Report No.: SIR-05-044
Revision No.: 0
Project No.: GE-10Q
File No.: GE-10Q-401
December 2005

Pressure-Temperature Limits Report Methodology for Boiling Water Reactors

Prepared for:

The BWR Owners' Group
(Under Contract to GE Nuclear Energy,
Contract Number 431003505 dated 01/13/2005)

Prepared by:

Structural Integrity Associates, Inc.
Centennial, Colorado

Prepared by: Gary L. Stevens — Gary L. Stevens, P. E. Date: 12/13/2005

Reviewed by: Keith R. Evon — K. R. Evon Date: 12/13/2005

Approved by: Gary L. Stevens — Gary L. Stevens, P. E. Date: 12/13/2005

REVISION CONTROL SHEET

Document Number: SIR-05-044

Title: Pressure-Temperature Limits Report Methodology for Boiling Water Reactors

Client: BWR Owners' Group (under contract to GE Nuclear Energy)

SI Project Number: GE-10Q

Section	Pages	Revision	Date	Comments
1 2 3 4 Appendix A	1-1 – 1-2 2–1 – 2-33 3-1 – 3-2 4-1 – 4-2 A-1	A	02/18/05	Initial Draft (Unverified) for Review.
--- 1 2 3 4 Appendix A Appendix B	i – vii 1-1 – 1-2 2–1 – 2-33 3-1 – 3-2 4-1 – 4-2 A-1 – A-17 B-1, Cover, TOC, 1-28	B	09/15/05	Revised Draft (Verified, GE Comments Incorporated) for BWROG Committee Review.
--- 1 2 3 4 Appendix A Appendix B	i – vii 1-1 – 1-3 2–1 – 2-33 3-1 – 3-2 4-1 – 4-2 A-1 – A-17 B-1, Cover, TOC, 1-28	C	10/04/05	Revised Draft (Verified, BWROG Committee Comments Incorporated).
--- 1 2 3 4 Appendix A Appendix B	i – vii 1-1 – 1-3 2–1 – 2-33 3-1 – 3-2 4-1 – 4-2 A-1 – A-17 B-1, Cover, TOC, 1-28	0	12/13/05	Initial issue (Verified, BWROG Prime Rep. Comments Incorporated).

FOREWORD

The principle objective of this report is to provide, for submittal to the NRC, the current Structural Integrity Associates, Inc. methodology for developing reactor coolant system (RCS) pressure test, core not critical, and core critical curves for BWRs at the request of the BWR Owners' Group (BWROG) Pressure-Temperature Curve Committee. When approved by the NRC, this methodology may be referenced by licensees to implement the Pressure Temperature Limits Report (PTLR). This report does not provide all of the methodologies which can be used to develop RCS pressure test, core not critical, and core critical curves, but rather methodologies that can be referenced by licensees when approved by the NRC to license the PTLR concept.

LEGAL NOTICE

IMPORTANT NOTICE REGARDING THE CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of Structural Integrity Associates, Inc. (SI) respecting information in this document are contained in the contract between SI and the General Electric Company (GE), as agent for the Boiling Water Reactor Owners' Group (BWROG), as identified in the Purchase Order for the performance of the work described herein, and nothing in this document shall be construed as changing that contract. The use of this information, except as defined by said contract, or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, SI nor any of the contributors to this document, makes any representation or warranty, expressed or implied, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

QUALITY ASSURANCE

This report was prepared by Structural Integrity Associates, Inc. (SI) for the BWROG Pressure-Temperature Curve Committee in accordance with the SI Quality Assurance Program, which is in compliance with the requirements of 10CFR50, Appendix B and ANSI/ASME NQA-1-1989, and meets the intent of applicable portions of ANSI N45.2. Since the work associated with this report is classified as safety-related, the provisions of 10 CFR 21 and 10 CFR 50, Appendix B apply. However, users are reminded that, prior to application of any information contained in this document to any safety-related application at a specific nuclear plant, the generic information contained in this document must also be verified as applicable to a specific nuclear plant through the user's own Quality Assurance Program.

PARTICIPATING UTILITIES

The utilities listed below contributed to the development of this report. However, while this report has been endorsed by a substantial number of the members of the BWR Owners' Group, it should not be interpreted as a commitment of any individual member to a specific course of action. Each member must formally endorse any BWROG position in order for that position to become the member's position.

UTILITY		PLANT
AmerGen		Clinton Oyster Creek
Energy Northwest		Columbia
Entergy		Pilgrim FitzPatrick Vermont Yankee
Exelon		Dresden LaSalle Quad Cities Limerick Peach Bottom
FENOC		Perry
NMC		Duane Arnold Monticello
NPPD		Cooper
PPL		Susquehanna
Progress Energy		Brunswick
PSEG Nuclear		Hope Creek
SNC		Hatch
TVA		Browns Ferry

DOCUMENT DISTRIBUTION

This BWROG report, and any associated products, are proprietary to the BWROG and the utilities that financially participated in their development. Recipients of this document have no authority or rights to release this product or any information contained herein to anyone or any organization outside their utility, restricted to the specific plants identified in "Participating Utilities" above. This product can, however, be shared with contractors performing related work directly for a participating plant, conditional upon appropriate proprietary agreements being in place with the contractor protecting this BWROG product or any information contained herein against unauthorized distribution or use.

Table of Contents

List of Figures

1.0 INTRODUCTION

1.1 Background

Nuclear Regulatory Commission (NRC) Generic Letter (GL) 96-03 [1] allows plants to relocate their pressure-temperature (P-T) curves and numerical values of other P-T limits (such as heatup/cooldown rates) from the plant Technical Specifications to a Pressure Temperature Limits Report (PTLR), which is a licensee-controlled document. As stated in GL 96-03, during the development of the improved standard technical specifications (STS), a change was proposed to relocate the P-T limits currently contained in the plant Technical Specifications to a PTLR. As one of the improvements to the STS, the NRC staff agreed with the industry that the curves may be relocated outside the plant Technical Specifications to a PTLR so that the licensee could maintain these limits efficiently and at a lower cost. One of the prerequisites for having the PTLR option is that all of the methods used to develop the P-T curves and limits be NRC approved, and that the associated Licensing Topical Report (LTR) for such approval is referenced in the plant Technical Specifications. Based on this prerequisite, the purpose of this report is to provide boiling water reactors (BWRs) with an NRC-approved LTR that can be referenced in plant Technical Specifications to establish BWR fracture mechanics methods for generating P-T curves/limits that allows BWR plants to adopt the PTLR option.

Historically, utilities who own BWRs have submitted license amendment requests to update their P-T curves. Lack of an NRC-approved methodology introduces regulatory uncertainty into the license amendment process. This uncertainty has created hardships on licensees in the past when updates to the limits were needed by specific dates. In addition, the current situation causes both the regulator and licensees to expend resources that could otherwise be devoted to other activities. The objective of this report is to avoid these situations by providing P-T curve methods that are generically approved by the NRC so that P-T curves can be documented in a PTLR.

Because many BWR utilities have used Structural Integrity Associates, Inc. (SI) to develop their P-T curves, this report documents the SI P-T curve fracture mechanics methodology in an LTR for NRC review and acceptance. This LTR documents the SI fracture mechanics methods and allows for a "plug and play" approach to reactor pressure vessel (RPV) P-T curve development and approval. Once accepted by the NRC, this LTR can be referenced by any BWR licensee who desires to use the SI methodology for their P-T curve development in a license amendment request to adopt GL 96-03 requirements for a PTLR.

It is noted that this report does not include development or licensing of vessel fluence methods, which are already covered by other LTRs. It is assumed that such fluence methods would be utilized to develop the necessary and appropriate inputs for use in the P-T curve development methodology outlined in this report.

1.2 Purpose of Topical Report

In order to implement the PTLR, the analytical methods used to develop the P-T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the plant Technical Specifications. The purpose of this report is to provide the current SI methodology for developing reactor coolant system (RCS) pressure test, core not critical, and core critical P-T curves for BWRs. When approved by the NRC, this methodology may be referenced by licensees to implement the PTLR.

This LTR does not provide all of the methodologies which can be used to develop RCS pressure test, core not critical, and core critical P-T curves, but rather methodologies that can be referenced by licensees when approved by the NRC to license the PTLR concept. It is also noted that the contents of this report are only intended to license how P-T curves are generated and not how the curves are applied in plant operation.

1.3 Content of Topical Report

This report contains the methodology used to develop the RCS pressure test, core not critical, and core critical P-T curves in Section 2.0. A simplified, step-by-step procedure for implementing the methodology is outlined in Section 3.0. Appropriate references are provided in Section 4.0. Appendix A provides guidance for application of the BWR Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) data to P-T curve development. A sample PTLR is provided in Appendix B, which has been approved for inclusion by PPL for their Susquehanna Steam Electric Station. The sample PTLR in Appendix B is intended to be a template for licensees to follow for development of their own plant-specific PTLRs.

2.0 PRESSURE-TEMPERATURE LIMIT CURVES

2.1 Introduction

Pressure test, core not critical, and core critical P-T limit curves are calculated using the most limiting value of RT_{NDT} (reference nil-ductility transition temperature) corresponding to the limiting material in the RPV, including the effects of neutron irradiation. Normally, the limiting RPV material is located in the RPV beltline region (the region adjacent to the core that is most affected by fast neutron irradiation). The most limiting RT_{NDT} of the material in the beltline region of the RPV is determined by using the unirradiated RPV material fracture toughness properties and estimating the irradiation-induced shift (ΔRT_{NDT}). The unirradiated RT_{NDT} is defined 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 mils of lateral expansion (both normal to the major working direction) minus 60°F.

The RT_{NDT} increases as the material is exposed to fast-neutron irradiation. 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 original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in RPV steels. The NRC has published a method for predicting radiation embrittlement in Regulatory Guide (RG) 1.99, Revision 2 [2]. RG 1.99, Revision 2 is used for the calculation of adjusted reference temperature (ART) values (irradiated RT_{NDT} with margins for uncertainties) at 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[1]. Using the ART values, P-T limit curves are determined in accordance with the requirements of Title 10, Part 50 of the U. S. Code of Federal Regulations (10CFR50) Appendix G [4], as augmented by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Nonmandatory Appendix G [5]. The procedure for establishing P-T limits is entirely deterministic. The conservatisms included in the limits are (but not limited to):

[1] The thickness of the cladding is neglected as specified in the ASME Code, Section III, Paragraph NB-3122.3 [3].

- An assumed flaw in the wall of the RPV that has a depth equal to 1/4 of the thickness of the RPV wall and a length equal to 1.5 times the vessel wall thickness (6-to-1 length-to-depth aspect ratio, a/ℓ).
- A safety factor of 1.5 (for pressure test conditions) or 2.0 (for core not critical and core critical conditions) is applied to the primary membrane stress intensity factor (K_{Im}) and the primary bending stress intensity factor (K_{Ib}).
- Two standard deviation (2σ) margins are applied in determining the ART.
- The limiting toughness is based upon a reference value (K_{Ia}, which is a lower bound of the dynamic crack initiation or arrest toughnesses, and/or K_{Ic}, which is a lower bound of static fracture toughness).

This section describes the methodology used by SI to develop allowable P-T relationships for pressure test, core not critical, and core critical conditions that are included in the PTLR. Separate subsections describing fracture toughness properties, ART calculation, criteria for allowable P-T relationships, and P-T curve generation are provided.

2.2 Fracture Toughness Properties

The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the requirements of 10 CFR Part 50 Appendix G [4], as augmented by the additional requirements in Subsection NB-2331 of Section III of the ASME Code [3]. These fracture toughness requirements are also summarized in Branch Technical Position MTEB 5-2 [6] of the NRC Regulatory Standard Review Plan.

These requirements are used to determine the value of the (RT_{NDT}) for unirradiated material (defined as initial RT_{NDT}, or IRT_{NDT}) and to calculate the ART as described in Section 2.3. Two types of tests are required to determine a material's value of IRT_{NDT}: (i) Charpy V-notch impact (C_v) tests, and (ii) drop-weight tests. The procedure for determining RT_{NDT} is as follows:

1. Determine a temperature, T_{NDT}, that is at or above the nil-ductility transition temperature by drop weight tests.

2. At a temperature not greater than T_{NDT} + 60°F, each specimen of the C_v test shall exhibit at least 35 mils of lateral expansion and not less than 50 ft-lb of absorbed energy. When these requirements are met, T_{NDT} is the reference temperature, or RT_{NDT}.

3. If the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens to determine the temperature, T_{Cv}, at which they are met. In this case, the RT_{NDT} is (T_{Cv} - 60°F). Thus, the RT_{NDT} is the higher of T_{NDT} and (T_{Cv} - 60°F).

4. If the C_v test has not been performed at T_{NDT} + 60°F, or when the C_v test at T_{NDT} + 60°F does not exhibit a minimum of 50 ft-lb of absorbed energy and 35 mils of lateral expansion, a temperature representing a minimum of 50 ft-lb of absorbed energy and 35 mils of lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all of the C_v tests performed, as shown in Figure 2-1.

Licensees that do not follow the fracture toughness requirements in Branch Technical Position MTEB 5-2 to determine IRT_{NDT} can use alternative procedures. However, sufficient technical justification and special circumstances per the criteria of 10CFR50.12(a)(2) [7] must be provided for an exemption from the regulations to be granted by the NRC.

2.3 Calculation of Adjusted Reference Temperature

The ART for each material in the beltline region is calculated in accordance with RG 1.99, Revision 2 [2]. The most limiting ART value (i.e., highest value at the 1/4t location) is used in determining the P-T limit curves. ART is calculated by the following equation:

$$ART = IRT_{NDT} + \Delta RT_{NDT} + Margin \qquad (2.3\text{-}1)$$

where: ART = the adjusted reference temperature (°F)

IRT_{NDT}	=	the reference temperature for the unirradiated material (°F)
ΔRT_{NDT}	=	the mean value of the shift in reference temperature (°F)
Margin	=	the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART (°F)

IRT_{NDT} is defined in Paragraph NB-2331 of Section III of the ASME Code [3], and determined as described in Section 2.2. If measured values of IRT_{NDT} are not available for the material in question, generic mean values for each class of material can 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 shift in reference temperature caused by irradiation, and is calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \tag{2.3-2}$$

where:

ΔRT_{NDT}	=	the mean value of the shift in reference temperature (°F)
CF	=	the chemistry factor (°F)
f	=	calculated fluence at depth, x (10^{19} n/cm^2, E > 1 MeV)

The CF is a function of copper (Cu) and nickel (Ni) content, and is given in Table 1 of RG 1.99, Revision 2 for weld metal, and in Table 2 of RG 1.99, Revision 2 for base metal (i.e., Position 1.1 of RG 1.99, Revision 2). In Tables 1 and 2 of RG 1.99, Revision 2, "weight-percent copper" and "weight-percent nickel" are the best-estimate values for the material and linear interpolation is permitted. When two or more credible surveillance data sets (as defined in RG 1.99, Revision 2, Paragraph B.4) become available, they may be used to calculate CF per Position 2.1 of RG 1.99, Revision 2, as follows:

$$CF = \frac{\sum_{i=1}^{n}[A_i \; f_i^{(0.28-0.10\log f_i)}]}{\sum_{i=1}^{n}[f_i^{(0.28-0.10\log f_i)}]^2} \tag{2.3-3}$$

where:	CF	=	the chemistry factor (°F)
	n	=	the number of surveillance data points
	A_i	=	the measured value of ΔRT_{NDT} for each surveillance data point, i (°F)
	f_i	=	the fluence for each surveillance data point, i (10^{19} n/cm^2, E > 1 MeV)

If Position 2.1 of RG 1.99, Revision 2 results in a higher value of ART than Position 1.1 of RG 1.99, Revision 2, the ART calculated per Position 2.1 must be used. However, if Position 2.1 of RG 1.99, Revision 2 results in a lower value of ART than Position 1.1 of RG 1.99, Revision 2, either value of ART may be used.

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t), the following formula is used to attenuate the fast neutron fluence (E > 1 MeV) at the specified depth:

$$f = f_{surface} * e^{(-0.24x)} \tag{2.3-4}$$

where:	f	=	calculated fluence at depth, x (10^{19} n/cm^2, E > 1 MeV)
	$f_{surface}$	=	the value of neutron fluence at the base metal surface of the RPV at the location of the postulated defect (10^{19} n/cm^2, E > 1 MeV)
	x	=	the depth into the vessel wall measured from the vessel clad/base metal interface (inches)

The resultant fluence is then put into Equation (2.3-2) to calculate ΔRT_{NDT} at the specified depth.

When two or more credible surveillance capsules have been removed, the measured increase in reference temperature (ΔRT_{NDT}) must be compared to the predicted increase in RT_{NDT} for each surveillance material. The predicted increase in RT_{NDT} is the mean shift in RT_{NDT} calculated by Equation (2.3-2) plus two standard deviations ($2\sigma_\Delta$) specified in RG 1.99, Revision 2. If the measured value exceeds the predicted value ($\Delta RT_{NDT} + 2\sigma_\Delta$), a supplement to the PTLR must be provided to demonstrate how the results affect the approved methodology.

Margin is the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART for the calculations required by 10CFR50 Appendix G [4]. Margin is calculated by the following equation:

$$\mathrm{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{2.3-5}$$

where:

where:	Margin	=	the temperature value that is included in the ART calculations to obtain conservative, upper-bound values of ART (°F)
	σ_I	=	the standard deviation for IRT_{NDT} (°F)
	σ_Δ	=	the standard deviation for ΔRT_{NDT} (°F)

If IRT_{NDT} is a measured value, σ_I is estimated from the precision of the test method ($\sigma_I = 0$ for a measured IRT_{NDT} of a single material). If IRT_{NDT} is not a measured value, and generic mean values for that class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. Per RG 1.99, Revision 2, σ_Δ is 28°F for welds and 17°F for base metal. When surveillance data is used to calculate ΔRT_{NDT}, σ_Δ values may be reduced by one-half. In all cases, σ_Δ need not exceed half of the mean value of ΔRT_{NDT}.

Consistent with the above methodology, the BWRVIP has established the ISP that allows sharing of surveillance program results among participating BWR plants. Appendix A of this report provides guidance on how to apply the BWRVIP ISP results in the determination of ART.

2.4 Criteria for Allowable Pressure-Temperature Relationships

The ASME Code requirements [5] for calculating the allowable P-T limit curves for pressure test, core not critical, and core critical conditions specify that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during plant heatup or cooldown conditions cannot be greater than the reference stress intensity factor, K_{Ia} or K_{Ic}, the fracture toughness for the metal temperature at that time. Two values of fracture toughness may be used, K_{Ia} or K_{Ic}, depending upon the ASME Code Edition employed in the calculations. K_{Ia} is obtained from the reference fracture toughness curve, defined in editions of ASME Code, Section XI, Nonmandatory Appendix G prior to 1999[2].

The K_{Ia} and K_{Ic} curves are given by the following equations:

$$K_{Ia} = 26.78 + 1.223 * \exp[0.0145 (T - ART + 160)] \qquad (2.4\text{-}1)^3$$

$$K_{Ic} = 33.20 + 20.734 * \exp[0.0200 (T - ART)] \qquad (2.4\text{-}2)$$

where:

	K_{Ia}	=	the reference stress intensity factor for lower bound of dynamic and crack arrest toughness (ksi $\sqrt{\text{inch}}$)
	K_{Ic}	=	the lower bound of static fracture toughness (ksi $\sqrt{\text{inch}}$)
	T	=	the metal temperature at the postulated 1/4t crack tip (°F)
	ART	=	the ART value calculated as shown in Section 2.3 for the limiting material for the RPV region under consideration (°F)

[2] In ASME Code, Section III, Nonmandatory Appendix G, the reference fracture toughness is denoted as K_{IR}, whereas in pre-1999 editions of ASME Code, Section XI Nonmandatory Appendix G, the reference fracture toughness is denoted as K_{Ia}. However, the K_{IR} and K_{Ia} curves are identical and are defined with the identical functional form.

