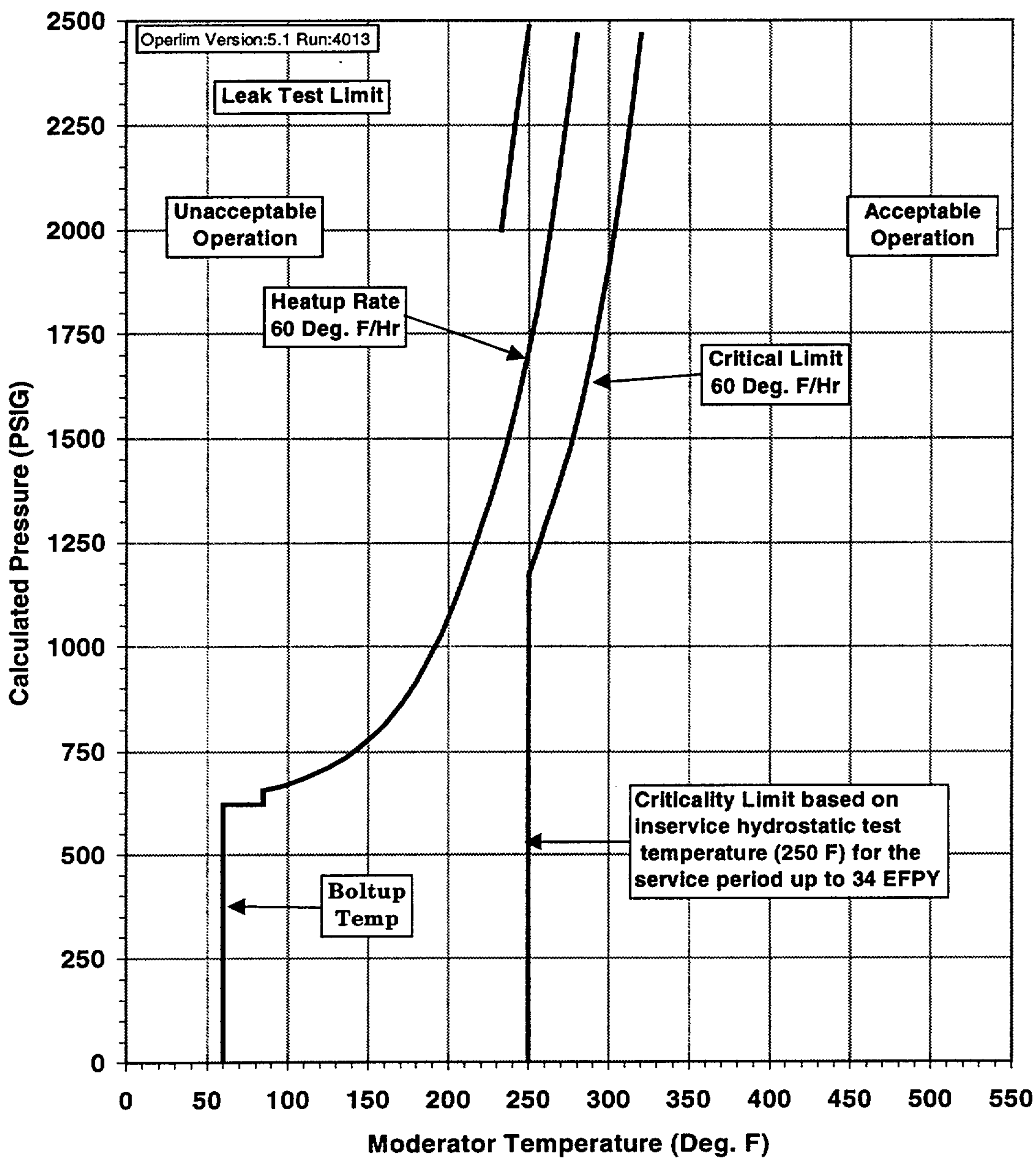

FIGURE 5-1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable to 34 EFPY (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDIANL WELD SEAMS 3-442A & C