[3] In some past editions of ASME Code, Section XI, Nonmandatory Appendix G, the equation for K_{Ia} yielded a slightly higher (0.8%) value than the value shown by Equation 2.4-1 due to a printing error that specified a constant of 1.233 instead of 1.223. The value of 1.223 is correct and consistent with Welding Research Council Bulletin 175 [8], and NRC Standard Review Plan 5.3.2 [6].

As documented in the Technical Basis Document [9] for ASME Code Case N-640 [10], K_{Ic} is the preferred fracture toughness value for use in P-T curve development since heatup and cooldown are slow processes, so static properties are appropriate. ASME Code Case N-640 was approved in February 1999 (and related Code Case N-641 [11] was approved in January 2000), and formed the basis for the change from K_{Ia} to K_{Ic} in editions of ASME Code, Section XI, Nonmandatory Appendix G starting with the 1999 Addenda. Based on this, all subsequent equations in this report utilize the K_{Ic} fracture toughness value. For P-T curve submittals where reference to K_{Ia} may be necessary, K_{Ia} can be substituted for K_{Ic} in the equations that follow.

Whereas the fracture toughness expressions are based on the metal temperature at the postulated 1/4t flaw tip, the coolant temperature should be used (i.e., the temperature increase between the RPV coolant and the 1/4t crack tip should be neglected). Use of the coolant temperature is conservative for the limiting cooldown condition described below because the metal temperature "lags", or is warmer than, the coolant temperature. Thus, the use of the coolant temperature will yield a lower (more limiting) value of fracture toughness than the crack tip metal temperature. The use of the coolant temperature is considered to be a necessary conservatism in P-T curve development to ensure that all design margins and safety factors are maintained.

The governing equation for generating P-T limit curves is defined in ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$SF * K_{Im} + K_{It} < K_{Ic} \tag{2.4-3}$$

where: K_{Im} = the stress intensity factor caused by membrane (pressure) stress (ksi$\sqrt{\text{inch}}$)

K_{It} = the stress intensity factor caused by thermal gradients through the RPV wall for Level A and Level B service limits (i.e., core not critical Curve B and core critical Curve C) (ksi$\sqrt{\text{inch}}$) *Note: K_{It} is set to zero for hydrostatic and leak test*

		calculations since these tests are performed at or near isothermal conditions (typically 25°F/hr or less).
SF	=	safety factor
	=	2.0 for Level A and Level B service limits (i.e., core not critical Curve B and core critical Curve C)
	=	1.5 for hydrostatic and leak test conditions when the reactor core is not critical (i.e., Curve A)
K_{Ic}	=	the lower bound of static fracture toughness as a function of the coolant temperature, T, and the ART (ksi $\sqrt{\text{inch}}$)

At specific times during the limiting cooldown transient, K_{Ic} is determined by the metal temperature at the tip of the postulated 1/4t flaw (conservatively assumed to be the same as the coolant temperature), the appropriate value for ART at the same location, and the K_{Ic} fracture toughness equation (Equation 2.4-2). The thermal stresses resulting from the temperature gradients through the vessel wall and the corresponding thermal stress intensity factor, K_{It}, for the reference flaw are calculated as described in Section 2.5. From Equation (2.4-3), the limiting pressure stress intensity factor is obtained and, from this, the allowable pressure is calculated as described in Section 2.5.

For the calculation of the allowable pressure versus coolant temperature during core not critical and core critical conditions, the reference 1/4t flaw of ASME Code, Section XI, Nonmandatory Appendix G is assumed to exist at the inside of the RPV wall. P-T curves developed with this flaw assumption are bounding because the controlling location of the flaw is always at the inside of the vessel wall. This is due to two reasons: (1) the thermal gradients that increase with increasing cooldown rates produce tensile stresses at the inside surface that would tend to open (propagate) the existing flaw, and (2) the ART for an inside surface (1/4t) flaw is more limiting than an outside (3/4t) flaw due to fluence attenuation effects through the RPV wall. Therefore, P-T curves developed for an inside surface (1/4t) postulated flaw under cooldown conditions bound the case of an outside surface (3/4t) postulated flaw under heatup conditions. P-T curves developed on this basis are valid for use under both heatup and cooldown conditions.

In addition to the above, it is also recognized that P-T limits generated for the RPV also are considered to cover all portions of the RCS piping. There are at least four reasons why the RPV P-T limits are considered to adequately bound fracture toughness requirements for the RCS piping: (1) the RPV is irradiated (thereby experiencing material degradation due to neutron embrittlement) whereas the RCS piping is not, (2) the philosophy behind the design codes used to evaluate the design of the RPV and piping generally recognize that the RPV is more limiting than the RCS piping from a structural standpoint, (3) much of the RCS piping is austenitic stainless steel, which has ductile behavior and does not experience the fracture concerns that ferritic material experiences, and (4) stresses are typically higher in the thicker-walled RPV than in the thin-walled RCS piping.

Allowable P-T curves are typically generated for a 100°F/hr[4] cooldown rate, which is the limiting cooldown rate typically specified in plant Technical Specifications. However, curves for other cooldown rates can also be generated to provide a basis for acceptability when Technical Specification cooldown rates may be exceeded (i.e., bottom head stratified conditions), or to help support cases where it is desirable to change plant Technical Specification cooldown rate limits.

Finally, the 1983 Amendment to 10CFR50 Appendix G has rules which address the metal temperature of the closure head flange and vessel flange regions. These rules state that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation and 90°F for hydrostatic pressure tests and leak tests when the pressure exceeds 20% of the preservice hydrostatic test pressure. In addition, when the core is critical, the P-T limits for core operation (except for low power physics tests) require that the reactor vessel 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

[4] In the context of P-T curve development, a linear 100°F/hr cooldown rate is typically assumed. Due to other conservatisms inherent in the methodology defined in this report, this rate is considered to cover all cases of "a 100°F change in temperature in any 1-hour period." In other words, as long as the temperature change in any 1-hour period is less than or equal to 100°F, the curves developed using a rate of 100°F/hr remain valid for use.

in the corresponding P-T curve for core not critical conditions. These limits must be incorporated into the P-T limit curves wherever applicable.

A petition to revise the 10CFR50 Appendix G flange requirements was submitted by the Westinghouse Owners Group (WOG) in November 1999 [12], which would eliminate the flange requirement completely. That petition has been suspended due to additional requirements requested by the NRC. Some licensees have since elected to pursue elimination of these requirements on a plant-specific basis. Therefore, until the text of 10CFR50, Appendix G is changed, the flange requirements remain in force, but can potentially be eliminated through a plant-specific exemption request.

Figure 2-2 shows an example of a set of pressure test curves applicable for the first 32 effective full power years (EFPY) of plant operation. Separate curves are defined for the beltline, non-beltline, and bottom head regions. Figure 2-3 shows an example of core not critical P-T curves using a rate of 100°F/hr applicable for 32 EFPY of plant operation. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 2-2 and 2-3. Note that the steps in these curves are due to the previously described flange requirements [4].

2.5 Pressure-Temperature Curve Generation Methodology

2.5.1 Thermal Stress Intensity Factor Calculations for Shell Regions

For shell regions remote from discontinuities, there are several methods available for computing the thermal stress intensity factor, K_{It}, for use in establishing P-T limits. Three methods routinely employed by SI are described in this section: (1) the Closed Form Solution Method, (2) the Section XI Nonmandatory Appendix G Method, and (3) the Welding Research Council (WRC) Bulletin No. 175 [8] Method. Each of these three methods is described next.

Closed Form Solution Method

For this method, the thermal stress intensity factor, K_{It}, may be calculated using a closed form solution using conventional heat transfer and thermal stress methodology. The time-dependent temperature solution utilized in the cooldown analysis may be based on the following one-dimensional transient heat conduction equation [17]:

$$\rho C \frac{\partial T}{\partial t} = K\left[\frac{\partial^2 T}{\partial r^2} + \frac{1}{r}\frac{\partial T}{\partial r}\right] \tag{2.5.1-1}$$

with the following boundary conditions applied at the inner and outer radii of the RPV:

at $r = r_i$: $$-K\frac{\partial T}{\partial r} = h(T - T_c) \tag{2.5.1-2}$$

at $r = r_o$: $$\frac{\partial T}{\partial r} = 0 \tag{2.5.1-3}$$

where:

r_i	=	the reactor vessel inner radius (inches)
r_o	=	the reactor vessel outer radius (inches)
ρ	=	the material density (lb/in^3)
C	=	the material specific heat (BTU/lb-°F)
K	=	the material thermal conductivity (BTU/sec-in-°F)
T	=	the local metal temperature (°F)
r	=	the radial location (inches)
t	=	time (sec)
h	=	the heat transfer coefficient between the coolant and the vessel wall (BTU/sec-in^2-°F)
T_c	=	the coolant temperature (°F)

These equations are solved numerically to generate the position and time-dependent temperature distributions, T(r,t), for all cooldown rates of interest.

With the results of the heat transfer analysis as input, position and time-dependent distributions of thermal hoop stress are calculated using the formula for the thermal stress in a hollow cylinder given by Timoshenko [13].

$$\sigma_\theta(r,t) = \frac{\alpha E}{1-\nu}\frac{1}{r^2}\left[\frac{r^2+r_i^2}{r_o^2-r_i^2}\int_{r_i}^{r_o} T(r,t)\,r\,dr + \int_{r_i}^{r} T(r,t)\,r\,dr - T(r,t)\,r^2\right] \qquad (2.5.1\text{-}4)$$

where:	$\sigma_\theta(r,t)$	=	the hoop stress at location, r, and time, t (psi)
	E	=	the modulus of elasticity (psi)
	α	=	the mean coefficient of linear expansion (in/in-°F)
	ν	=	Poisson's ratio = 0.3

The quantities E and α are temperature-dependent properties. However, to simplify the analysis, E and α may be evaluated at an equivalent wall temperature at a given time:

$$T_{eqv} = \frac{2\int_{r_i}^{r_o} T(r)\,r\,dr}{r_o^2 - r_i^2} \qquad (2.5.1\text{-}5)$$

E and α are calculated as a function of this equivalent temperature and the Eα product in Equation (2.5.1-4) is treated as a constant in the computation of thermal hoop stress.

The secondary linear bending (σ_{sb}) and constant secondary membrane (σ_{sm}) stress components of the thermal hoop stress profile are approximated by the linearization technique presented in ASME Code, Section XI, Nonmandatory Appendix A [14]. These stress components are used for determining the thermal stress intensity factor, K_{It}, based on Paragraph G-2200 of ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$K_{It} = M_m\,\sigma_{sm} + M_b\,\sigma_{sb} = M_m\,(\sigma_{sm} + 0.667\,\sigma_{sb}) \qquad (2.5.1\text{-}6)$$

where:	K_{It}	=	the thermal stress intensity factor (ksi $\sqrt{\text{inch}}$)
	σ_{sm}	=	the secondary membrane hoop stress (psi)
	σ_{sb}	=	the secondary bending hoop stress (psi)
	M_m	=	the membrane correction factor for an inside axial surface flaw, per Paragraph G-2214.1, as follows

1.85 for $\sqrt{t} < 2$

$0.926\ t$ for $2 \le \sqrt{t} \le 3.464$

3.21 for $\sqrt{t} > 3.464$

In the above expression, the bending correction factor, M_b, has been set equal to $(2/3)M_m$ in accordance with Paragraph G-2214.2 of ASME Code, Section XI, Nonmandatory Appendix G [5].

Section XI Nonmandatory Appendix G Method

For this method, the thermal stress intensity factor, K_{It}, may be calculated using the stress intensity factor expression from ASME Code, Section XI, Nonmandatory Appendix G [5].

The maximum K_{It} produced by a radial thermal gradient for a postulated inside surface defect is:

$$K_{It} = 0.953\text{x}10^{-3}\ (CR)\ (t^{2.5}) \qquad (2.5.1\text{-}7)$$

where:	CR	=	the cooldown rate (°F/hr)
	t	=	the RPV wall thickness (inches)
	K_{It}	=	the thermal stress intensity factor (ksi $\sqrt{\text{inch}}$)

The through-wall temperature difference associated with the maximum thermal stress intensity factor, K_{It}, is determined from Figure G-2214-1 of ASME Code, Section XI, Nonmandatory Appendix G. The temperature at any radial distance from the vessel surface can be determined from Figure G-2214-2 of ASME Code, Section XI, Nonmandatory Appendix G for the maximum thermal stress intensity factor, K_{It}, with the following restrictions:

(a) The maximum thermal stress intensity factor, K_{It}, relationship and the temperature relationship in Figure G-2214-1 are applicable only for the conditions given in Paragraph G-2214.3(a)(1) and (2) of ASME Code, Section XI, Nonmandatory Appendix G.

(b) Alternatively, the K_{It} for a radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during the cooldown for a 1/4t inside surface defect using the following relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)\sqrt{\pi a} \qquad (2.5.1\text{-}8)$$

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 cooldown using the following form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.5.1\text{-}9)$$

where: x = the radial distance from the inside surface to any point on the crack front (inches)

a = the maximum crack depth (inches)

Welding Research Council Bulletin No. 175 Method

For this method, the thermal stress intensity factor, K_{It}, may be calculated using the stress intensity factor expression from WRC Bulletin 175 [8]:

$$K_{It} = \left[\sigma_{sm} 1.1M_K + \sigma_{sb} M_B\right]\sqrt{\frac{\pi a}{Q}} \qquad (2.5.1\text{-}10)$$

where: K_{It} = the thermal stress intensity factor (ksi $\sqrt{\text{inch}}$)

σ_{sm}	=	the constant secondary membrane stress component from the linearized thermal hoop stress distribution (psi)
σ_{sb}	=	the linear secondary bending stress component from the linearized thermal hoop stress distribution (psi)
M_K	=	the correction factor for membrane stress as a function of relative flaw depth, a/t (see Figure 2-4)
M_B	=	the correction factor for bending stress as a function of relative flaw depth, a/t (see Figure 2-5)
a	=	crack depth (inches)
Q	=	the flaw shape factor modified for plastic zone size, interpolated from the following:

σ/σ_y	0.1	0.3	0.5	0.7	1.0
Q	1.235	1.215	1.190	1.135	1.030

σ	=	the total thermal stress (psi) $=\sigma_{sm} + \sigma_{sb}$
σ_y	=	the material yield stress (psi)

2.5.2 Allowable Pressure Stress Intensity Factor Calculations for Shell Regions

The minimum allowable pressure is calculated as a function of coolant temperature using the allowable fracture toughness, K_{Ic}, the applied thermal stress intensity factor, K_{It}, and the required safety factor.

For shell regions remote from discontinuities, since BWR RPVs are classified as thin-walled cylindrical pressure vessels (R/t > 10), the stress due to applied pressure may be considered as entirely membrane in nature. Thus, for membrane tension, the membrane tension stress intensity factor, K_{Im}, for a postulated 1/4t defect is defined in ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$K_{Im} = (K_{Ic} - K_{It}) / SF \qquad (2.5.1\text{-}11)$$

where: K_{Im} = the allowable stress intensity factor caused by membrane (pressure) stress (ksi$\sqrt{\text{inch}}$)

K_{Ic} = the lower bound of static fracture toughness as a function of the coolant temperature, T, and the limiting ART for all beltline weld and plate materials from Equation 2.4-2 (ksi$\sqrt{\text{inch}}$)

K_{It} = the thermal stress intensity factor (ksi$\sqrt{\text{inch}}$)

Note that the thermal stress intensity factor is neglected (i.e., K_{It} = 0) for developing the inservice hydrostatic and leak test P-T curve since the hydrostatic leak test is performed at or near isothermal conditions (typically 25°F/hr or less).

SF = safety factor

= 2.0 for Level A and Level B service limits (i.e., for core not critical Curve B and core critical Curve C)

= 1.5 for hydrostatic and leak test conditions when the reactor core is not critical (i.e., for Curve A)

The allowable pressure for a 1/4t postulated limiting (axial) defect is defined based on membrane pressure stress in ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$P_{allow} = (K_{Im}\, t) / (M_m\, R_i) \qquad (2.5.1\text{-}12)$$

where: P_{allow} = the allowable internal pressure (psi)

K_{Im} = the allowable stress intensity factor caused by the membrane (pressure) stress (ksi$\sqrt{\text{inch}}$)

t = the RPV wall thickness (inches)

M_m = the membrane correction factor for an inside axial surface flaw:

1.85 for $\sqrt{t} < 2$

$0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

3.21 for $\sqrt{t} > 3.464$

R_i = the vessel inner radius (inches)

Note in the above expression, since the pressure stress is treated entirely as a membrane stress, the stress intensity factor due to primary (pressure) bending has been neglected.

2.5.3 Thermal and Pressure Stress Intensity Factor Calculations for Discontinuity Regions

In more recent years, in addition to P-T limits established for the RPV beltline shell region, separate P-T limits have typically also been developed for two discontinuity regions: (1) the RPV bottom head, and (2) the RPV non-beltline region, which is typically controlled by the feedwater nozzle and flange regions. Separate P-T limit curves for the beltline, non-beltline, and bottom head regions allows greater operational flexibility during transient conditions where temperatures experienced in these other regions can be significantly different than in the beltline region. For these discontinuity regions, the same general procedures as those described in Sections 2.5.1 and 2.5.2 for shell regions apply, except that certain modifications are made to develop appropriate stresses for determining the thermal stress intensity factor, K_{It}, and the pressure stress intensity factor, K_{Ip}, under the presence of discontinuity stresses. Methods for calculating thermal and pressure stress intensity factors for each of these typical discontinuity regions is described in this section. For cases where there is a desire to establish P-T limits for discontinuity regions other than those described herein, the same general methods as those described below may be applied.

<u>*Bottom Head Region*</u>

For the thermal stress intensity factor, K_{It}, the methodology described in Section 2.5.1 may be used for the bottom head region. Although the methodology described in Section 2.5.1 is based on one-dimensional heat transfer and stress solutions for a cylindrical structure, the solution closely approximates the thermal stress solutions for a sphere or a flat plate due to the large diameter of the BWR RPV (on the order of 200 inches). Therefore, the K_{It} solution contained in Section 2.5.1 is deemed appropriate for use in the bottom head region for normal heatup and cooldown transients. Available thermal stresses from existing stress reports may also be used, as well as other solution techniques (such as finite element analysis) to develop the bottom head region thermal stresses.

The minimum allowable pressure is different for the bottom head region compared to the beltline shell region due to the spherical bottom head configuration, as well as the presence of bottom head penetrations. Therefore, methodology is provided below for the calculation of allowable pressure for the bottom head region that properly accounts for these differences.

For the bottom head region, the stress due to applied pressure is considered as entirely membrane in nature, with a conservative stress concentration factor applied to account for the bottom head penetrations. Thus, the membrane tension stress intensity factor, K_{Im}, for a postulated 1/4t defect is defined in ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$K_{Im} = (K_{Ic} - K_{It}) / SF \qquad (2.5.1\text{-}13)$$

where:

K_{Im} = the allowable stress intensity factor caused by the membrane (pressure) stress (ksi$\sqrt{\text{inch}}$)

K_{Ic} = the lower bound of static fracture toughness as a function of the coolant temperature, T, and the limiting RT_{NDT} for all bottom head plate and weld materials from Equation 2.4-2 (ksi$\sqrt{\text{inch}}$)

K_{It} = the thermal stress intensity factor (ksi$\sqrt{\text{inch}}$)

Note that the thermal stress intensity factor is neglected (i.e., $K_{It} = 0$) for developing the inservice hydrostatic and leak test P-T curve since the hydrostatic leak test is performed at or near isothermal conditions (typically 25°F/hr or less).

SF = safety factor

= 2.0 for Level A and Level B service limits (i.e., for core not critical Curve B and core critical Curve C)

= 1.5 for hydrostatic and leak test conditions when the reactor core is not critical (i.e., for Curve A)

The allowable pressure for a 1/4t postulated limiting (axial) defect is defined based on spherical membrane pressure stress as follows:

$$P_{allow} = (2K_{Im}\, t) / (SCF\, M_m\, R_i) \qquad (2.5.1\text{-}14)$$

where:

P_{allow} = the allowable internal pressure (psi)

K_{Im} = the allowable stress intensity factor caused by membrane (pressure) stress (ksi $\sqrt{\text{inch}}$)

t = the bottom head wall thickness (inches)

SCF = conservative stress concentration factor to account for bottom head penetration discontinuities = 3.0

M_m = the membrane correction factor for an inside axial surface flaw:

1.85 for $\sqrt{t} < 2$

$0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$

3.21 for $\sqrt{t} > 3.464$

R_i = the bottom head inner radius (inches)

Note in the above expression, since the pressure stress is treated entirely as a membrane stress, the stress intensity factor due to primary (pressure) bending has been neglected.

Non-Beltline Region

P-T limits for the non-beltline region are intended to encompass and bound all locations outside of the beltline region (excluding the bottom head, if it is evaluated separately). The non-beltline regions are defined as all RPV locations with fluence values less than $1\text{x}10^{17}$ n/cm^2 (E > 1 MeV). Typically, the limiting location outside of the beltline region is the feedwater nozzle, where stresses are highest due to the most severe thermal transients. However, determination of the limiting location must also consider the material RT_{NDT}. In many cases, a worst-case assumption of feedwater nozzle stresses and the highest RT_{NDT} of all locations outside of the beltline region (excluding the bottom head region, if it is evaluated separately) is used. In addition, the flange requirements discussed in Sections 2.7 and 2.8 are also applied to the non-beltline region P-T limits. Based on this reasoning, the discussion that follows is based on stresses determined for the feedwater nozzle.

The stress intensity factors for the feedwater nozzle may be calculated using the results of a detailed finite element model of the nozzle. In many cases, such results are already available from the governing design basis stress report for the feedwater nozzle. The details of the finite element process are not included here; rather, the extraction of the appropriate finite element results and their use in developing P-T limit curves is discussed.

For a path through the limiting nozzle inner blend radius corner, the following stresses should be extracted from the finite element model using the linearization technique presented in ASME Code, Section XI, Nonmandatory Appendix A [14] at a depth of 1/4t for the section selected:

- Primary membrane hoop stress due to operating pressure, σ_{pm}
- Primary bending hoop stress due to operating pressure, σ_{pb}
- Secondary membrane hoop stress due to limiting normal/upset transient, σ_{sm}
- Secondary bending hoop stress due to limiting normal/upset transient, σ_{sb} .

The thermal stress intensity factor, K_{It}, is computed based on Paragraph G-2220 of ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$K_{It} = R\,M_m\,\sigma_{sm} + R\,M_b\,\sigma_{sb} = R\,M_m\,(\sigma_{sm} + 0.667\,\sigma_{sb}) \qquad (2.5.1\text{-}15)$$

where: K_{It} = the thermal stress intensity factor (ksi $\sqrt{\text{inch}}$)

R = correction factor, calculated to consider the nonlinear effects in the plastic region based on the assumptions and recommendation of WRC Bulletin 175 [8]:

$$R = \frac{\left[\sigma_y - \sigma_{pm} + \frac{(\sigma_{total} - \sigma_y)}{30}\right]}{(\sigma_{total} - \sigma_{pm})}$$

σ_y	=	the material yield stress (psi)
σ_{pm}	=	the primary membrane hoop stress due to operating pressure (psi)
σ_{pb}	=	the primary bending hoop stress due to operating pressure (psi)
σ_{total}	=	the total stress (psi) = $\sigma_{pm} + \sigma_{pb} + \sigma_{sm} + \sigma_{sb}$
σ_{sm}	=	the secondary membrane hoop stress due to the limiting normal/upset transient (psi)
σ_{sb}	=	the secondary bending hoop stress due to the limiting normal/upset transient (psi)
M_m	=	the membrane correction factor for an inside axial surface flaw:

$$1.85 \text{ for } \sqrt{t} < 2$$

$$0.926\sqrt{t} \text{ for } 2 \le \sqrt{t} \le 3.464$$

$$3.21 \text{ for } \sqrt{t} > 3.464$$

In the above expression, the bending correction factor, M_b, has been set equal to $(2/3)M_m$ in accordance with Paragraph G-2214.2 of ASME Code, Section XI, Nonmandatory Appendix G [5].

The allowable pressure stress intensity factor, K_{Ip}, for a postulated 1/4t defect is defined in ASME Code, Section XI, Nonmandatory Appendix G [5] as follows:

$$K_{Ip} = (K_{Ic} - K_{It}) / SF \qquad (2.5.1\text{-}16)$$

where: K_{Ip} = the allowable stress intensity factor caused by pressure stress ($ksi\sqrt{inch}$)

K_{Ic} = the lower bound of static fracture toughness as a function of the coolant temperature, T, and the limiting RT_{NDT} for all non-beltline locations (excluding the bottom head region, if it is addressed separately) from Equation 2.4-2 (ksi $\sqrt{\text{inch}}$)

K_{It} = the thermal stress intensity factor (ksi $\sqrt{\text{inch}}$)

Note that the thermal stress intensity factor is neglected (i.e., K_{It} = 0) for developing the inservice hydrostatic and leak test P-T curve since the hydrostatic leak test is performed at or near isothermal conditions (typically 25°F/hr or less).

SF = safety factor

= 2.0 for Level A and Level B service limits (i.e., for core not critical Curve B and core critical Curve C)

= 1.5 for hydrostatic and leak test conditions when the reactor core is not critical (i.e., for Curve A)

The applied pressure stress intensity factor, $K_{Ip\text{-applied}}$, is computed as follows:

$$K_{Ip-applied} = F(a, r_n)\,\sigma_{pm}\sqrt{\pi a} \qquad (2.5.1\text{-}17)$$

where:

$K_{Ip\text{-applied}}$ = the applied pressure stress intensity factor (ksi $\sqrt{\text{inch}}$)

$F(a,r_n)$ = nozzle stress factor for "Proposed Nozzle Corner Flaws" line from Figure A5-1 of WRC Bulletin 175 (see Figure 2-6)

r_n = the apparent radius of nozzle (inches)

= $r_i + 0.29r_c$

r_i = the inner radius of the nozzle (inches)

r_c = the nozzle inner corner radius (inches)

σ_{pm} = the primary membrane hoop stress due to operating pressure, P (psi)

a = the 1/4t postulated crack depth (inches)

t = the nozzle corner section thickness (inches)

The allowable pressure, P_{allow}, for a 1/4t postulated limiting (axial) defect is defined as follows:

$$P_{allow} = (K_{Ip}\,P) / K_{Ip\text{-}applied} \qquad (2.5.1\text{-}18)$$

where:

	P_{allow}	=	the allowable internal pressure (psi)
	K_{Ip}	=	the allowable pressure stress intensity factor (ksi $\sqrt{\text{inch}}$)
	P	=	the operating pressure (psi)
	$K_{Ip\text{-}applied}$	=	the applied pressure stress intensity factor (ksi $\sqrt{\text{inch}}$)

2.6 Final P-T Limits and Instrument Uncertainties

Once the allowable pressure versus coolant temperature relationship has been calculated in accordance with one of the methods described in Section 2.5, final P-T limits are calculated as follows:

$$T_{P\text{-}T} = T + U_T \qquad (2.6\text{-}1)$$

$$P_{P\text{-}T} = P - P_H - U_P \qquad (2.6\text{-}2)$$

where :

	$T_{P\text{-}T}$	=	the allowable coolant (metal) temperature (°F)
	U_T	=	the temperature instrument uncertainty (°F)
	$P_{P\text{-}T}$	=	the allowable reactor pressure (psig)
	P_H	=	the pressure head to account for the column of water in the RPV (psig) = $\rho\Delta h$
	ρ	=	water weight density at ambient temperature (lb/in^3)
	Δh	=	elevation between the reactor pressure instrument and the elevation of the RPV bottom head inside surface (inches)
	U_P	=	the pressure instrument uncertainty (psig)

Temperature and pressure instrument uncertainties shall be determined using appropriate techniques and good engineering practice. The signs applied to the uncertainties in the above expressions cause the most conservative shift in P-T limits (i.e., downward and to the right).

2.7 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G [4] contains the requirements for the minimum metal temperature of the closure head flange and vessel flange regions. These requirements state that the metal temperature of the closure flange regions must meet the following requirements:

Curve A (Hydrostatic Pressure and Leak Tests)

The following additional minimum temperature requirements apply to the non-beltline P-T limits for Curve A (core is not critical and with fuel in the vessel), per Table 1 of 10CFR50, Appendix G [4]:

- If the pressure is greater than 20% of the pre-service hydro test pressure[5], the temperature must be greater than the RT_{NDT} of the limiting flange material + 90°F.
- If the pressure is less than or equal to 20% of the pre-service hydro test pressure, the minimum temperature must be greater than or equal to the RT_{NDT} of the limiting flange material.

The above requirements cause a temperature shift in Curve A at 20% of the pre-service system hydrotest pressure.

Curve B (Core Not Critical)

The following additional minimum temperature requirements apply to the non-beltline P-T limits for Curve B, per Table 1 of 10CFR50, Appendix G [4]:

- If the pressure is greater than 20% of the pre-service hydro test pressure, the temperature must be greater than the RT_{NDT} of the limiting flange material + 120°F.
- If the pressure is less than or equal to 20% of the pre-service hydro test pressure, the minimum temperature must be greater than or equal to the RT_{NDT} of the limiting flange material.

The above requirements cause a temperature shift in Curve B at 20% of the pre-service system hydrotest pressure.

[5] Typically, the pre-service system hydrotest pressure is 1,563 psig, which corresponds to 1.25 times the typical GE BWR design pressure of 1,250 psig. Thus, 20% of the pre-service system hydrotest pressure corresponds to 312 psig.

Curve C (Core Critical)

Curve C is generated from the requirements of 10CFR50, Appendix G [4]. The following additional minimum temperature requirements apply to the non-beltline P-T limits for Curve C, per Table 1 of 10CFR50, Appendix G:

- The Curve C P-T limits shall be 40°F above any Curve A or B limits. Curve B is more limiting than Curve A (due to a higher safety factor and the presence of thermal stresses), so Curve C values are at least Curve B plus 40°F.
- For a BWR with water level within the normal range for power operation, the allowed temperature for initial criticality at the closure flange region is (RT_{NDT} + 60°F) at pressures below 20% of the pre-service system hydrotest pressure. In addition, above 20% of the pre-service system hydrotest pressure, the Curve C temperature must be at least the greater of the RT_{NDT} of the limiting closure region + 160°F, or the temperature required for the hydrostatic pressure test (Curve A at the test pressure).

The above requirements cause a temperature shift in Curve C at 20% of the pre-service system hydrotest pressure.

The above flange requirements were 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% of the steady-state stress, without being at steady-state temperature. The temperature margins and the pressure limitation of 20% of pre-service 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 a more recent change to allow the use of K_{Ic} in the development of P-T curves, as discussed previously (i.e., ASME Code Cases N-640 and N-641).

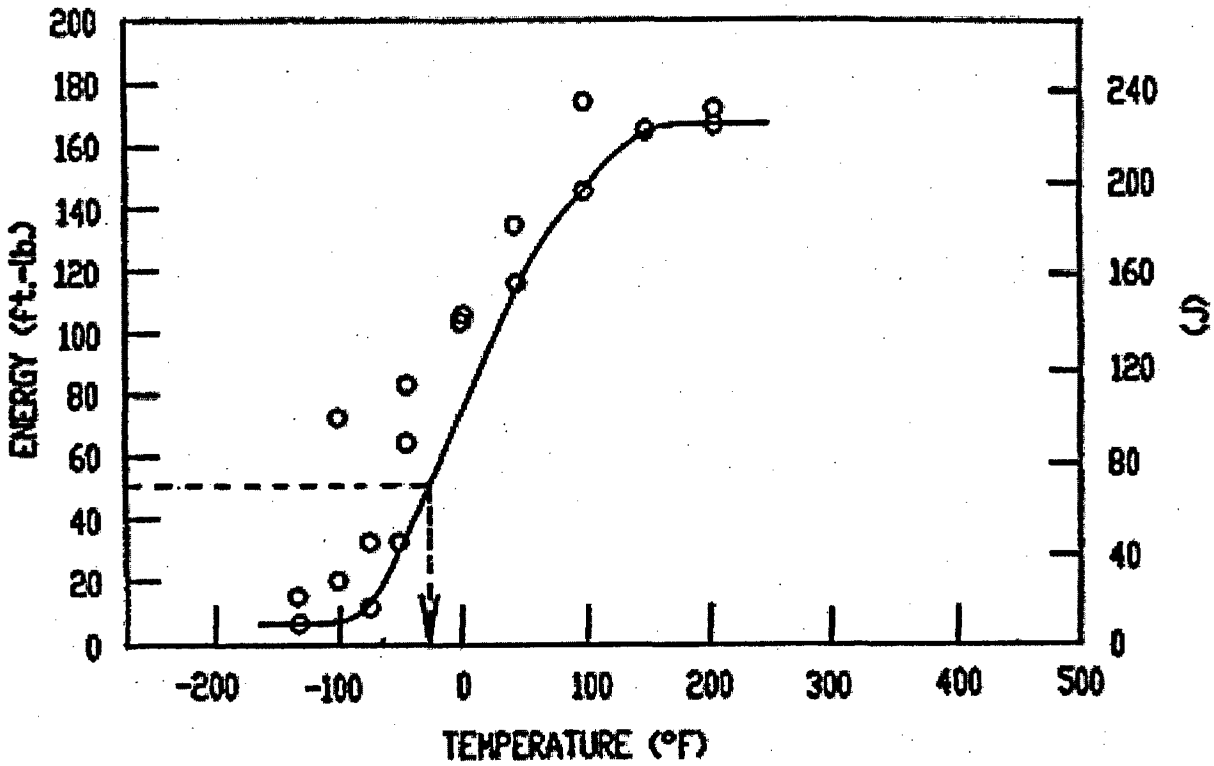
As discussed in Section 2.4, a petition was made by the WOG in November 1999 [12] to eliminate the flange requirements contained in 10CFR50, Appendix G. From that petition, the discussion given in WCAP-15315 [15] concludes that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using the K_{Ic} fracture toughness. In addition, there are no known mechanisms of degradation for this region, other than fatigue. However, the calculated design fatigue usage for this region is typically high for most BWR plants (i.e., greater than 0.8) due to the high bolt preload stresses, so it cannot be necessarily concluded that fatigue flaws are unlikely to initiate in this region. Therefore, the boltup requirements contained in 10CFR50, Appendix G should be used until a revision to 10CFR50, Appendix G is made, or unless a plant-specific exemption is performed to demonstrate that the flange requirements can be eliminated from the P-T curves.

2.8 Minimum Boltup Temperature

For conditions where the core is not critical, the minimum boltup temperature is equal to the material RT_{NDT} of the limiting region affected by boltup stresses per Table 1 of 10CFR50, Appendix G [4]. The RT_{NDT} is calculated in accordance with the methods described in Branch Technical Position MTEB 5-2 [6]. Consistent with the Westinghouse position [16], the minimum boltup temperature shall not be lower than 60°F. Thus, the minimum boltup temperature should be 60°F or the material RT_{NDT}, whichever is higher.

As discussed in Section 2.7, for conditions where the core is critical, the minimum boltup temperature is equal to the material RT_{NDT} of the limiting region affected by boltup stresses + 60°F.

Figure 2-1. Example of a Charpy Impact Energy Curve Used to Determine IRT_{NDT}



(Note: A lateral expansion of 35 mils is required at the indicated temperature.)