LIMITING ART VALUES AT 34 EFPY: ¼T, 190°F
¾T, 129°F

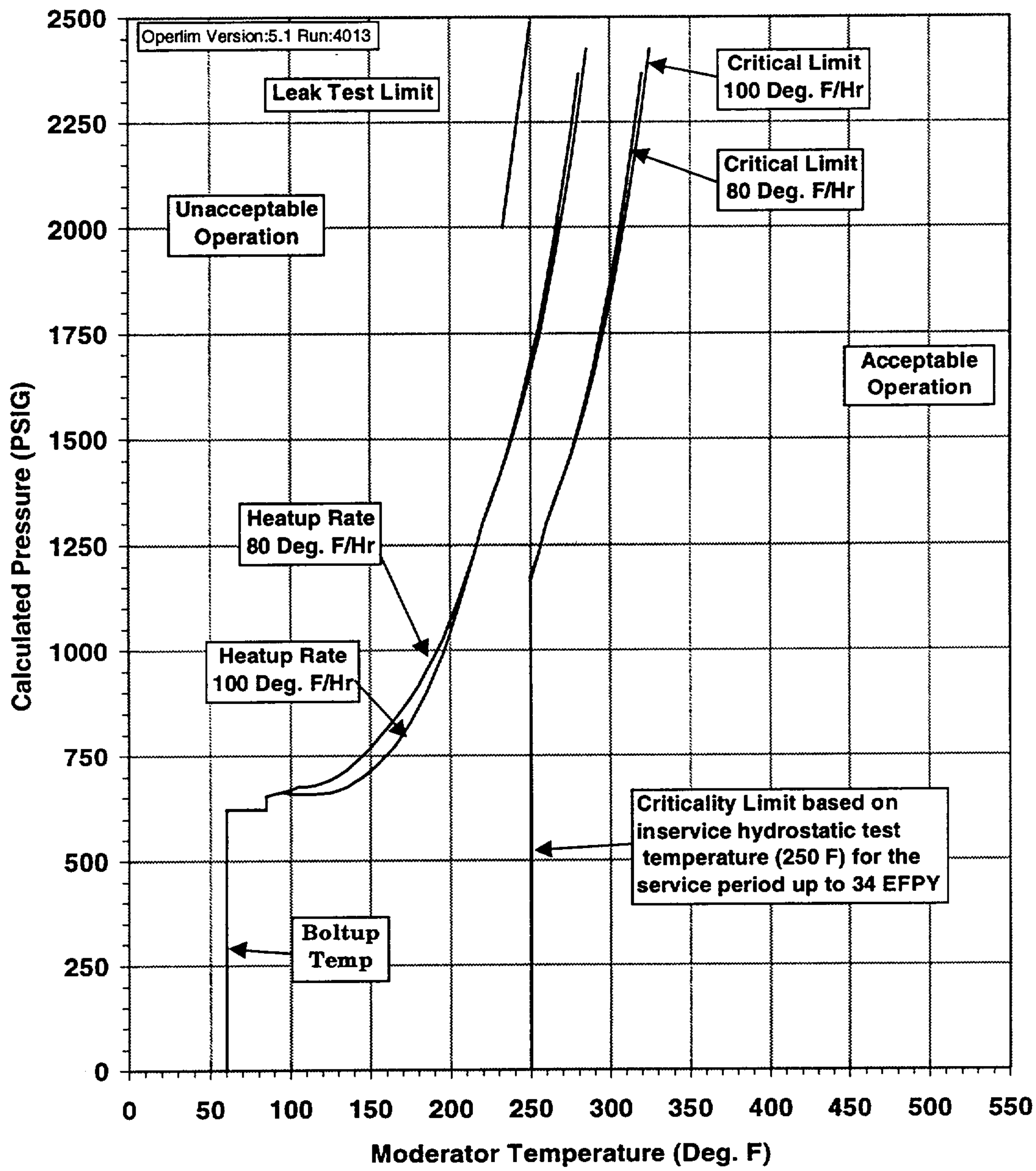

FIGURE 5-2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 and 100°F/hr) Applicable to 34 EFPY (Without Margins of for Instrumentation Errors)