Figure 2-2. Sample Pressure Test P-T Limit Curves for 32 EFPY

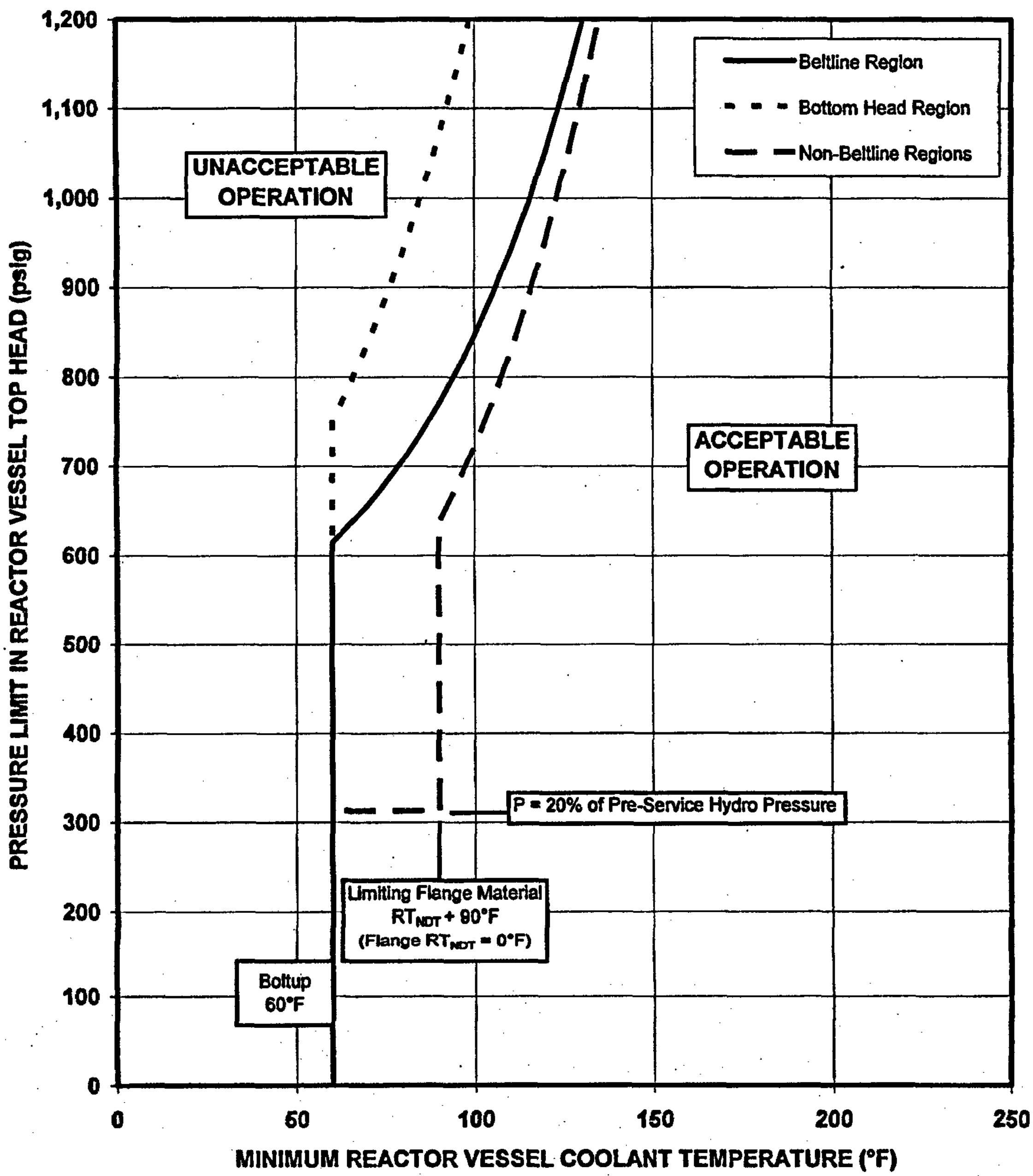


Figure 2-3. Sample Core Not Critical P-T Limit Curves for 32 EFPY

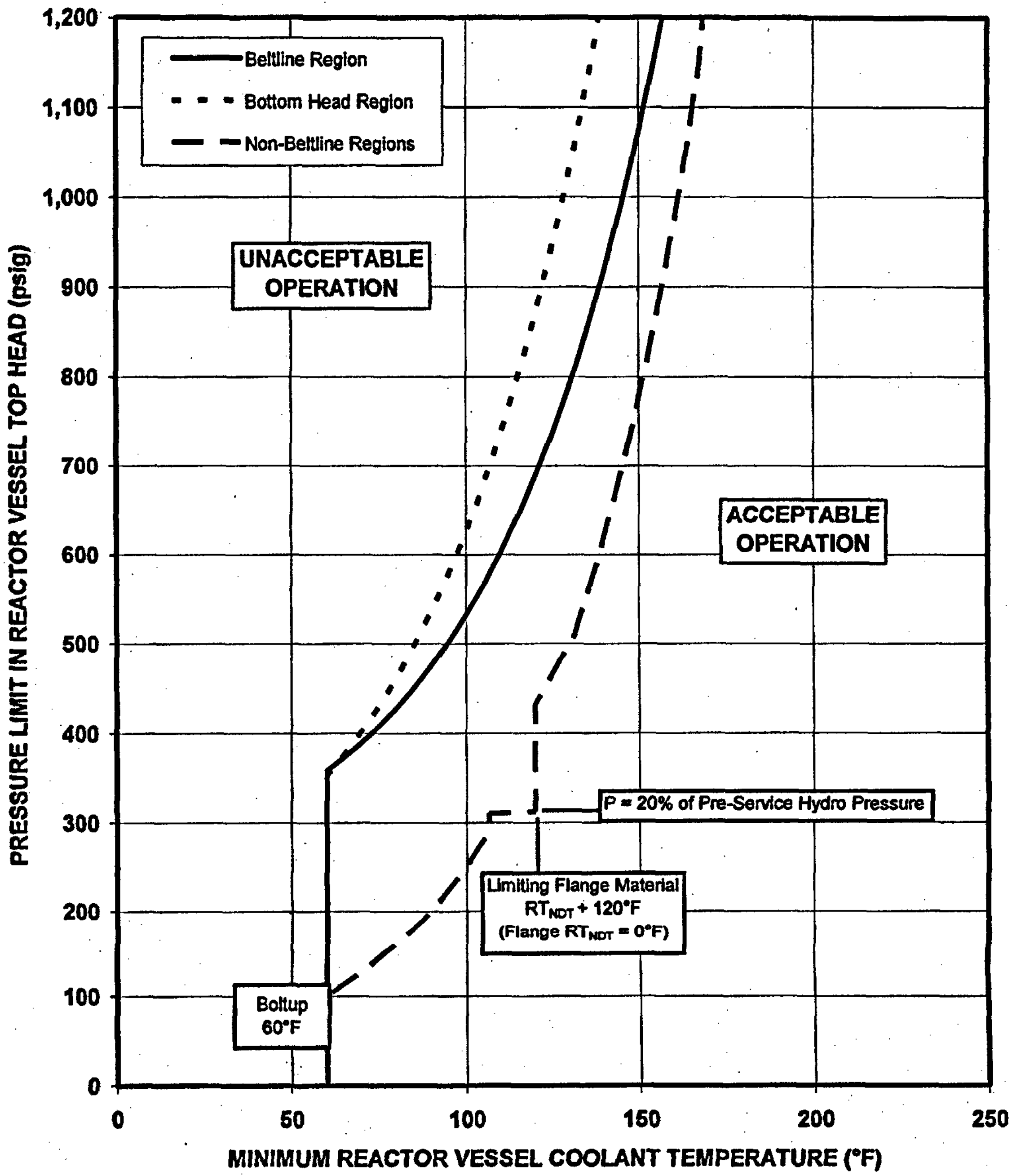


Figure 2-4. Membrane Stress Correction Factor (M_K) (WRC Bulletin No. 175 Method)

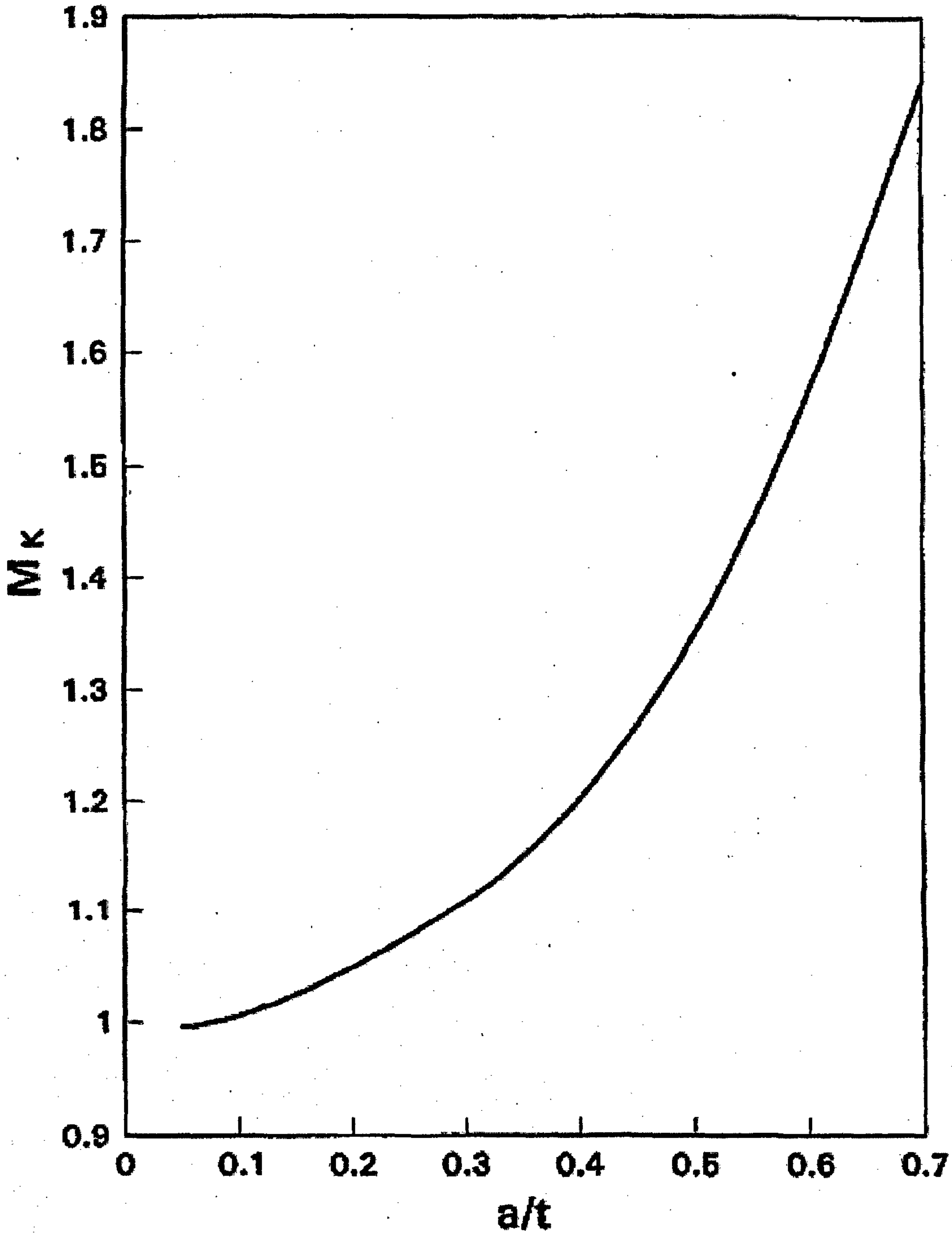


Figure 2-5. Bending Stress Correction Factor (M_B) (WRC Bulletin No. 175 Method)

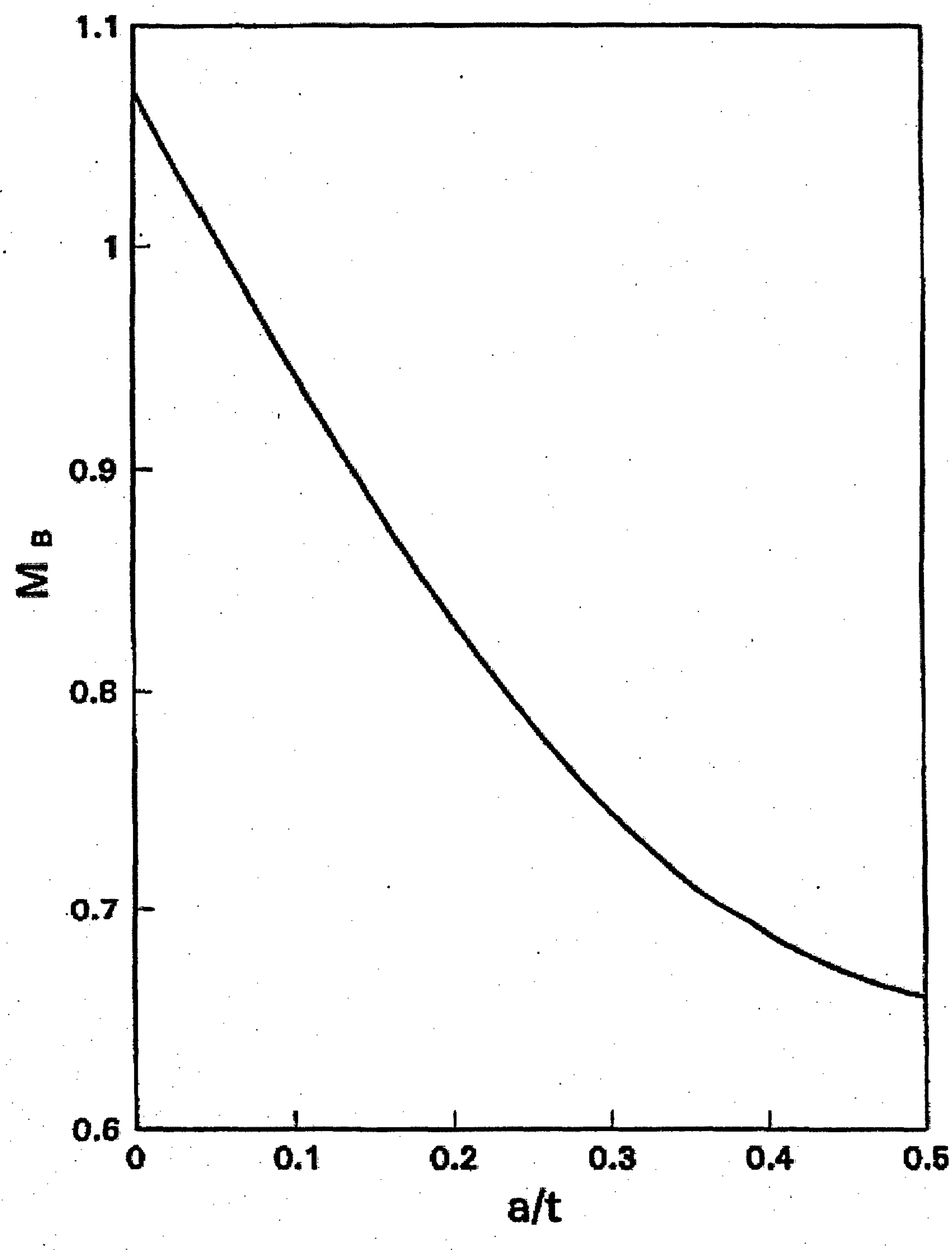
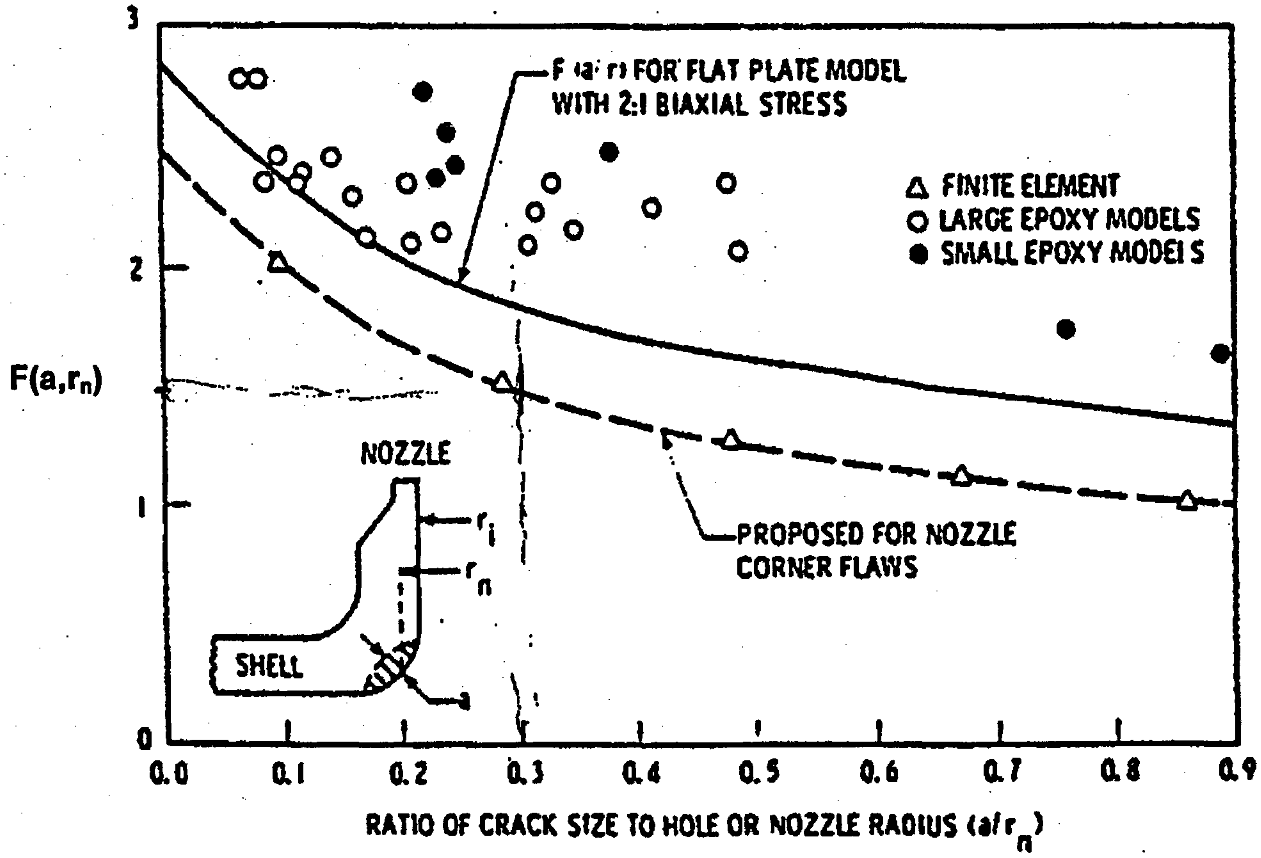


Figure 2-6. Nozzle Stress Intensity Factors (Figure A5-1 of WRC Bulletin No. 175)



3.0 STEP-BY-STEP PROCEDURE FOR CALCULATING P-T LIMIT CURVES

A step-by-step procedure for developing P-T limits using the methodology described in Section 2.5 is provided in this section.

There are typically three RPV regions that are evaluated with respect to P-T limits: (1) the beltline region, (2) the bottom head region, and (3) the non-beltline region including the flanges. Most typically, the non-beltline region is controlled by the feedwater nozzle, where thermal stresses are highest. The non-beltline region should account for the worst RT_{NDT} of all RPV materials outside of the beltline region, as well as minimum flange temperature requirements (see Sections 2.7 and 2.8). P-T limit curves may also be developed for other RPV regions to provide additional operating flexibility.

The approach used for calculating the pressure test (Curve A), core not critical (Curve B), and core critical (Curve C) P-T limit curves for each of these regions is summarized as follows:

a. Assume a coolant temperature, T. The temperature drop from the fluid to the metal temperature at the assumed flaw tip (i.e., T at 1/4t) is conservatively assumed to be zero and metal temperature is assumed equivalent to coolant temperature.

b. Calculate the allowable stress intensity factor, K_{Ic}, using Equation 2.4-2 for the assumed fluid temperature, T, and the limiting ART for the region being evaluated.

c. Calculate the thermal stress intensity factor, K_{It}, using one of the methods described in Sections 2.5.1 or 2.5.3.

d. Calculate the allowable pressure stress intensity factor, K_{Im} or K_{Ip}, using the methods described in Sections 2.5.2 or 2.5.3.

e. Calculate the allowable pressure, P_{allow}, using the methods described in Sections 2.5.2 or 2.5.3.

f. Repeat steps (a) through (f) for other temperatures to generate a series of P-T points. The resulting pressure and temperature series constitutes the P-T curve. The P-T curve relates the minimum required coolant temperature to the allowable measured reactor pressure.

g. For the non-beltline P-T limits, apply the additional minimum temperature requirements described in Sections 2.7 and 2.8.

h. Apply any applicable adjustments to the final temperatures and pressures, as described in Section 2.6.

Typical P-T limit Curves A and B generated from the above procedure are shown in Figures 2-2 and 2-3.

A template PTLR is included in Appendix B of this report. The supporting documents referenced by the PTLR contain all calculations necessary for the development of the P-T curves contained in the PTLR in accordance with the above steps and the methodology provided in this report.

4.0 REFERENCES

1. U. S. Nuclear Regulatory Commission, Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996.

2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," Office of Nuclear Regulatory Research, (Task ME 305-4), May 1988.

3. ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, "Class 1 Components," 2004 Edition.

4. U. S. Code of Federal Regulations, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix G, "Fracture Toughness Requirements," 1/1/05 Edition.

5. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 2004 Edition.

6. Materials and Chemical Engineering Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements," NUREG-0800 Standard Review Plan, Section 5.3.2, "Pressure-Temperature Limits," July 1981, Revision 1.

7. U. S. Code of Federal Regulations, Title 10, Part 50, Domestic Licensing of Production and Utilization Facilities, §50.12, "Specific Exemptions," 1/1/05 Edition.

8. Welding Research Council Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," PVCR Ad Hoc Group on Toughness Requirements, Welding Research Council, August 1972.

9. W. H. Bamford, G. L. Stevens, T. J. Griesbach, and S. N. Malik, "Technical Basis for Revised P-T Limit Curve Methodology," American Society of Mechanical Engineers, Pressure Vessels and Piping Division (Publication), ASME PVP Conference, Volume 407, pp. 169-178, 2000.

10. ASME Boiler and Pressure Vessel Code, Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves," Section XI, Division 1, Approved February 26, 1999.

11. ASME Boiler and Pressure Vessel Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements," Section XI, Division 1, Approved January 17, 2000.

12. Westinghouse Owners Group Letter No. OG-02-219 from Robert H. Bryan (WOG Chairman) to Document Control Desk (NRC), "Transmittal of WCAP-15315, Rev. 1, 'Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants,' (MUHP-3073)," May 23, 2002.

13. S. P. Timoshenko and J. N. Goodier, Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

14. ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix A, "Analysis of Flaws," 2004 Edition.

15. Westinghouse Report No. WCAP-15315, Revision 1, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants," Westinghouse Non-Proprietary Class 3, 2002.

16. Westinghouse Report No. WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Westinghouse Non-Proprietary Class 3, 2002.

17. J. P. Holman, Heat Transfer, Fourth Edition, McGraw-Hill Book Co., New York, 1976, pp. 2-19.

APPENDIX A:
GUIDANCE FOR THE USE OF BWRVIP ISP SURVEILLANCE DATA

This appendix provides guidance for the use of BWR surveillance data for developing pressure-temperature limit curves and other vessel integrity evaluations.