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

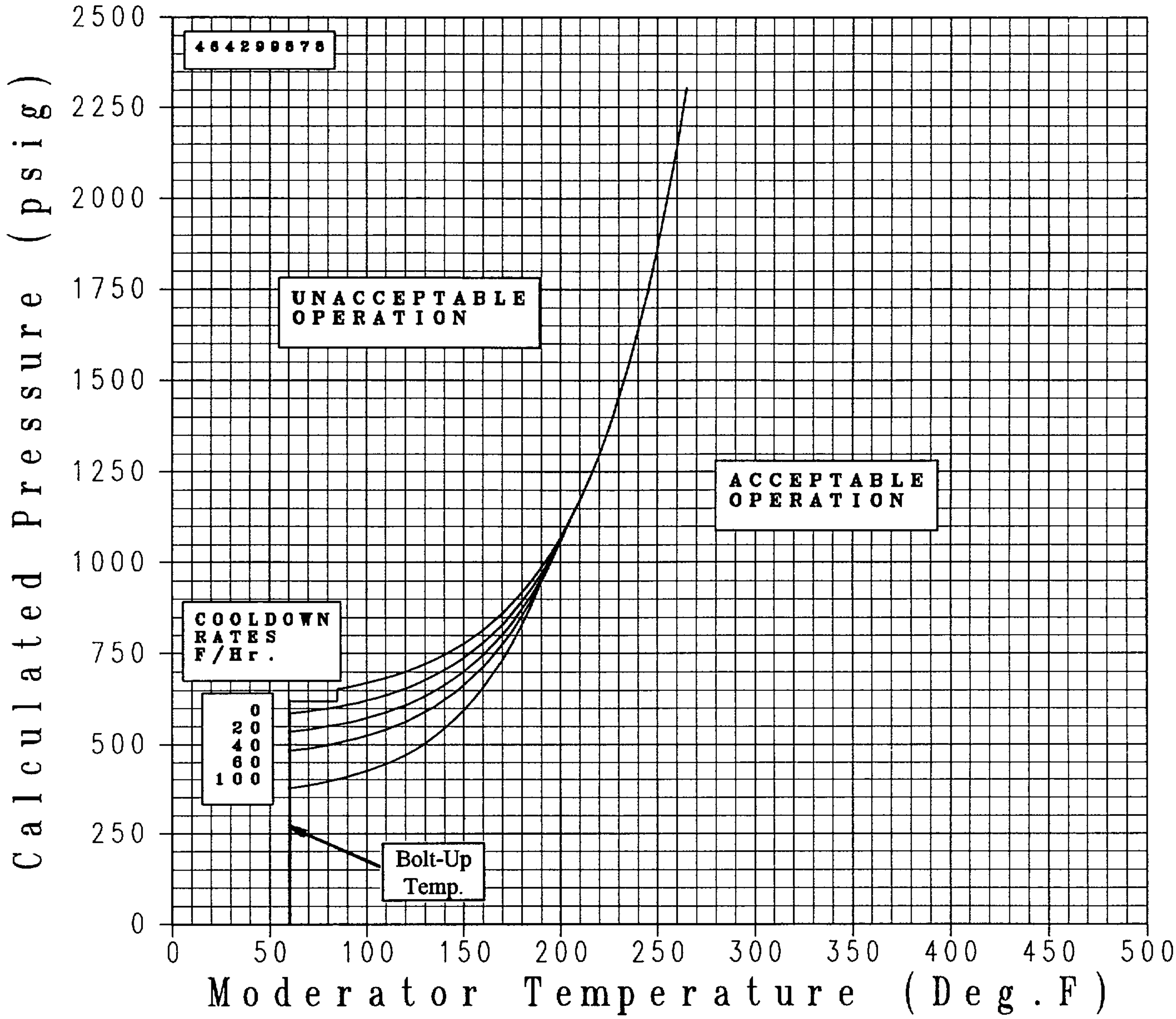

FIGURE 5-3 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable to 34 EFPY (Without Margins for Instrumentation Errors)

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	621	250	2485
65	621	250	621		
70	621	250	694		
75	621	250	695		
80	621	250	697		
85	621	250	699		
85	621	250	702		
85	654	250	708		
90	658	250	710		
95	664	250	721		
100	670	250	734		
105	676	250	751		
110	684	250	770		
115	692	250	791		
120	701	250	816		
125	711	250	844		
130	721	250	876		
135	733	250	911		
140	747	250	941		
145	761	250	960		
150	778	250	981		
155	795	250	1004		
160	815	250	1029		
165	837	250	1057		
170	861	250	1087		
175	888	250	1121		
180	918	250	1157		
185	950	250	1173		
190	986	255	1232		
195	1026	260	1291		
200	1070	265	1345		
205	1119	270	1405		
210	1173	275	1470		
215	1232	280	1543		
220	1291	285	1622		
225	1345	290	1711		