A.1 Introduction

The BWRVIP Integrated Surveillance Program (ISP) replaces individual plant reactor pressure vessel surveillance capsule programs with representative weld and base materials data from host reactors [A-1]. A representative material is a plate or weld material that is selected from among all the existing plant surveillance programs or the Supplemental Surveillance Program (SSP) [A-2] to represent one or more limiting plate or weld materials in a plant. The BWRVIP ISP is responsible to provide each BWR plant with surveillance data for the materials assigned to represent that plant's limiting vessel weld and base materials. Plant owners, in turn are responsible to evaluate the data using the methods in Regulatory Guide 1.99, Revision 2 [A-3], in accordance with 10CFR50, Appendix G, for determination of Adjusted Reference Temperature (ART) values.

Surveillance and chemistry data for all representative materials in the ISP have been evaluated by the BWRVIP. For each material that has been designated as an ISP representative material, a comprehensive material summary has been developed. All baseline and irradiated Charpy data for ISP surveillance materials have been obtained from past surveillance program and capsule reports. The data were reanalyzed, using consistent analysis standards and protocols. Best estimate chemistry values were also calculated in a manner consistent with USNRC guidance [A-4].

The BWRVIP ISP has been generically approved by the NRC and is documented in a safety evaluation [A-5]. Owners incorporate the ISP on a plant-specific basis via a license amendment.

A.2 Guidance for Processing Surveillance Data

The following process is recommended for evaluating surveillance data:

1. If there is new surveillance data for any heat which is located in the vessel beltline (e.g., heat numbers match), then Procedure #1 can be used as a guide for evaluating the new information. A new ART should be calculated for the vessel material to determine whether plant vessel integrity evaluations are affected.

2. If there is new information but that same heat number is not contained in the vessel beltline, then Procedure #2 can be used as a guide for evaluating the new information.

A.3 Reporting

The following information should be reported to the BWRVIP following the evaluation of surveillance data.

1. After vessel integrity evaluations (e.g., ART tables) are updated, the plant should provide an informational copy of the revised ART tables for the beltline materials to the BWRVIP ISP Project Manager. This will assist the BWRVIP during its annual ISP program review to revalidate the ISP Test Matrix.

2. As an ongoing "maintenance" activity, all plants should inform the BWRVIP ISP Project Manager whenever its fluence calculations are updated. It is essential that the following information be promptly reported to the BWRVIP ISP Project Manager:

   a. Updated fluence values for the beltline region inside surface and 1/4t positions;
   b. Revised capsule fluence estimates;
   c. Revised ART calculations for beltline materials resulting from the revised fluence, with fluence, CF, and margin clearly specified for each material.

This information is particularly vital to the BWRVIP ISP, because any revisions to capsule fluence estimates can affect RT_{NDT} shift calculations for that material – with a direct effect on any other plants using that data for CF.

Procedure #1

Recommended Guidance for the Use of ISP Surveillance Data when Vessel Material and Surveillance Material Heat Numbers Are Identical

Prerequisites

This procedure provides recommended guidance for the use of BWRVIP ISP surveillance data only when the following condition is met:

1. The heat number of the vessel beltline material being evaluated and the heat number of the surveillance material (e.g., the ISP Representative Material or other material) are identical.

Objective

The objective of this procedure is to determine the Adjusted Reference Temperature (ART) for the vessel material as determined by the following expression:

$$\text{ART} = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (1)$$

This procedure is designed to determine the "ΔRT_{NDT}" and "Margin" terms of the ART equation. The "Initial RT_{NDT}" is established by the plant according to the definition below.

Definitions and Background

The guidance provided by this procedure is based on Regulatory Guide 1.99, Rev. 2, with clarifications as noted by References [A-4] (1998 NRC Presentation) and [A-6] (10CFR50.61, PTS Rule).

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. Some plants have measured values of Initial RT_{NDT}; other plants use generic values. For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes [A-6].

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation, as calculated by the equation:

$$\Delta RT_{NDT} = (CF)\, f^{(0.28 - 0.1 \log f)} \qquad (2)$$

where CF (°F) is the chemistry factor. The CF can either be a function of copper and nickel content, as given in Reg. Guide 1.99 Rev. 2, Table 1 (welds) or Table 2 (base metal), which are repeated in this appendix as Table A-1 and Table A-2, respectively, or a factor based on the "best fit" of two or more surveillance test data.

The neutron fluence at any depth in the vessel wall, f (10^{19} n/cm^2, E > 1 MeV), is determined as follows:

$$f = f_{surf}\,(e^{-0.24x}) \qquad (3)$$

where f_{surf} (10^{19} n/cm^2, E > 1 MeV) is the calculated value of the neutron fluence at the vessel inner surface, and x (in inches) is the depth into the vessel wall measured from the vessel inner surface. The depth of interest for this calculation is the 1/4t position in the vessel wall.

The fluence factor, $f^{(0.28 - 0.1 \log f)}$, is determined by calculation from the fluence.

"Margin" is the quantity, °F, that is to be added to obtain conservative upper-bound values of adjusted reference temperature required by Appendix G to 10CFR, Part 50,

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \qquad (4)$$

where σ_I is the standard deviation for the initial RT_{NDT}. If a measured value of initial RT_{NDT} for the material in question is available, σ_I is to be estimated from the precision of the test method (and it is normally taken to be 0°F). If not, and generic mean values for the class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. If the generic mean Initial RT_{NDT} value of a Linde 80, 0091, 1092 and 124 or ARCOS B-5 weld is used, then σ_I is 17°F [A-6]. The standard deviation for ΔRT_{NDT}, σ_Δ, is 28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

Procedural Steps

1. ***Verify Heat Number Match***

   This recommended procedure is applicable only in the case that the heat number of the vessel beltline material being evaluated and the heat number of the surveillance material (e.g., the ISP Representative Material or other material) are identical. If not, then Procedure #2, "Recommended Guidance for the Use of ISP Surveillance Data When the Vessel Material and Surveillance Material Heat Numbers Do Not Match," should be used.

2. ***Identify Available Surveillance Data for this Heat***

   Review the ISP surveillance data for this heat. Are there two or more reported surveillance data points for this material? If YES, proceed to Step 3. If NO, then skip to Step 5.

3. ***Determine Credibility of Surveillance Data***

   The objective of this step is to verify that there are two or more valid, credible surveillance data points for this heat.

   The BWRVIP analysis of the surveillance data for this heat should be reviewed.

a. Confirm that the vessel wall temperature at the cladding/base metal interface (in the beltline region) is within +/- 25°F of the BWR capsule irradiation temperature range of 525°F to 535°F.
b. If the vessel wall temperature is an outlier, appropriate temperature adjustments to the surveillance data may be required.
c. If the vessel temperature credibility criterion is confirmed, then the plant should declare the surveillance data to be "credible" or "not credible" for its vessel, depending on the BWRVIP evaluation of the data scatter criterion.

    Note: Classification of the surveillance data as "credible" or "not credible" does not determine whether or not the data will be used. Under certain circumstances, the NRC requires the Chemistry Factor to be based on non-credible surveillance data, if the Table CF is non-conservative in comparison [A-4]. Those circumstances will be explained in detail in the following steps.

4. ***Determine Chemistry Factor (Two or more Surveillance Data)***

This step applies only when there are two or more surveillance data points available. If there is only one surveillance data point, or no data, then skip to Step 5.

The CF is based either on the Reg. Guide 1.99 Rev. 2 tables, or on the best fit of the surveillance data, according to the guidance below.

If the material being evaluated is a plate, determine the Chemistry Factor according to Step 4.a. If the material is a weld, determine Chemistry Factor according to Step 4.b.

4.a. Determine CF for a Plate Material

1) Determine the Table CF (that is, the CF given in Table 2 of Reg. Guide 1.99 Rev. 2, duplicated in this appendix as Table A-2) for the best estimate chemistry of the vessel plate.
2) Compare this Table CF to the surveillance CF (e.g., the CF determined by a best fit to the surveillance data) reported by the BWRVIP.

3) If the fitted data give a higher value of CF than the tables, then surveillance data CF should be used. This is true even if the surveillance data were not credible (Reference [A-4], Case 3).
4) If the fitted results give a lower value, and the surveillance data are credible, then either the Table CF or the surveillance CF value may be used. If the fitted results give a lower value, and the surveillance data are not credible, then the higher (e.g., Table CF) must be used (Reference [A-4], Case 2).
5) Skip to Step 6.

4.b. Determine CF for a Weld Material

If the measured copper or nickel content of the surveillance weld differs from that of the vessel weld of the same heat, (i.e., the surveillance weld best estimate chemistry differs from the vessel weld best estimate chemistry), the fitted CF from the surveillance data should be adjusted by multiplying it by the ratio of the Reg. Guide 1.99 Rev. 2 table chemistry factor for the vessel weld to that for the surveillance weld. The following steps incorporate this adjustment:

1) Determine the Table CF (that is, the CF given in Table 1 of Reg. Guide 1.99 Rev. 2, duplicated in this appendix as Table A-1) for the best estimate chemistry of the vessel weld.

   Note: Revised best estimate chemistries for selected BWR welds and plates have been calculated by the BWRVIP. Calculation of the best estimate chemistries for all other vessel materials is the responsibility of the plant.

2) Determine the Table CF for the best estimate chemistry of the surveillance weld (Table $CF_{Surv.\ Chem}$).
3) Calculate an Adjusted Surveillance CF by the following equation:

$$\text{Adjusted Surv. CF} = \left(\frac{\text{Table } CF_{\text{Vessel Chem.}}}{\text{Table } CF_{\text{Surv. Chem.}}}\right) * CF_{\text{Fitted Data}} \qquad (5)$$

4) Compare the Adjusted Surveillance CF to the Table $CF_{Vessel\ Chem.}$.

5) If the Adjusted Surveillance CF is higher than the Table $CF_{Vessel\ Chem}$, then the Adjusted Surveillance CF should be used as the CF in Step 5 (calculation of ΔRT_{NDT}). This is true even if the surveillance data were not credible because of excessive scatter.
6) If the Adjusted Surveillance CF is less than the Table $CF_{Vessel\ Chem}$, and the surveillance data are credible, then either the Table CF or the Adjusted Surv. CF value may be used. If the Adjusted Surveillance CF is less than the Table $CF_{Vessel\ Chem}$, and the surveillance data are not credible, then the higher (e.g., Table $CF_{Vessel\ Chem}$) must be used.
7) Skip to Step 6.

5. Determine Chemistry Factor (No Surveillance Data, or One Data Point)

This step applies only when there is only one, or less, surveillance data points available. If there are two or more surveillance data points, do not use Step 5; go back to Step 4.

The CF for the vessel material should be determined from the Reg. Guide 1.99 Rev. 2 tables (duplicated in this appendix as Tables A-1 and A-2), based on the best estimate chemistry of the vessel material.

> *Note: Revised best estimate chemistries for selected BWR welds and plates have been calculated by the BWRVIP. Calculation of the best estimate chemistries for all other vessel materials is the responsibility of the plant.*

After the CF associated with the best estimate chemistry of the vessel heat is determined from Reg. Guide 1.99 Rev. 2, Table 1 (Welds) or Table 2 (Plates), duplicated in this appendix as Tables A-1 and A-2, respectively, proceed to Step 6.

6. *Calculate* ΔRT_{NDT}

Calculate the transition temperature shift at the 1/4t position in the vessel, $\Delta RT_{NDT\ 1/4T}$, using the appropriate CF value determined in Step 4 or 5 and the projected fluence at the 1/4t location, $f_{1/4T}$, using equation (6):

$$\Delta RT_{NDT1/4T} = CF\,(f_{1/4T})^{(0.28-0.1\log f_{1/4T})} \tag{6}$$

7. *Determine Margin*

The margin term is calculated by Equation (4). If the surveillance data are credible, the values given there for σ_Δ may be cut in half. Therefore:

a) For credible surveillance data, σ_Δ is the lower of the following:
   - a) 14°F for welds, 8.5°F for base metal, <u>or</u>
   - b) 0.50 times the mean value of ΔRT_{NDT}.

b) If the surveillance data are <u>not</u> credible, then σ_Δ is the lower of the following:
   - a) 28°F for welds, 17°F for base metal, <u>or</u>
   - b) 0.50 times the mean value of ΔRT_{NDT}.

8. *Calculate the ART for the Vessel Material*

Calculate the ART for the vessel material using Equation (1) and the values for ΔRT_{NDT} and Margin determined above.

Procedure #2

Recommended Guidance for the Use of ISP Surveillance Data when Vessel Material and Surveillance Material Heat Numbers Do Not Match

Prerequisites

This procedure provides recommended guidance for the use of BWRVIP ISP surveillance data only when the heat number of the vessel beltline material being evaluated and the heat number of the surveillance material (e.g., the ISP Representative Material) do not match.

Objective

The objective of this procedure is to determine the Adjusted Reference Temperature (ART) for the vessel material as determined by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (1)$$

This procedure is designed to assist the plants in using the ISP surveillance data to determine the "ΔRT_{NDT}" and "Margin" terms of the ART equation. The "Initial RT_{NDT}" is established by the plant according to the definition below.

Definitions and Background

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. Some plants have measured values of Initial RT_{NDT}; other plants use generic values. For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation, as calculated by the equation:

$$\Delta RT_{NDT} = (CF)\, f^{(0.28 - 0.1 \log f)} \qquad (2)$$

where CF (°F) is the chemistry factor. The CF can either be a function of copper and nickel content, as given in Reg. Guide 1.99 Rev. 2, Table 1 (welds) or Table 2 (base metal), duplicated in this appendix as Tables A-1 and A-2, respectively, or a factor based on the "best fit" of two or more surveillance test data. For the materials being evaluated by this procedure, only the Reg. Guide tables will be used.

The neutron fluence at any depth in the vessel wall, f (10^{19} n/cm^2, E > 1 MeV), is determined as follows:

$$f = f_{surf}\,(e^{-0.24x}) \qquad (3)$$

where f_{surf} (10^{19} n/cm^2, E > 1 MeV) is the calculated value of the neutron fluence at the vessel inner surface, and x (in inches) is the depth into the vessel wall measured from the vessel inner surface. The depth of interest for this calculation is the 1/4t position in the vessel wall.

The fluence factor, $f^{(0.28 - 0.1 \log f)}$, is determined by calculation from the fluence.

"Margin" is the quantity, °F, that is to be added to obtain conservative upper-bound values of adjusted reference temperature required by Appendix G to 10CFR, Part 50,

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \qquad (4)$$

where σ_I is the standard deviation for the initial RT_{NDT}. If a measured value of initial RT_{NDT} for the material in question is available, σ_I is to be estimated from the precision of the test method (and it is normally taken to be 0°F). If not, and generic mean values for the class of material are used, σ_I is the standard deviation obtained from the set of data used to establish the mean. If the generic mean Initial RT_{NDT} value of a Linde 80, 0091, 1092, and 124 or

ARCOS B-5 weld is used, then σ_I is 17°F. The standard deviation for ΔRT_{NDT}, σ_Δ , is 28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

Procedural Steps

1. ***Verify Heat Numbers Do Not Match***

   This recommended procedure is applicable only in the case that the heat number of the vessel beltline material being evaluated and the heat number of the surveillance material (e.g., the ISP Representative Material or other material) do not match. If they do match, then Procedure #1, "Recommended Guidance for the Use of ISP Surveillance Data When Vessel Material and Surveillance Material Heat Numbers Are Identical" should be used.

2. ***Review Surveillance Data for the Assigned ISP Representative Material***

   All surveillance data for the ISP representative materials have been analyzed by the BWRVIP.

3. ***Determine Chemistry Factor***

   The CF for the vessel material should be determined from the Reg. Guide 1.99 Rev. 2 Table 1 (Welds) or Table 2 (Plates), duplicated in this appendix as Tables A-1 and A-2, respectively, based on the best estimate chemistry of the vessel material.

   > *Note: Revised best estimate chemistries for selected BWR welds and plates have been calculated by the BWRVIP. Calculation of the best estimate chemistries for all other vessel materials is the responsibility of the plant.*

4. ***Calculate*** ΔRT_{NDT}

Calculate the transition temperature shift at the 1/4T position in the vessel, $\Delta RT_{NDT\ 1/4T}$, using the CF value determined in Step 3 and the projected fluence at the 1/4T location, $f_{1/4T}$, using equation (6):

$$\Delta RT_{NDT1/4T} = CF\,(f_{1/4T})^{(0.28-0.1\log f_{1/4T})} \tag{6}$$

5. ***Determine Margin***

The margin term is calculated by Equation (4). σ_Δ is the lower of the following:

a. 28°F for welds, 17°F for base metal, <u>or</u>

b. 0.50 times the mean value of ΔRT_{NDT}.

6. ***Calculate the ART for the Vessel Material***

Calculate the ART for the vessel material using Equation (1) and the values for ΔRT_{NDT} and Margin determined above.

Table A-1

Chemistry Factor for Welds, °F

Copper Wt-%	Nickel, Wt-%						
	0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	21	26	27	27	27	27	27
0.03	22	35	41	41	41	41	41
0.04	24	43	54	54	54	54	54
0.05	26	49	67	68	68	68	68
0.06	29	52	77	82	82	82	82
0.07	32	55	85	95	95	95	95
0.08	36	58	90	106	108	108	108
0.09	40	61	94	115	122	122	122
0.10	44	65	97	122	133	135	135
0.11	49	68	101	130	144	148	148
0.12	52	72	103	135	153	161	161
0.13	58	76	106	139	162	172	176
0.14	61	79	109	142	168	182	188
0.15	66	84	112	146	175	191	200
0.16	70	88	115	149	178	199	211
0.17	75	92	119	151	184	207	221
0.18	79	95	122	154	187	214	230
0.19	83	100	126	157	191	220	238
0.20	88	104	129	160	194	223	245
0.21	92	108	133	164	197	229	252
0.22	97	112	137	167	200	232	257
0.23	101	117	140	169	203	236	263
0.24	105	121	144	173	206	239	268
0.25	110	126	148	176	209	243	272
0.26	113	130	151	180	212	246	276
0.27	119	134	155	184	216	249	280
0.28	122	138	160	187	218	251	284
0.29	128	142	164	191	222	254	287
0.30	131	146	167	194	225	257	290
0.31	136	151	172	198	228	260	293
0.32	140	155	175	202	231	263	296
0.33	144	160	180	205	234	266	299
0.34	149	164	184	209	238	269	302
0.35	153	168	187	212	241	272	305
0.36	158	172	191	216	245	275	308
0.37	162	177	196	220	248	278	311
0.38	166	182	200	223	250	281	314
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

Table A-2

Chemistry Factor for Base Metal, °F

Copper Wt-%	Nickel, Wt-% 0	0.20	0.40	0.60	0.80	1.00	1.20
0	20	20	20	20	20	20	20
0.01	20	20	20	20	20	20	20
0.02	20	20	20	20	20	20	20
0.03	20	20	20	20	20	20	20
0.04	22	26	26	26	26	26	26
0.05	25	31	31	31	31	31	31
0.06	28	37	37	37	37	37	37
0.07	31	43	44	44	44	44	44
0.08	34	48	51	51	51	51	51
0.09	37	53	58	58	58	58	58
0.10	41	58	65	65	67	67	67
0.11	45	62	72	74	77	77	77
0.12	49	67	79	83	86	86	86
0.13	53	71	85	91	96	96	96
0.14	57	75	91	100	105	106	106
0.15	61	80	99	110	115	117	117
0.16	65	84	104	118	123	125	125
0.17	69	88	110	127	132	135	135
0.18	73	92	115	134	141	144	144
0.19	78	97	120	142	150	154	154
0.20	82	102	125	149	159	164	165
0.21	86	107	129	155	167	172	174
0.22	91	112	134	161	176	181	184
0.23	95	117	138	167	184	190	194
0.24	100	121	143	172	191	199	204
0.25	104	126	148	176	199	208	214
0.26	109	130	151	180	205	216	221
0.27	114	134	155	184	211	225	230
0.28	119	138	160	187	216	233	239
0.29	124	142	164	191	221	241	248
0.30	129	146	167	194	225	249	257
0.31	134	151	172	198	228	255	266
0.32	139	155	175	202	231	260	274
0.33	144	160	180	205	234	264	282
0.34	149	164	184	209	238	268	290
0.35	153	168	187	212	241	272	298
0.36	158	173	191	216	245	275	303
0.37	162	177	196	220	248	278	308
0.38	166	182	200	223	250	281	313
0.39	171	185	203	227	254	285	317
0.40	175	189	207	231	257	288	320

A.5 References

A-1. BWRVIP-86-A: BWR Vessel and Internal Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan, EPRI, Palo Alto, CA: 2002. 1003346.