TABLE 5-1 con't
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.	Press.	Temp.	Press.	Temp.	Press.
(°F)	(psig)	(°F)	(psig)	(°F)	(psig)
230	1405	295	1808		
235	1470	300	1915		
240	1543	305	2033		
245	1622	310	2164		
250	1711	315	2308		
255	1808	320	2467		
260	1915				
265	2033				
270	2164				
275	2308				
280	2467				

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	621	60	621	250	621	250	2485
65	621	250	621	65	621	250	621		
70	621	250	677	70	621	250	659		
75	621	250	677	75	621	250	660		
80	621	250	679	80	621	250	660		
85	621	250	679	85	621	250	663		
85	621	250	685	85	621	250	664		
85	654	250	685	85	654	250	668		
90	658	250	692	90	658	250	671		
95	664	250	694	95	659	250	676		
100	670	250	702	100	659	250	680		
105	676	250	706	105	659	250	686		
110	677	250	715	110	659	250	692		
115	679	250	730	115	659	250	698		
120	685	250	748	120	660	250	707		
125	692	250	769	125	663	250	713		
130	702	250	793	130	668	250	731		
135	715	250	820	135	676	250	751		
140	730	250	850	140	686	250	774		
145	748	250	885	145	698	250	801		
150	769	250	923	150	713	250	830		
155	793	250	966	155	731	250	864		
160	815	250	1013	160	751	250	902		
165	837	250	1066	165	774	250	944		
170	861	250	1110	170	801	250	990		
175	888	250	1141	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	1347	200	1042	265	1353		
205	1119	270	1402	205	1100	270	1403		
210	1173	275	1461	210	1164	275	1458		
215	1232	280	1527	215	1232	280	1518		
220	1298	285	1600	220	1298	285	1584		
225	1347	290	1680	225	1353	290	1658		

TABLE 5-2 (Cont.)
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	1402	295	1768	230	1403	295	1738		
235	1461	300	1865	235	1458	300	1826		
240	1527	305	1972	240	1518	305	1924		
245	1600	310	2091	245	1584	310	2031		
250	1680	315	2221	250	1658	315	2149		
255	1768	320	2365	255	1738	320	2280		
260	1865			260	1826	325	2423		
265	1972			265	1924				
270	2091			270	2031				
275	2221			275	2149				
280	2365			280	2280				
				285	2423				

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.

3 1996 ASME Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure", December 1996.

4 ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 1999.

5 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

6 WCAP-13949, "Analysis of Capsule V Specimens and Dosimeters and Analysis of Capsule Z Dosimeters from the Duke Power Company McGuire Unit 1 Reactor Vessel Radiation Surveillance Program", Ed Terek, et al., February 1994.

7 WCAP-14993, "Analysis of Capsule Y from the Duke Power McGuire Unit 1 Reactor Vessel Radiation Surveillance Program", T. J. Laubham, et al., December 1997.

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 4.1, developed by ATI Consulting, March 1996.

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

11 WCAP-10786, "Analysis of Capsule U from the Duke Power McGuire Unit 1 Reactor Vessel Radiation Surveillance Program," S. E. Yanichko, et. al., February, 1985.

12 WCAP-14363, "Analysis of Capsule Y for Pacific Gas and Electric Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," P. A. Peter, et. al., August 1995.

13 Duke Energy Letter to NRC from H. B. Barron Dated 1/7/99 which References Technical Report No. ATI-98-012-T005, Revision 1, "Evaluation of McGuire Unit 1 Surveillance Weld Data Credibility", Dated November 1998, By Tim Greisbach.

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION

ATTACHMENT 4

WCAP-15201 REVISION 1

WCAP-15201, Revision 1

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

J. H. Ledger

April 2001

Prepared by the Westinghouse Electric Company LLC
for the Duke Power Company

Approved: J. M. B... FOR
C. H. Boyd, Manager
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PREFACE

This report has been technically reviewed and verified by:

T. J. Laubham ______

Revision 1:

An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15201 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15201 Rev. 0.