A-2. "Progress Report on Phase 2 of the BWR Owners' Group Supplemental Surveillance Program," T.A. Caine, S. Ranganath, and S.J. Stark, GE Nuclear Energy, GE-NE-523-99-0792, January 1992.

A-3. "Radiation Embrittlement of Reactor Vessel Materials," USNRC Regulatory Guide 1.99, Revision 2, May 1988.

A-4. USNRC, Generic Letter 92-01 and RPV integrity Workshop Handouts, K. Wichman, M. Mitchell, and A. Hiser, NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998.

A-5. Letter from William H. Bateman (NRC) to Carl Terry (BWRVIP Chairman), Safety Evaluation Regarding EPRI Proprietary Reports "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)" and "BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," dated February 1, 2002.

A-6. 10 CFR 50.61, *Fracture toughness requirements for protection against pressurized thermal shock events*, Federal Register, Volume 60, No. 243, dated December 19, 1995.

APPENDIX B:
TEMPLATE PTLR

PPL Susquehanna, LLC

Susquehanna Steam Electric Station
Units 1 and 2

Pressure And Temperature Limits Report (PTLR) up to 32 Effective Full-Power Years (EFPY)

Revision 0

Prepared by: ______________________ Date: ______

Reviewed by: ______________________ Date: ______

Approved by: ______________________ Date: ______
[Director, Engineering]

Concurred by: ______________________ Date: ______
[Manager, Licensing]

Table of Contents

1.0 Purpose

The purpose of the Susquehanna Steam Electric Station (SSES) Pressure and Temperature Limits Report (PTLR) is to present operating limits relating to:

1) Reactor Coolant System (RCS) Pressure versus Temperature limits during Heatup, Cooldown and Hydrostatic/Class 1 Leak Testing;
2) RCS Heatup and Cooldown rates;
3) Reactor Pressure Vessel (RPV) to RCS coolant ΔT requirements during Recirculation Pump startups;
4) RPV bottom head coolant temperature to RPV coolant temperature ΔT requirements during Recirculation Pump startups;
5) RPV head flange boltup temperature limits.

This report has been prepared in accordance with the requirements of Technical Specification (TS) 5.6.6, "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)."

2.0 Applicability

This report is applicable to the SSES Units 1 and 2 RPVs up to 32 Effective Full-Power Years (EFPY).

The following TS is affected by the information contained in this report:

TS 3.4.10 RCS Pressure and Temperature (P/T) Limits;

3.0 Methodology

The limits in this report were derived from the NRC-approved methods listed in TS 5.6.6, using the specific revisions listed below:

1) The neutron fluence was calculated per the DORT computer code, approved in Reference 6.1.

2) The pressure and temperature limits were calculated per Structural Integrity Associates, Inc. Report No. SIR-05-044, Revision C, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," October 2005. The methodology used was previously approved in Reference 6.2.

3) This revision of the pressure and temperature limits is to incorporate the following changes:

   - Initial issue of PTLR.

Changes to the curves, limits, or parameters within this PTLR, based upon new irradiation fluence data of the RPV, or other plant design assumptions in the Updated Final Safety Analysis Report (UFSAR), can be made pursuant to 10 CFR 50.59, provided the above methodologies are utilized. The revised PTLR shall be submitted to the NRC upon issuance.

Changes to the curves, limits, or parameters within this PTLR, based upon new surveillance capsule data of the RPV, cannot be made without prior NRC approval. Such analysis and revisions shall be submitted to the NRC for review prior to incorporation into the PTLR.

4.0 Operating Limits

The pressure-temperature (P-T) curves included in this report represent steam dome pressure versus minimum vessel metal temperature and incorporate the appropriate non-beltline limits and irradiation embrittlement effects in the beltline region.

The operating limits for pressure and temperature are required for three categories of operation: (a) hydrostatic pressure tests and leak tests, referred to as Curve A; (b) core not critical operation, referred to as Curve B; and (c) core critical operation, referred to as Curve C.

Complete P-T curves were developed for 32 EFPY for SSES Units 1 and 2, as documented in Reference 6.3. The SSES Unit 1 P-T curves are provided in Figures 1 through 3, and a tabulation of the curves is included in Tables 1 through 3. The SSES Unit 2 P-T curves are provided in Figures 4 through 6, and a tabulation of the curves is included in Tables 4 through 6.

Heatup and Cooldown rate limit during Hydrostatic and Class 1 Leak Testing (Figures 1 and 4: Curve A): **≤ 25°F/hour**[1].

Normal Operating Heatup and Cooldown rate limit (Figures 2 and 5: Curve B - non-nuclear heating, and Figures 3 and 6: Curve C - nuclear heating): **≤ 100°F/hour**[2].

---

[1] Interpreted as the temperature change in any 1-hour period is less than or equal to 25°F.

[2] Interpreted as the temperature change in any 1-hour period is less than or equal to 100°F.

RPV bottom head coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 145°F.**

Recirculation loop coolant temperature to RPV coolant temperature ΔT limit during Recirculation Pump startup: **≤ 50°F.**

RPV flange and adjacent shell temperature limit: **≥ 70°F.**

5.0 Discussion

The adjusted reference temperature (ART) of the limiting beltline material is used to adjust the beltline P-T curves to account for irradiation effects. Regulatory Guide 1.99, Revision 2 (RG 1.99) provides the methods for determining the ART. The RG 1.99 methods for determining the limiting material and adjusting the P-T curves using ART are discussed in this section.

The vessel beltline copper and nickel values were obtained from the evaluation of the SSES Surveillance Capsules (References 6.4 and 6.5). The copper (Cu) and nickel (Ni) values were used with Tables 1 and 2 of RG 1.99 to determine a chemistry factor (CF) per Paragraph 1.1 of RG 1.99 for welds and plates, respectively.

The peak RPV ID fluence used in the P-T curve evaluation for 32 EFPY is 9.2×10^{17} n/cm^2 for SSES Unit 1, and 7.8×10^{17} n/cm^2 for SSES Unit 2[*, which were calculated using methods that comply with the guidelines of RG 1.190 (Reference 6.1) – Editorial Note: It is* *recognized that this clause does not apply for the example given in this template, but this is a mandatory requirement for any PTLR submittals].*

These fluence values apply to the limiting lower-intermediate plates for both SSES units. The fluence values were adjusted for the lower intermediate plates based upon an attenuation factor of 0.691 for a postulated 1/4t flaw. As a result, the 1/4t fluence for the limiting lower-intermediate plates is 6.4×10^{17} n/cm^2 for SSES Unit 1, and 5.4×10^{17} n/cm^2 for SSES Unit 2.

The P-T curves for the core not critical and core critical operating conditions at a given EFPY apply for both the 1/4t and 3/4t locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4t location (inside surface flaw) and the 3/4t location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stress at the 1/4t location is assumed to be tensile for both heatup and cooldown. This results in the approach of applying the maximum tensile stress at the 1/4t location. This approach is conservative because irradiation effects cause the allowable toughness at 1/4t to be less than that at 3/4t for a given metal temperature. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, which is well above the P-T curve limits.

For the core not critical curve (Curve B) and the core critical curve (Curve C), the P-T curves specify a coolant heatup and cooldown temperature rate of ≤ 100°F/hr for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram and the nozzle thermal cycle diagrams. For the hydrostatic pressure and leak test curve (Curve A), a coolant heatup and cooldown temperature rate of ≤ 25°F/hr must be maintained. The P-T limits and corresponding limits of either Curve A or B may be applied, if necessary, while achieving or recovering from test conditions. So, although Curve A applies during pressure testing, the limits of Curve B may be conservatively used during pressure testing if the pressure test heatup/cooldown rate limits cannot be maintained.

6.0 References

6.1 SSES FSAR, Section 4.1.5.

6.2 *NRC approval letter for Structural Integrity Associates, Inc. Report No. SIR-05-044, Revision 0, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," December 2005 -- LATER.*

6.3 Structural Integrity Associates, Inc. Report No. SIR-00-167, Revision 0, "Revised Pressure-Temperature Curves for Susquehanna Units 1 and 2," February 20, 2001.

6.4 General Electric Report No. GE-NE-523-169-1292, DRF B13-01666, "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," March 1993.

6.5 General Electric Report No. GE-NE-523-107-0893, DRF 137-0010-6, Revision 1A, "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," November 20, 2002.