Note that only the heatup curves and associated data point tables have changed, however the 100 Deg. F/Hr. Heatup Rates were not affected by this error for McGuire Unit 2. The cooldown curves and data points remain valid and were not changed.

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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.

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√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√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√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.

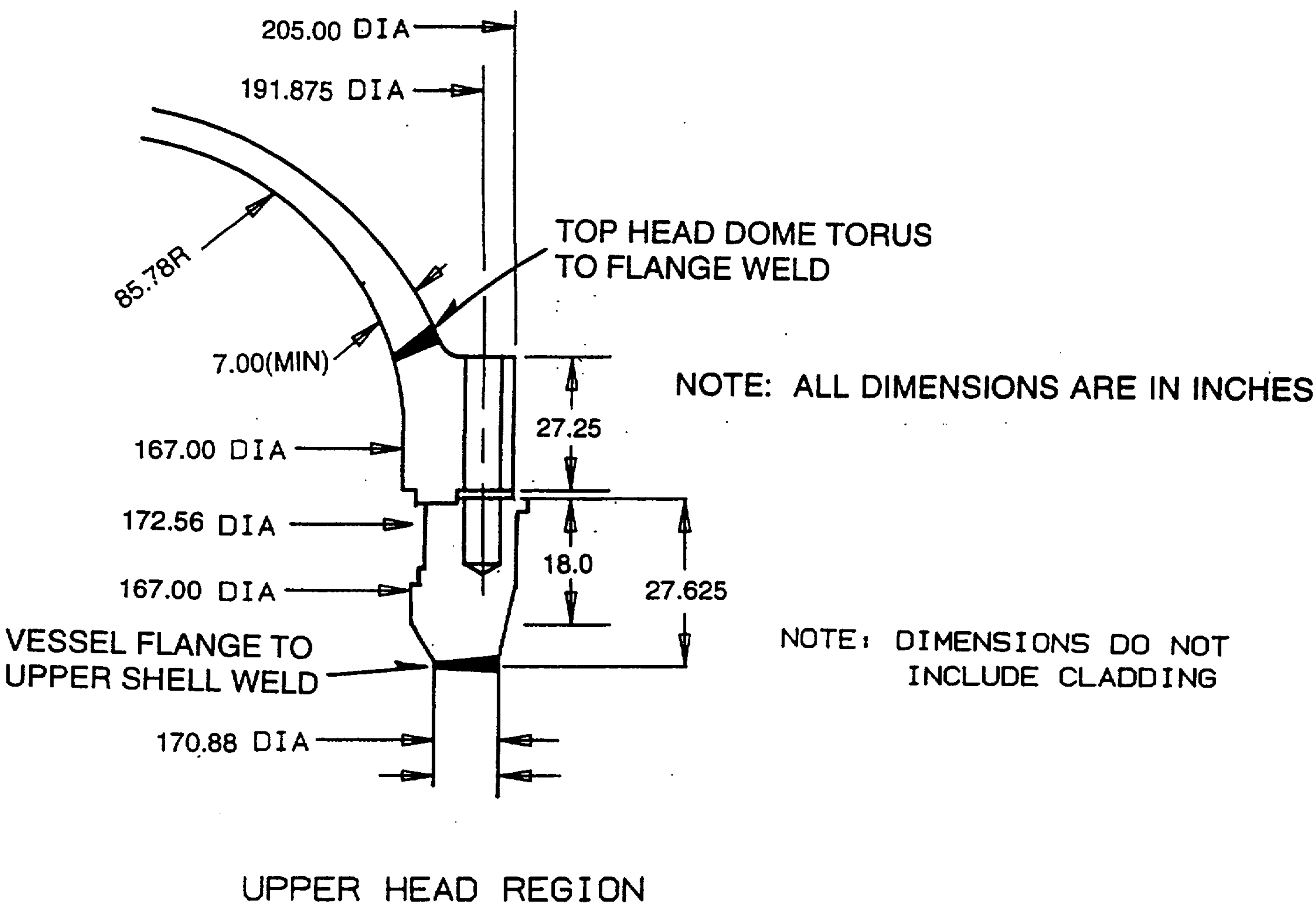

Figure 1 Geometry of the Upper Head/Flange Region of a Typical Westinghouse Four Loop Plant Reactor Vessel

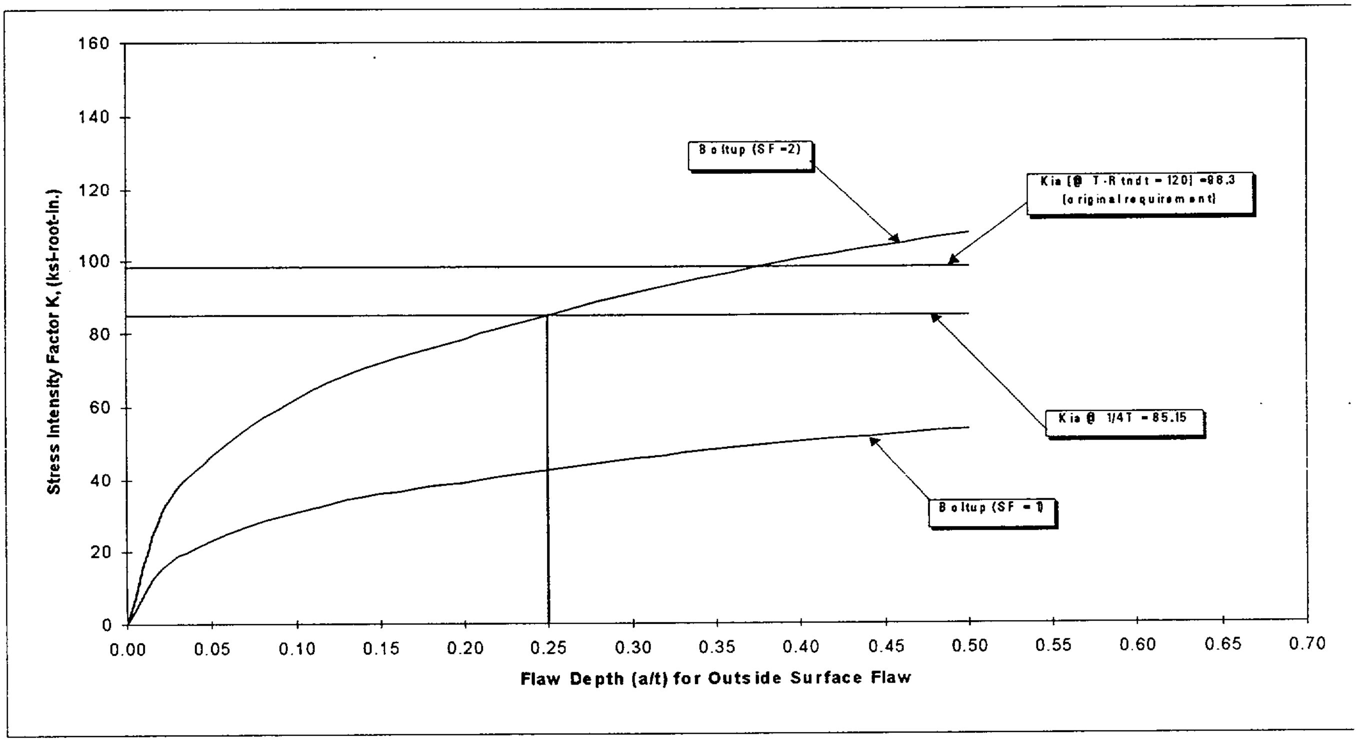

Figure 2 Crack Driving Force as a Function of Flaw Size: Outside Surface Flaw in the Closure Head to Flange Region Weld