Figure 1: SSES Unit 1 Pressure Test (Curve A) P-T Curves

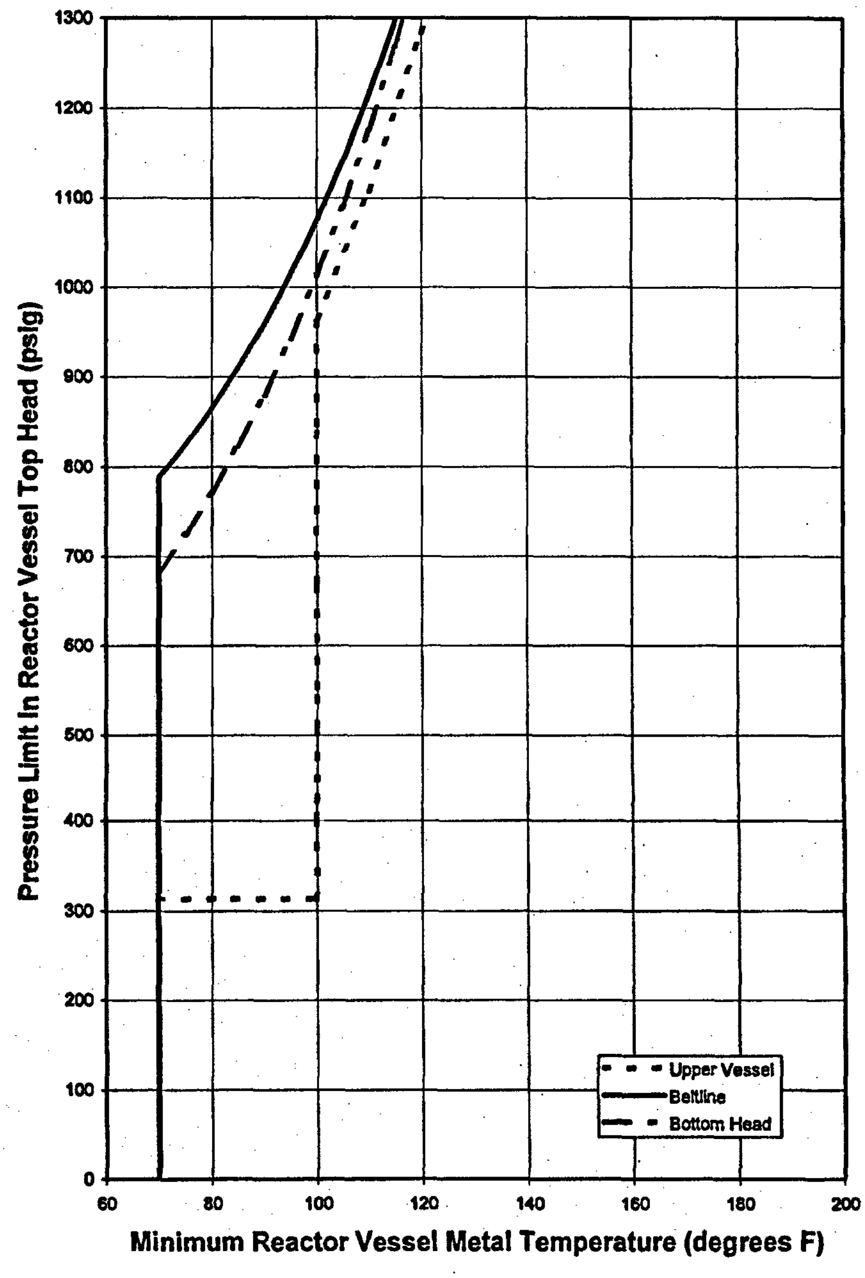


Figure 2: SSES Unit 1 Core Not Critical (Curve B) P-T Curves

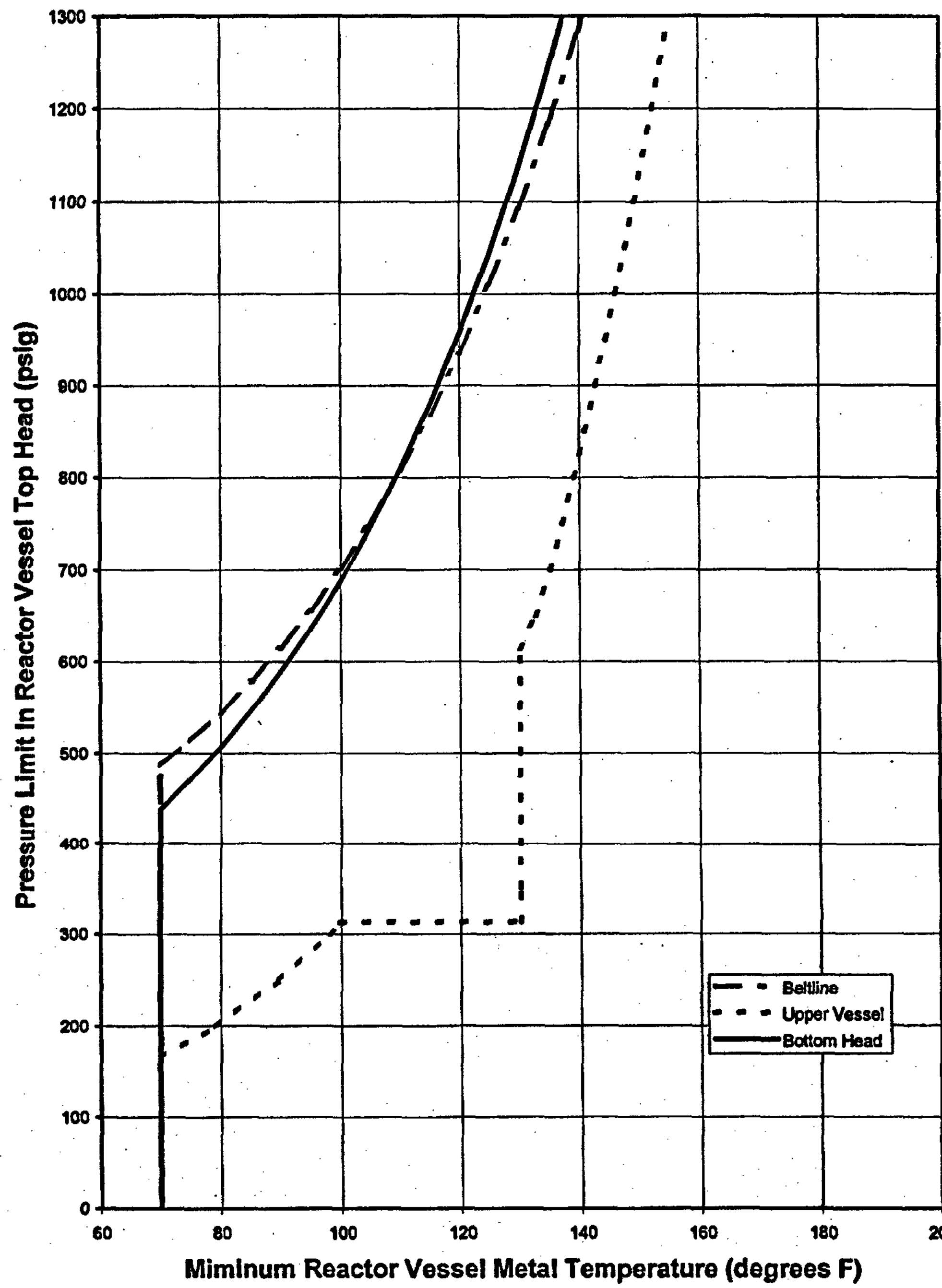


Figure 3: SSES Unit 1 Core Critical (Curve C) P-T Curve

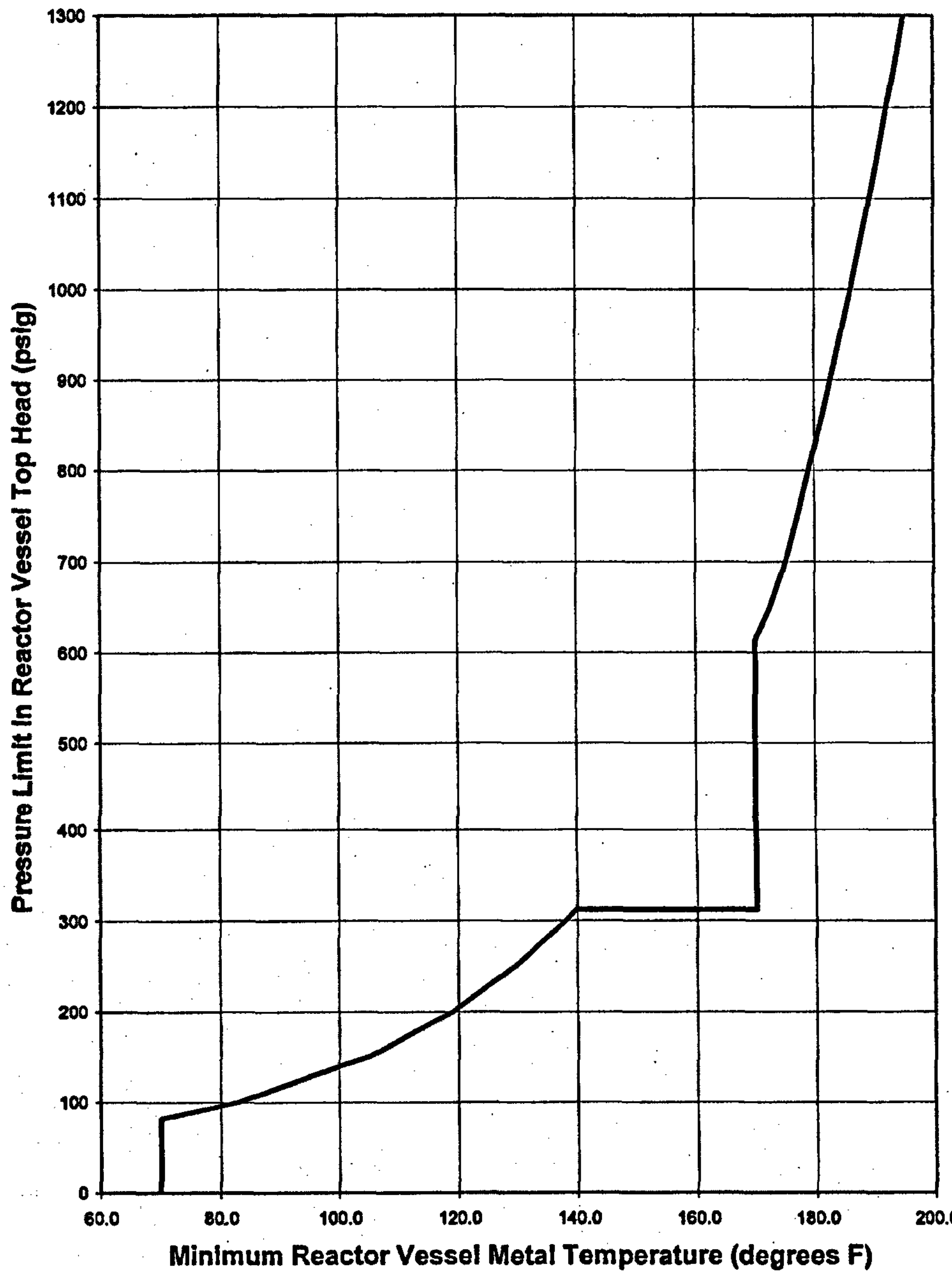


Figure 4: SSES Unit 2 Pressure Test (Curve A) P-T Curves

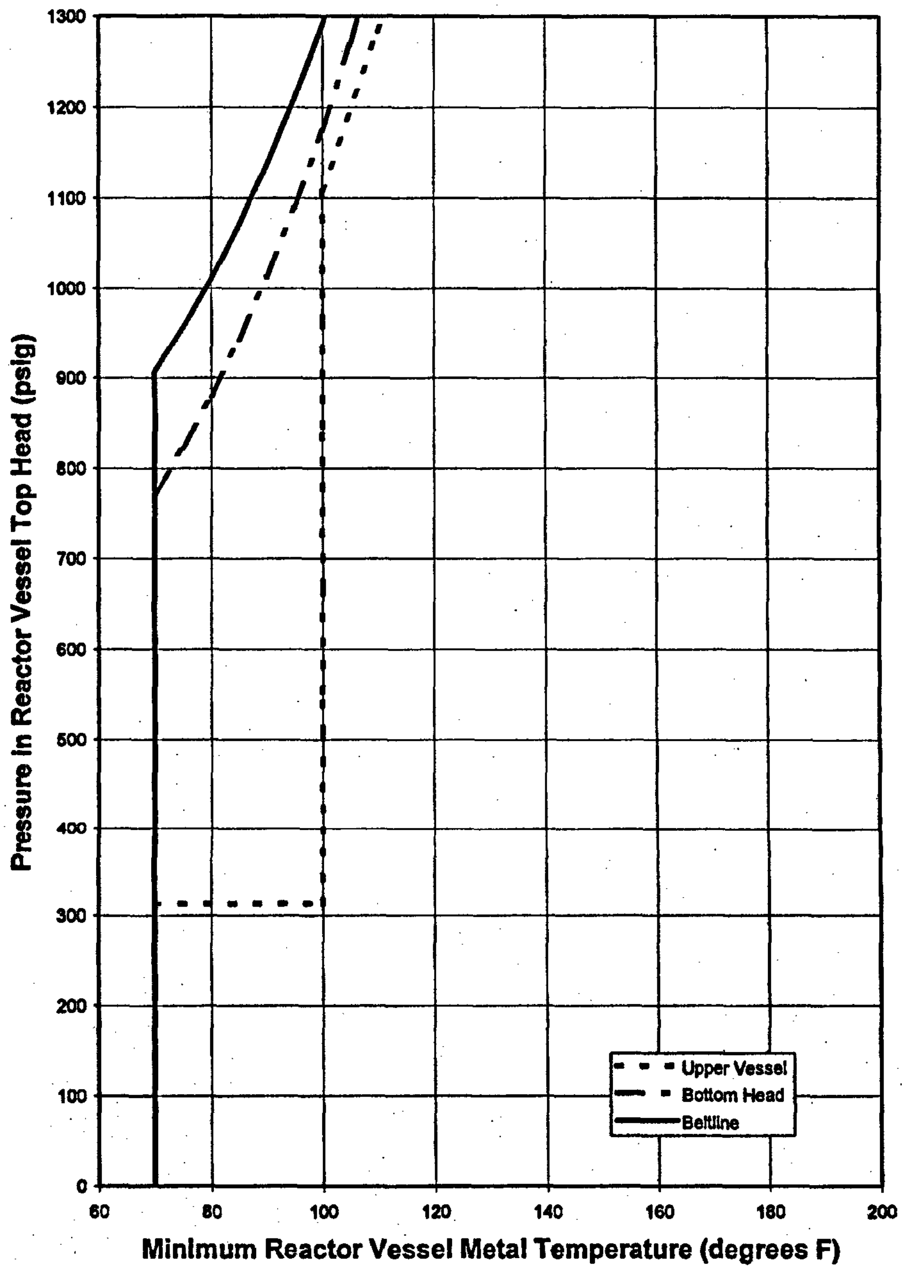


Figure 5: SSES Unit 2 Core Not Critical (Curve B) P-T Curves

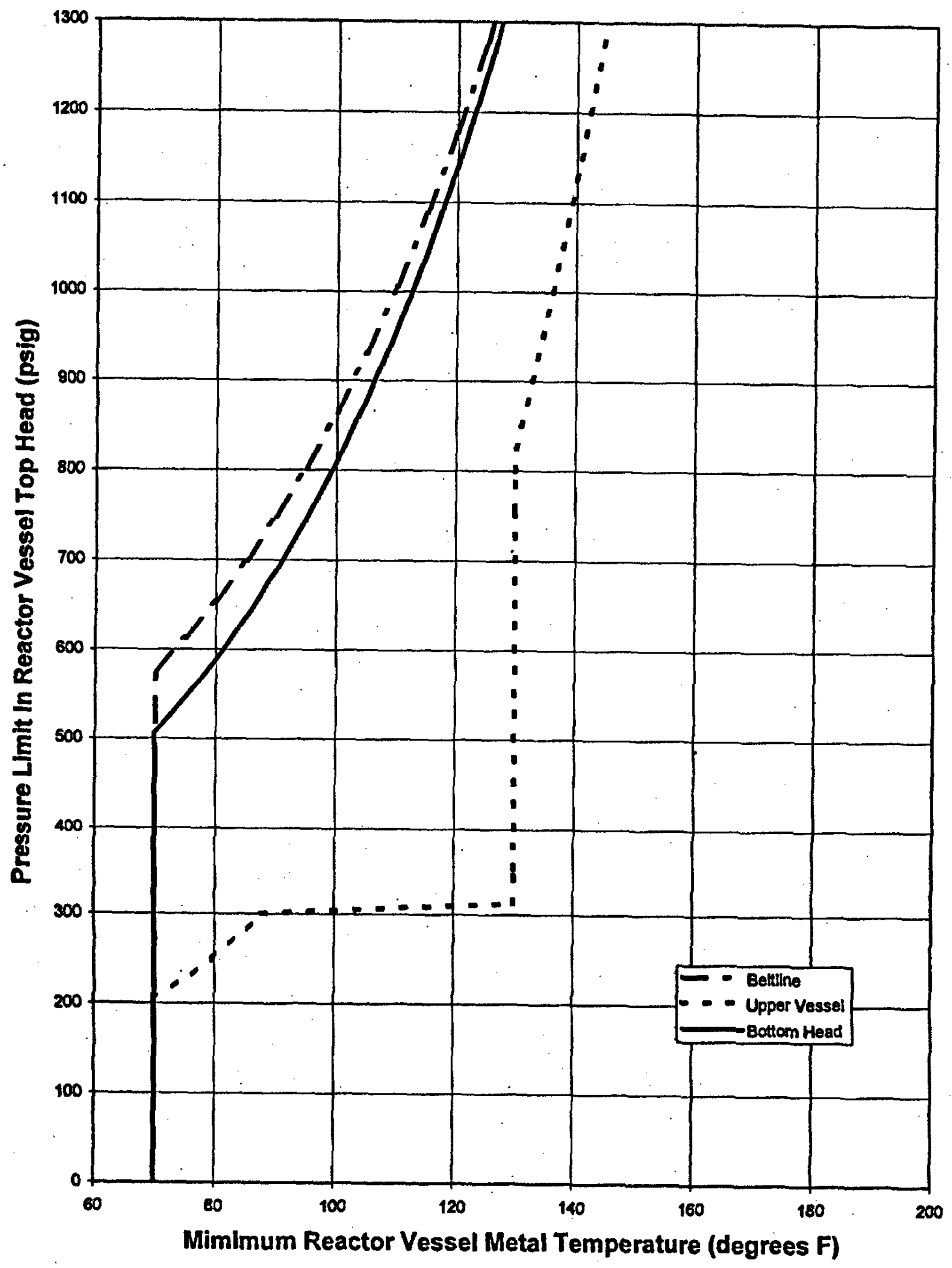


Figure 6: SSES Unit 2 Core Critical (Curve C) P-T Curve

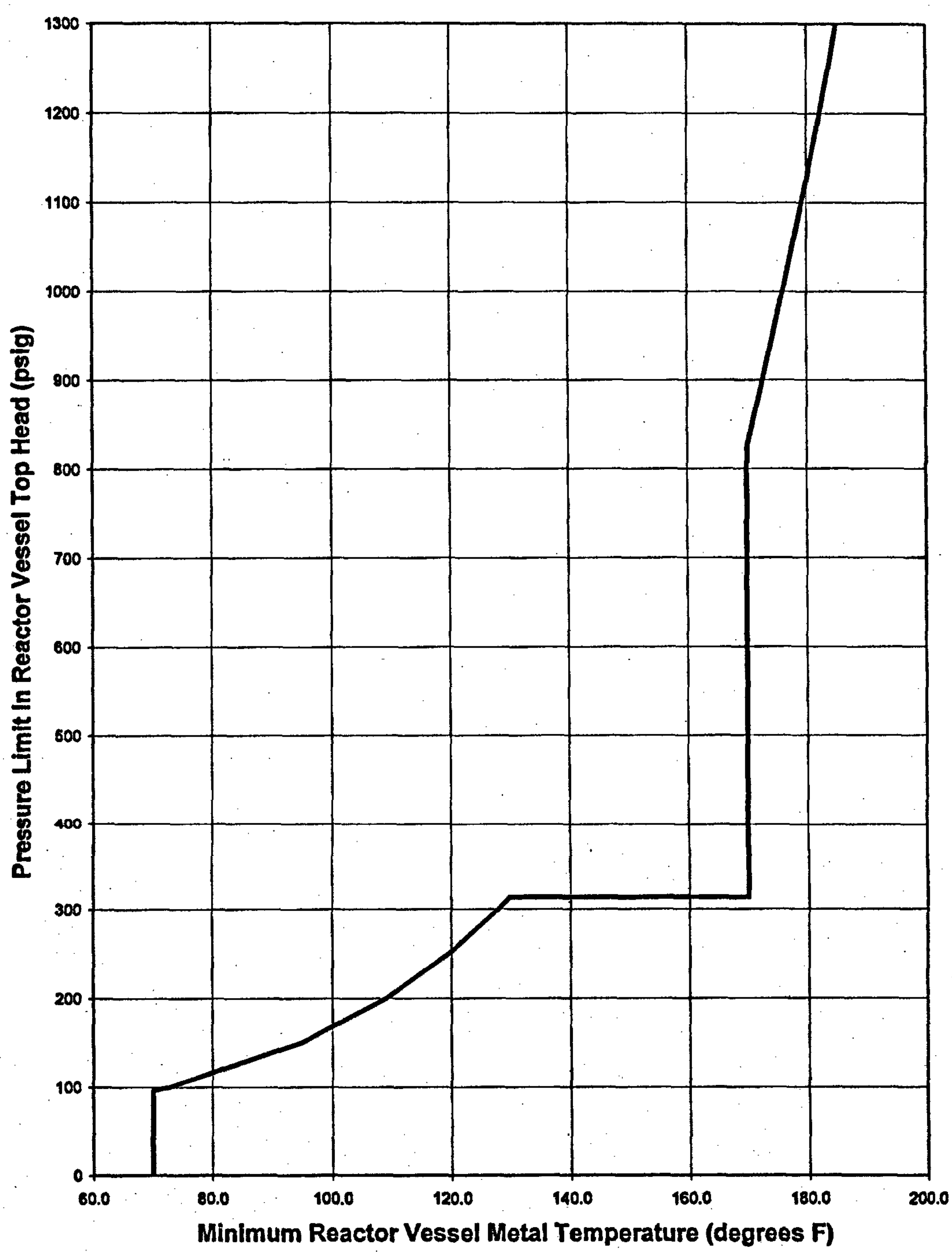


Table 1: SSES Unit 1 Pressure Test (Curve A) P-T Curves

Beltline Region

Revised Pressure-Temperature Curve Calculation
(Pressure Test = Curve A)

Inputs:

Plant =	Susquehanna	
Component =	Beltline	
Vessel thickness, t =	6.1875	inches, so √t = 2.487 √inch
Vessel Radius, R =	126.6875	inches
ART_{NDT} =	61.4	°F ======> 32 EFPY
K_{lt} =	0.0	ksi*inch$^{1/2}$
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)
Safety Factor =	1.5	(for pressure test)
M_m =	2.303	
Temperature Adjustment =	0.0	°F
Pressure Adjustment =	30	psig (hydrostatic pressure for a full vessel)
Hydro Test Pressure =	1,563	psig
Flange RT_{NDT} =	10.0	°F

Fluid Temperature T (°F)	1/4t Temperature (°F)	Kic (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	57.83	38.55	817	70	787
75	75	60.42	40.28	854	75	824
80	80	63.28	42.18	894	80	864
85	85	66.44	44.29	939	85	909
90	90	69.94	46.62	989	90	959
95	95	73.80	49.20	1043	95	1,013
100	100	78.07	52.05	1104	100	1,074
105	105	82.79	55.19	1170	105	1,140
110	110	88.00	58.67	1244	110	1,214
115	115	93.77	62.51	1325	115	1,295
120	120	100.14	66.76	1416	120	1,386

Table 1: SSES Unit 1 Pressure Test (Curve A) P-T Curves (continued)

Feedwater Nozzle/Upper Vessel Region

Inputs:				
Plant =	Susquehanna			
Component =	Upper Vessel	(based on FW nozzle)		
ART_{NDT} =	40.0	°F ======>	All EFPYs	
Vessel thickness, t =	6.5	inches, so √t	2.55	√inch
Vessel Radius, R =	126.7	inches		
F(a/m) =	1.6	nozzle stress factor		
Crack Depth, a =	1.63	inches		
Safety Factor =	1.5			
Temperature Adjustment =	0.0	°F		
Pressure Adjustment =	0.0	psig		
Unit Pressure =	1,563	psig		
Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
-	-	-	-	-	70	312.5
-	-	-	-	-	100	312.5
0	0	42.52	28.34	402	100	402
10	10	44.58	29.72	421	100	421
20	20	47.10	31.40	445	100	445
30	30	50.18	33.45	474	100	474
40	40	53.93	35.96	509	100	509
50	50	58.52	39.02	553	100	553
60	60	64.13	42.75	606	100	606
70	70	70.98	47.32	670	100	670
80	80	79.34	52.90	750	100	750
90	90	89.56	59.71	846	100	846
100	100	102.04	68.03	964	100	964
110	110	117.28	78.19	1108	110	1108
120	120	135.90	90.60	1284	120	1284
130	130	158.63	105.76	1498	130	1498

Table 1: SSES Unit 1 Pressure Test (Curve A) P-T Curves (concluded)

Bottom Head Region

Revised Pressure-Temperature Curve Calculation
(Pressure Test = Curve A)

Inputs:

Plant =	Susquehanna			
Component =	Bottom Head			
Vessel thickness, t =	6.1875	inches, so √t =	2.487	√inch
Vessel Radius, R =	126.6875	inches		
ART_{NDT} =	34.0	°F ======>	32 EFPY	
K_{It} =	0.0	ksi*inch$^{1/2}$		
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)		
Safety Factor =	1.5	(for pressure test)		
Stress Concentration Factor =	3.0	Bottom head penetrations		
M_m =	2.303			
Temperature Adjustment =	0.0	°F		
Height of Water for a Full Vessel =	882.0	inches		
Pressure Adjustment =	31.85	psig (hydrostatic pressure at bottom head for a full vessel at 70°F)		
Hydro Test Pressure =	1,563	psig		
Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Kic (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	75.80	50.53	714	70	682
75	75	80.28	53.52	757	75	725
80	80	85.23	56.82	803	80	771
85	85	90.70	60.47	855	85	823
90	90	96.75	64.50	912	90	880
95	95	103.43	68.95	975	95	943
100	100	110.82	73.88	1044	100	1,012
105	105	118.98	79.32	1121	105	1,089
110	110	128.00	85.33	1206	110	1,174
115	115	137.97	91.98	1300	115	1,268
120	120	148.99	99.33	1404	120	1,372

Table 2: SSES Unit 1 Core Not Critical (Curve B) P-T Curves

Beltline Region

Inputs:

	Plant =	Susquehanna			
	Component =	Beltline			
	Vessel thickness, t =	6.1875	inches, so √t	2.487	√inch
	Vessel Radius, R =	126.6875	inches		
	ART_{NDT} =	61.4	°F ======>	32 EFPY	
	Cooldown Rate =	100.0	°F/hr		
	K_{It} =	9.08	ksi*inch$^{1/2}$		
	Safety Factor =	2.0			
	M_m =	2.303			
	Temperature Adjustment =	0.0	°F		
	Pressure Adjustment =	30.0	psig (hydrostatic pressure for a full vessel)		
	Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Klc (ksi*inch$^{1/2}$)	Klp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	57.83	24.37	517	70	487
75	75.0	60.42	25.67	544	75	514
80	80.0	63.28	27.10	575	80	545
85	85.0	66.44	28.68	608	85	578
90	90.0	69.94	30.43	645	90	615
95	95.0	73.80	32.36	686	95	656
100	100.0	78.07	34.50	731	100	701
105	105.0	82.79	36.86	782	105	752
110	110.0	88.00	39.46	837	110	807
115	115.0	93.77	42.35	898	115	868
120	120.0	100.14	45.53	965	120	935
125	125.0	107.18	49.05	1040	125	1010
130	130.0	114.96	52.94	1123	130	1093
135	135.0	123.56	57.24	1214	135	1184
140	140.0	133.06	61.99	1314	140	1284
145	145.0	143.56	67.24	1426	145	1396

Table 2: SSES Unit 1 Core Not Critical (Curve B) P-T Curves (continued)

Feedwater Nozzle/Upper Vessel Region

Inputs:					
Plant =	Susquehanna				
Component =	Upper Vessel				
ART_{NDT} =	40.0	°F			
σ_{pm} =	20.49	ksi @	1050 psig		
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp	
σ_{sm} =	16.19	ksi @	546 °F		90 °F
σ_{sb} =	19.84	ksi @	546 °F		90 °F
σ_{ys} =	45.0	ksi			
M_m =	2.54				
Safety Factor =	2.0				
$F(a/r_n)$ =	1.6				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	0.0	psig			
Hydro Test Pressure =	1563	psig			
Flange RT_{NDT} =	10.0	°F			

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{sm} (ksi)	σ_{sb} (ksi)	σ_{total} (ksi)	R	Kit (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Total Kic (ksi*inch$^{1/2}$)	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	0.0	70.0	50
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	0.0	70.0	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	64.9	70.0	150
165.9	373.4	3.24	0.03	10.06	11.83	25.16	1.00	45.5	12.7	71.0	70.0	70.0	166
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	79.1	79.1	200
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	89.6	89.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	98.0	98.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	99.9	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	130.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	105.0	130.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	111.0	130.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	116.2	130.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.6	120.9	130.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	125.1	130.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	129.0	130.0	600
614	491.6	11.98	0.13	14.26	16.77	43.13	1.00	64.5	47.1	158.7	130.0	130.0	614
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	132.5	132.5	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	135.2	135.2	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	137.2	137.2	750
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	139.1	139.1	800
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	140.9	140.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	142.7	142.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	144.4	144.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	146.1	146.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	147.7	147.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	149.3	149.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	150.8	150.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	152.2	152.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	153.7	153.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	155.0	155.0	1300

Table 2: SSES Unit 1 Core Not Critical (Curve B) P-T Curves (concluded)

Bottom Head Region

Inputs:

Plant =	Susquehanna			
Component =	Bottom Head	(Penetrations Portion)		
Vessel thickness, t =	6.1875	inches, so √t	2.487	√inch
Vessel Radius, R =	126.6875	inches		
Cooldown Rate =	100.0	°F/hr		
Safety Factor =	2.0			
Stress Concentration Factor =	3.0			
ART_{NDT} =	34.0	°F		
M_m =	2.303			
K_{it} =	9.08	$ksi*inch^{1/2}$		
Temperature Adjustment =	0.00	°F		
Height of full vessel =	882.0	inches		
Pressure Adjustment =	31.85	psig		
Unit Pressure =	1563	psig		
Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	KIc ($ksi*inch^{1/2}$)	KIp ($ksi*inch^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	75.80	33.36	472	70	440
75	75.0	80.28	35.60	503	75	471
80	80.0	85.23	38.08	538	80	506
85	85.0	90.70	40.81	577	85	545
90	90.0	96.75	43.84	620	90	588
95	95.0	103.43	47.18	667	95	635
100	100.0	110.82	50.87	719	100	687
105	105.0	118.98	54.95	777	105	745
110	110.0	128.00	59.46	841	110	809
115	115.0	137.97	64.45	911	115	879
120	120.0	148.99	69.96	989	120	957
125	125.0	161.17	76.05	1075	125	1043
130	130.0	174.63	82.78	1170	130	1138
135	135.0	189.50	90.21	1275	135	1243
140	140.0	205.94	98.43	1391	140	1360

Table 3: SSES Unit 1 Core Critical (Curve C) P-T Curve

Inputs: Plant =	Susquehanna				
Component =	Upper Vessel				
ART_{NDT} =	40.0	°F			
σ_{pm} =	20.49	ksi @	1050 psig		
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp	
σ_{sm} =	16.19	ksi @	546 °F		90 °F
σ_{sb} =	19.04	ksi @	546 °F		90 °F
σ_{ys} =	45.0	ksi			
M_m =	2.64				
Safety Factor =	2.0				
$F(a/r_n)$ =	1.6				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	0.0	psig			
Hydro Test Pressure =	1563	psig			
Flange RT_{NDT} =	10.0	°F			

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{sm} (ksi)	σ_{sb} (ksi)	σ_{total} (ksi)	R	Kit ksi*inch$^{1/2}$	Kip ksi*inch$^{1/2}$	Total Kic ksi*inch$^{1/2}$	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	-	70.0	50
81.4	324.7	1.59	0.02	8.33	9.80	19.73	1.00	37.7	6.2	50.2	30.0	70.0	81
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	42.8	82.8	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	64.9	104.9	150
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	79.1	119.1	200
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	89.6	129.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	98.0	138.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	99.9	139.9	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.8	24.0	101.9	99.9	170.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	105.0	170.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	111.0	170.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	116.2	170.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	120.9	170.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	125.1	170.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	129.0	170.0	600
614	491.6	11.98	0.13	14.26	16.77	43.13	1.00	64.5	47.1	158.7	130.0	170.0	614
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	132.5	172.5	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	135.2	175.2	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	137.2	177.2	750
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	139.1	179.1	800
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	140.9	180.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	142.7	182.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	144.4	184.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	146.1	186.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	147.7	187.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	149.3	189.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	150.8	190.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	152.2	192.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	153.7	193.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	155.0	195.0	1300

Table 4: SSES Unit 2 Pressure Test (Curve A) P-T Curves

Beltline Region

Revised Pressure-Temperature Curve Calculation
(Pressure Test = Curve A)

Inputs:

Plant =	Susquehanna	
Component =	Beltline	
Vessel thickness, t =	6.1875	inches, so $\sqrt{t}$ = 2.487 $\sqrt{\text{inch}}$
Vessel Radius, R =	126.6875	inches
ART_{NDT} =	46.7	°F ======> 32 EFPY
K_R =	0.0	ksi*inch$^{1/2}$
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)
Safety Factor =	1.5	(for pressure test)
M_m =	2.303	
Temperature Adjustment =	0.0	°F
Pressure Adjustment =	30	psig (hydrostatic pressure for a full vessel)
Hydro Test Pressure =	1,563	psig
Flange RT_{NDT} =	10.0	°F

Fluid Temperature T (°F)	1/4t Temperature (°F)	Kic (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	66.24	44.16	936	70	906
75	75	69.72	46.48	986	75	956
80	80	73.56	49.04	1040	80	1,010
85	85	77.80	51.87	1100	85	1,070
90	90	82.49	55.00	1166	90	1,136
95	95	87.68	58.45	1239	95	1,209
100	100	93.41	62.27	1320	100	1,290
105	105	99.74	66.49	1410	105	1,380

Table 4: SSES Unit 2 Pressure Test (Curve A) P-T Curves (continued)

Feedwater Nozzle/Upper Vessel Region

Inputs:				
	Plant =	Susquehanna		
	Component =	Upper Vessel	(based on FW nozzle)	
	ART_{NDT} =	30.0	°F ======>	All EFPYs
	Vessel thickness, t =	6.5	inches, so √t	2.55 √inch
	Vessel Radius, R =	126.7	inches	
	F(a/m) =	1.6	nozzle stress factor	
	Crack Depth, a =	1.63	inches	
	Safety Factor =	1.5		
	Temperature Adjustment =	0.0	°F	
	Pressure Adjustment =	0.0	psig	
	Unit Pressure =	1,563	psig	
	Flange RT_{NDT} =	10.0	°F	

Fluid Temperature T (°F)	1/4t Temperature (°F)	Kic (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
-	-	-	-	-	70	312.5
-	-	-	-	-	100	312.5
0	0	44.58	29.72	421	100	421
10	10	47.10	31.40	445	100	445
20	20	50.18	33.45	474	100	474
30	30	53.93	35.96	509	100	509
40	40	58.52	39.02	553	100	553
50	50	64.13	42.75	606	100	606
60	60	70.98	47.32	670	100	670
70	70	79.34	52.90	750	100	750
80	80	89.56	59.71	846	100	846
90	90	102.04	68.03	964	100	964
100	100	117.28	78.19	1108	100	1108
110	110	135.90	90.60	1284	110	1284
120	120	158.63	105.76	1498	120	1498

Table 4: SSES Unit 2 Pressure Test (Curve A) P-T Curves (concluded)

Bottom Head Region

Revised Pressure-Temperature Curve Calculation
(Pressure Test = Curve A)

Inputs:

Plant =	Susquehanna			
Component =	Bottom Head			
Vessel thickness, t =	6.1875	inches, so √t =	2.487	√inch
Vessel Radius, R =	126.6875	inches		
ART_{NDT} =	24.0	°F ======>	32 EFPY	
K_{it} =	0.0	$ksi*inch^{1/2}$		
$\Delta T_{1/4t}$ =	0.0	°F (no thermal for pressure test)		
Safety Factor =	1.5	(for pressure test)		
Stress Concentration Factor =	3.0	Bottom head penetrations		
M_m =	2.303			
Temperature Adjustment =	0.0	°F		
Height of Water for a Full Vessel =	882.0	inches		
Pressure Adjustment =	31.85	psig (hydrostatic pressure at bottom head for a full vessel at 70°F)		
Hydro Test Pressure =	1,563	psig		
Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Kic ($ksi*inch^{1/2}$)	Kip ($ksi*inch^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70	85.23	56.82	803	70	771
75	75	90.70	60.47	855	75	823
80	80	96.75	64.50	912	80	880
85	85	103.43	68.95	975	85	943
90	90	110.82	73.88	1044	90	1,012
95	95	118.98	79.32	1121	95	1,089
100	100	128.00	85.33	1206	100	1,174
105	105	137.97	91.98	1300	105	1,268
110	110	148.99	99.33	1404	110	1,372

Table 5: SSES Unit 2 Core Not Critical (Curve B) P-T Curves

Beltline Region

Inputs:

Plant =	Susquehanna				
Component =	Beltline				
Vessel thickness, t =	6.1875	inches, so √t	2.487	√inch	
Vessel Radius, R =	126.6875	inches			
ART_{NDT} =	46.7	°F ======>	32 EFPY		
Cooldown Rate =	100.0	°F/hr			
K_{It} =	9.08	ksi*inch$^{1/2}$			
Safety Factor =	2.0				
M_m =	2.303				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	30.0	psig (hydrostatic pressure for a full vessel)			
Flange RT_{NDT} =	10.0	°F			

Fluid Temperature T (°F)	1/4t Temperature (°F)	KIc (ksi*inch$^{1/2}$)	KIp (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	66.24	28.58	606	70	576
75	75.0	69.72	30.32	643	75	613
80	80.0	73.56	32.24	684	80	654
85	85.0	77.80	34.36	729	85	699
90	90.0	82.49	36.71	778	90	748
95	95.0	87.68	39.30	833	95	803
100	100.0	93.41	42.17	894	100	864
105	105.0	99.74	45.33	961	105	931
110	110.0	106.74	48.83	1035	110	1005
115	115.0	114.47	52.70	1117	115	1087
120	120.0	123.02	56.97	1208	120	1178
125	125.0	132.46	61.69	1308	125	1278
130	130.0	142.90	66.91	1419	130	1389

Table 5: SSES Unit 2 Core Not Critical (Curve B) P-T Curves (continued)

Feedwater Nozzle/Upper Vessel Region

Inputs:					
Plant =	Susquehanna				
Component =	Upper Vessel				
ART_{NDT} =	30.0	°F			
σ_{pm} =	20.49	ksi @	1050 psig		
σ_{pb} =	6.22	ksi @	1050 psig	Base Temp	
σ_{sm} =	16.19	ksi @	546 °F		90 °F
σ_{sb} =	19.04	ksi @	546 °F		90 °F
σ_{ys} =	45.0	ksi			
M_m =	2.54				
Safety Factor =	2.0				
$F(a/r_n)$ =	1.6				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	0.0	psig			
Hydro Test Pressure =	1563	psig			
Flange RT_{NDT} =	10.0	°F			

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{sm} (ksi)	σ_{sb} (ksi)	σ_{total} (ksi)	R	Kit (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Total Kic (ksi*inch$^{1/2}$)	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	0.0	70.0	50
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	0.0	70.0	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	54.9	70.0	150
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	69.1	70.0	200
203.7	389.4	3.98	0.04	10.63	12.50	27.15	1.00	48.1	15.6	79.3	70.0	70.0	204
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	79.6	79.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	88.0	88.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	130.0	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	130.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	95.0	130.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	101.0	130.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	106.2	130.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	110.9	130.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	115.1	130.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	119.0	130.0	600
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	122.5	130.0	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	125.2	130.0	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	178.0	127.2	130.0	750
757	514.3	14.77	0.16	15.06	17.71	47.71	0.92	62.7	58.0	178.7	127.4	130.0	757
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	129.1	130.0	800
825	523.9	16.10	0.17	15.41	18.12	49.79	0.86	60.1	63.2	186.5	130.0	130.0	825
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	130.9	130.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	132.7	132.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	134.4	134.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	136.1	136.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	137.7	137.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	139.3	139.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	140.8	140.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	142.2	142.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	143.7	143.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	145.0	145.0	1300

Table 5: SSES Unit 2 Core Not Critical (Curve B) P-T Curves (concluded)

Bottom Head Region

Inputs:

Plant =	Susquehanna			
Component =	Bottom Head	(Penetrations Portion)		
Vessel thickness, t =	6.1875	inches, so √t	2.487	√inch
Vessel Radius, R =	126.6875	inches		
Cooldown Rate =	100.0	°F/hr		
Safety Factor =	2.0			
Stress Concentration Factor =	3.0			
ART_{NDT} =	24.0	°F		
M_m =	2.303			
K_{lt} =	9.08	ksi*inch$^{1/2}$		
Temperature Adjustment =	0.00	°F		
Height of full vessel =	882.0	inches		
Pressure Adjustment =	31.85	psig		
Unit Pressure =	1563	psig		
Flange RT_{NDT} =	10.0	°F		

Fluid Temperature T (°F)	1/4t Temperature (°F)	Kic (ksi*inch$^{1/2}$)	Kip (ksi*inch$^{1/2}$)	Calculated Pressure P (psig)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	70	0
70	70.0	85.23	36.08	538	70	506
75	75.0	90.70	40.81	577	75	545
80	80.0	96.75	43.84	620	80	588
85	85.0	103.43	47.18	667	85	635
90	90.0	110.82	50.87	719	90	687
95	95.0	118.98	54.95	777	95	745
100	100.0	128.00	59.46	841	100	809
105	105.0	137.97	64.45	911	105	879
110	110.0	148.99	69.96	989	110	957
115	115.0	161.17	76.05	1075	115	1043
120	120.0	174.63	82.78	1170	120	1138
125	125.0	189.50	90.21	1275	125	1243
130	130.0	205.94	98.43	1391	130	1360

Table 6: SSES Unit 2 Core Critical (Curve C) P-T Curve

Inputs:					
Plant =	Susquehanna				
Component =	Upper Vessel				
ART_{NDT} =	30.0	°F			
σ_{pm} =	20.49	ksi @	1050 psig		
σ_{pb} =	0.22	ksi @	1050 psig	Base Temp	
σ_{sm} =	16.19	ksi @	546 °F		90 °F
σ_{sb} =	19.04	ksi @	546 °F		90 °F
σ_{ys} =	45.0	ksi			
M_m =	2.64				
Safety Factor =	2.0				
$F(a/r_n)$ =	1.6				
Temperature Adjustment =	0.0	°F			
Pressure Adjustment =	0.0	psig			
Hydro Test Pressure =	1563	psig			
Flange RT_{NDT} =	10.0	°F			

Pressure P (psig)	Saturation Temperature (°F)	σ_{pm} (ksi)	σ_{pb} (ksi)	σ_{sm} (ksi)	σ_{sb} (ksi)	σ_{total} (ksi)	R	Kit ksi*inch$^{1/2}$	Kip ksi*inch$^{1/2}$	Total Kic ksi*inch$^{1/2}$	Calculated Temperature T (°F)	Adjusted Temperature for P-T Curve (°F)	Adjusted Pressure for P-T Curve (psig)
-	-	-	-	-	-	-	-	-	-	-	-	70.0	0
50	297.3	0.98	0.01	7.36	8.65	17.00	1.00	33.3	3.8	41.0	-	70.0	50
95.5	334.7	1.86	0.02	8.69	10.22	20.79	1.00	39.3	7.3	54.0	30.0	70.0	96
100	337.7	1.95	0.02	8.79	10.34	21.11	1.00	39.8	7.7	55.1	32.8	72.8	100
150	365.8	2.93	0.03	9.79	11.52	24.27	1.00	44.3	11.5	67.3	54.9	94.9	150
200	387.9	3.90	0.04	10.58	12.44	26.96	1.00	47.9	15.3	78.5	69.1	109.1	200
250	406.2	4.88	0.05	11.23	13.20	29.36	1.00	50.8	19.2	89.1	79.6	119.6	250
300	422.1	5.85	0.06	11.79	13.86	31.57	1.00	53.3	23.0	99.4	88.0	128.0	300
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	129.9	312.5
312.5	425.7	6.10	0.07	11.92	14.02	32.10	1.00	53.9	24.0	101.9	89.9	170.0	312.5
350	436.0	6.83	0.07	12.28	14.45	33.63	1.00	55.6	26.8	109.3	95.0	170.0	350
400	448.5	7.81	0.08	12.73	14.97	35.59	1.00	57.6	30.7	118.9	101.0	170.0	400
450	459.9	8.78	0.09	13.13	15.45	37.45	1.00	59.4	34.5	128.4	106.2	170.0	450
500	470.4	9.76	0.10	13.51	15.88	39.25	1.00	61.1	38.3	137.8	110.9	170.0	500
550	480.1	10.73	0.12	13.85	16.29	40.99	1.00	62.7	42.2	147.0	115.1	170.0	550
600	489.1	11.71	0.13	14.17	16.67	42.67	1.00	64.1	46.0	156.1	119.0	170.0	600
650	497.6	12.68	0.14	14.47	17.02	44.31	1.00	65.5	49.9	165.2	122.5	170.0	650
700	505.6	13.66	0.15	14.76	17.35	45.92	0.97	64.9	53.7	172.3	125.2	170.0	700
750	513.2	14.64	0.16	15.03	17.67	47.49	0.93	63.0	57.5	176.0	127.2	170.0	750
757	514.3	14.77	0.16	15.06	17.71	47.71	0.92	62.7	58.0	178.7	127.4	170.0	757
800	520.4	15.61	0.17	15.28	17.97	49.03	0.88	61.1	61.3	183.6	129.1	170.0	800
825	523.9	16.10	0.17	15.41	18.12	49.79	0.86	60.1	63.2	186.5	130.0	170.0	825
850	527.3	16.59	0.18	15.53	18.26	50.55	0.84	59.1	65.1	189.3	130.9	170.9	850
900	533.9	17.56	0.19	15.76	18.53	52.04	0.80	57.2	68.9	195.0	132.7	172.7	900
950	540.1	18.54	0.20	15.98	18.80	53.52	0.76	55.3	72.7	200.6	134.4	174.4	950
1000	546.2	19.51	0.21	16.20	19.05	54.97	0.73	53.4	76.5	206.3	136.1	176.1	1000
1050	552.0	20.49	0.22	16.40	19.29	56.40	0.69	51.4	80.3	212.0	137.7	177.7	1050
1100	557.6	21.47	0.23	16.60	19.52	57.82	0.66	49.5	84.1	217.6	139.3	179.3	1100
1150	563.0	22.44	0.24	16.79	19.75	59.23	0.63	47.6	87.8	223.3	140.8	180.8	1150
1200	568.2	23.42	0.25	16.98	19.97	60.62	0.59	45.6	91.6	228.9	142.2	182.2	1200
1250	573.3	24.39	0.26	17.16	20.18	61.99	0.56	43.7	95.4	234.6	143.7	183.7	1250
1300	578.2	25.37	0.27	17.33	20.38	63.36	0.53	41.8	99.2	240.2	145.0	185.0	1300

Appendix A

SSES Reactor Vessel Material Surveillance Programs

SSES Unit 1:

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, the first surveillance capsule was removed from the SSES Unit 1 reactor vessel on March 27, 1992 as a part of refueling outage number 6 (RFO6) activities. The surveillance capsule contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The flux wires and test specimens removed from the capsule were tested according to ASTM E185-82. The methods and results of testing are presented in Reference 6.5, as required by 10 CFR 50, Appendices G and H.

As described in the SSES Unit 1 Updated Final Safety Analysis Report (UFSAR) Section 5.3.1.6, Material Surveillance, and UFSAR Table 5.3-3, the second surveillance capsule is slated to be removed at 15 EFPY.

SSES Unit 2:

In accordance with 10 CFR 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, the first surveillance capsule was removed from the SSES Unit 2 reactor vessel on September 28, 1992 as a part of refueling outage number 5 (RFO5) activities. The surveillance capsule contained flux wires for neutron fluence measurement, Charpy V-Notch impact test specimens and uniaxial tensile test specimens fabricated using materials from the vessel materials within the core beltline region. The flux wires and test specimens removed from the capsule were tested according to ASTM E185-82. The methods and results of testing are presented in Reference 6.6, as required by 10 CFR 50, Appendices G and H.

As described in the SSES Unit 2 Updated Final Safety Analysis Report (UFSAR) Section 5.3.1.6, Material Surveillance, and UFSAR Table 5.3-3, the second surveillance capsule is slated to be removed at 15 EFPY.