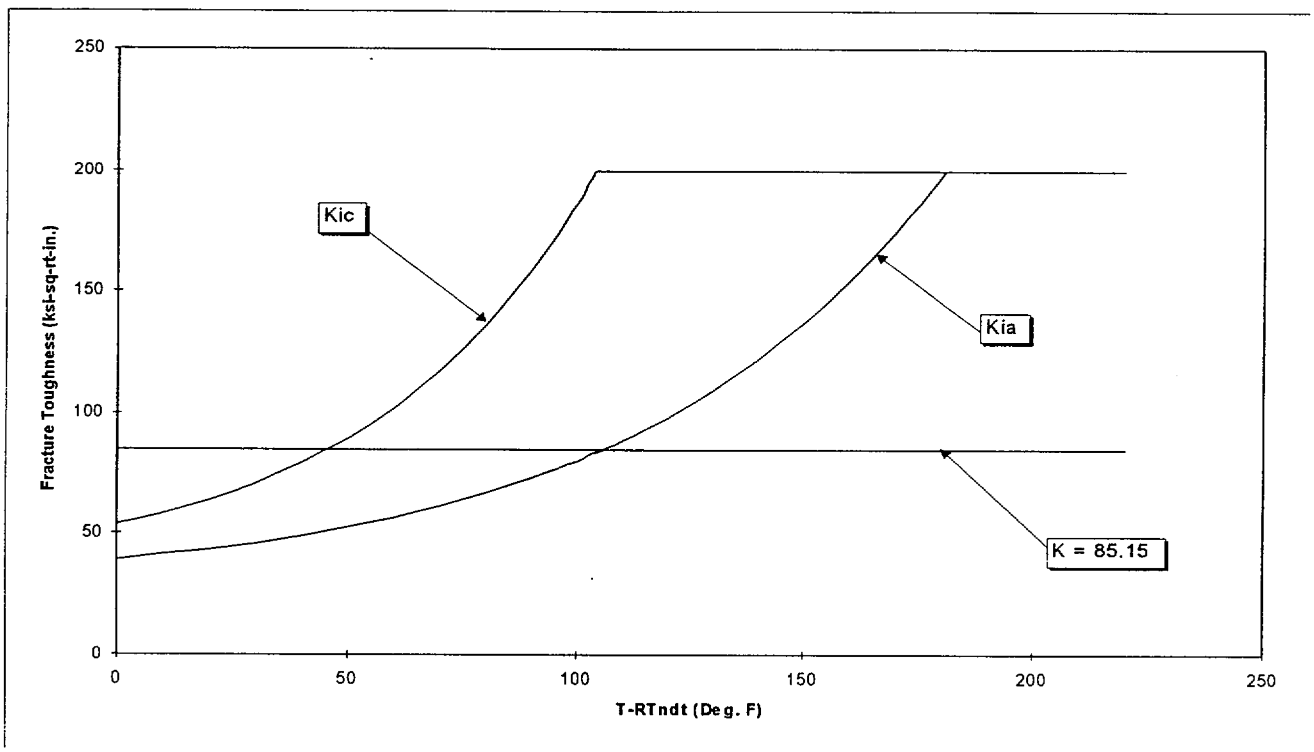

Figure 3 Determination of Boltup Requirement, Using K_{Ic}

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:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (7)$$

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:

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

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.

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

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	1.93×10^{19}	1.16×10^{19}	4.21×10^{18}

* Clad/Base Metal Interface

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

Capsule	Fluence
V	3.268×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.406×10^{19} n/cm^2, (E > 1.0 MeV)
U	1.962×10^{19} n/cm^2, (E > 1.0 MeV)
W	2.969×10^{19} n/cm^2, (E > 1.0 MeV)
Z*	2.348×10^{19} n/cm^2, (E > 1.0 MeV)
Y*	1.967×10^{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 Forging 05 (Axial)	V	0.3268	0.692	58.64	40.6	0.479
	X	1.406	1.09	91.12	99.3	1.19
	U	1.962	1.18	84.14	99.3	1.39
	W	2.969	1.29	130.33	168.1	1.66
Intermediate Shell Forging 05 (Tangential)	V	0.3268	0.692	68.97	47.7	0.479
	X	1.406	1.09	98.28	107.1	1.19
	U	1.962	1.18	91.18	107.6	1.39
	W	2.969	1.29	102.03	131.6	1.66
				SUM	801.3	9.44
	$CF_{Forging\ 05} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (801.3) \div (9.44) = 84.9°F$					
Circumferential Weld Seam[4]	V	0.3268	0.692	38.51	26.6	0.479
	X	1.406	1.09	35.93	39.2	1.19
	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)