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Project No.: GE-10Q
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April 2007

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:

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REVISION CONTROL SHEET

Document Number: SIR-05-044-A

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
---	i – xxv	0	04/11/07	Initial issue of "–A" version of report that incorporates NRC RAIs and Final SER.
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Appendix C	C-1 – C-22			

February 6, 2007

Mr. Randy C. Bunt
Chair, BWR Owners Group
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SUBJECT: FINAL SAFETY EVALUATION FOR THE BOILING WATER REACTOR OWNERS' GROUP (BWROG) STRUCTURAL INTEGRITY ASSOCIATES TOPICAL REPORT (TR) SIR-05-044, "PRESSURE TEMPERATURE REPORT METHODOLOGY FOR BOILING WATER REACTORS" (TAC NO. MC9694)

Dear Mr. Bunt:

By letter dated December 20, 2005, and supplement dated August 29, 2006, the BWROG submitted TR SIR-05-044, "Pressure Temperature Report Methodology for Boiling Water Reactors," Revision 0 to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated November 14, 2006, an NRC draft safety evaluation (SE) regarding our approval of SIR-05-044 was provided for your review and comments. By letter dated December 20, 2006, the BWROG commented on the draft SE. The NRC staff's disposition of BWROG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that SIR-05-044 is acceptable for referencing in licensing applications for General Electric-designed boiling water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that the BWROG publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the BWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: Final SE

cc w/encl: See next page

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ADAMS ACCESSION NO.: **ML070180483** *No major changes to SE input. NRR-043

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT (LTR) SIR-05-044

"PRESSURE TEMPERATURE REPORT METHODOLOGY FOR BOILING WATER

REACTORS," REVISION 0

BOILING WATER REACTORS OWNERS' GROUP (BWROG)

PROJECT NO. 691

1.0 INTRODUCTION

In a letter dated December 20, 2005, the Boiling Water Reactor Owners' Group (BWROG) submitted LTR SIR-05-044, "Pressure Temperature Limits Report Methodology for Boiling Water Reactors", Revision 0, dated December 2005 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML053560336) to the U.S. Nuclear Regulatory Commission (NRC) for review and acceptance for referencing in subsequent licensing actions. The BWROG provided this LTR to support the ability of boiling water reactor (BWR) licensees to relocate their pressure-temperature (P/T) curves and associated numerical values (such as heatup/cooldown rates) from facility Technical Specifications (TS) to a Pressure Temperature Limits Report (PTLR), a licensee-controlled document, using the guidelines provided in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 1). Proposed revisions to this LTR and responses to NRC staff requests for additional information (RAIs) were provided in letter from the BWROG dated August 29, 2006 (ADAMS Accession No. ML062440387).

2.0 REGULATORY EVALUATION

2.1 Requirements for Generating P/T Limits for Light-Water Reactors

The NRC has established requirements in Appendix G of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix G; Reference 2), in order to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The regulation at 10 CFR Part 50, Appendix G requires that the P/T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Reference 3, ASME Code, Section XI, Appendix G) were used to generate the P/T limits. The regulation at 10 CFR Part 50, Appendix G, also requires that applicable surveillance data from reactor pressure vessel (RPV) material surveillance

ENCLOSURE

programs be incorporated into the calculations of plant-specific P/T limits, and that the P/T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials.

Table 1 to 10 CFR Part 50, Appendix G provides the NRC staff's criteria for meeting the P/T limit requirements of ASME Code, Section XI, Appendix G, as well as the minimum temperature requirements of the rule for bolting up the vessel during normal and pressure testing operations. In addition, the NRC staff regulatory guidance related to P/T limit curves is found in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (Reference 4), and Standard Review Plan Chapter 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock" (Reference 5).

The regulation at 10 CFR Part 50, Appendix H (Reference 6), provides the NRC staff's criteria for the design and implementation of RPV material surveillance programs for operating light-water reactors.

In March 2001, the NRC staff issued RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 7). Fluence calculations are acceptable if they are done with approved methodologies or with methods which are shown to conform to the guidance in RG 1.190.

2.2 Technical Specification Requirements for P/T Limits

Section 182a of the Atomic Energy Act of 1954 requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36 (Reference 8). That regulation requires that the TS include items in five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.

The regulation at 10 CFR 50.36(c)(2)(ii) requires that LCOs be established for the P/T limits, because the parameters fall within the scope of the Criterion 2 identified in the rule:

> A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

The P/T limits for BWR-designed light-water reactors fall within the scope of Criterion 2 of 10 CFR 50.36(c)(2)(ii) and are therefore ordinarily required to be included within the TS LCOs for a plant-specific facility operating license. On January 31, 1996, the NRC staff issued GL 96-03 to inform licensees that they may request a license amendment to relocate the P/T limit curves from the TS LCOs into a PTLR or other licensee-controlled document that would be controlled through the Administrative Controls Section of the TS. In GL 96-03, the NRC staff informed licensees that in order to implement a PTLR, the P/T limits for light-water reactors would need to be generated in accordance with an NRC-approved methodology and that the methodology to generate the P/T limits would need to comply with the requirements of 10 CFR Part 50, Appendices G and H; be documented in an NRC-approved topical report or plant-specific submittal; and be incorporated by reference in the Administrative Controls Section of the TS. The GL also mandated that the TS Administrative Controls Section would need to

reference the NRC staff's safety evaluation (SE) issued on the PTLR request and that the PTLR be defined in Section 1.0 of the TS. Attachment 1 to GL 96-03 provided a list of the criteria that the approved methodology and PTLR would be required to meet.

TS Task Force (TSTF) Traveler No. TSTF-419 (Reference 9) amended the Standard TS (STS) (NUREGs-1430, -1431, -1432, -1433, and -1434) to: (1) delete references to the TS LCO specifications for the P/T limits in the TS definition of the PTLR, and (2) revise STS 5.6.6 to identify, by number and title, NRC-approved topical reports that document PTLR methodologies, or the NRC safety evaluation for a plant-specific methodology by NRC letter and date. A requirement was added to the reviewers note to specify the complete citation of the PTLR methodology in the plant-specific PTLR, including the report number, title, revision, date, and any supplements. Only the figures, values, and parameters associated with the P/T limits are relocated to the PTLR. The methodology for their development must be reviewed and approved by the NRC staff. TSTF-419 did not change the requirements associated with the review and approval of the methodology or the requirement to operate within the limits specified in the PTLR. Any changes to a methodology that had not been approved by the NRC staff would continue to require NRC staff review and approval pursuant to the license amendment request provisions and requirements of 10 CFR 50.90 (Reference 10).

3.0 TECHNICAL EVALUATION

As discussed in Section 2.1 of this SE, 10 CFR Part 50, Appendix G, requires licensees to establish limits on the pressure and temperature of the RCPB in order to protect the RCPB against brittle failure (i.e., against brittle "fast-fracture"). These limits are defined by P/T limit curves for normal operations (including heatup and cooldown operations of the reactor coolant system (RCS), normal operation of the RCS with the reactor being in a critical condition, and transient operating conditions) and during pressure testing conditions (i.e., either inservice leak rate testing and/or hydrostatic testing conditions).

The BWROG LTR SIR-05-44 was prepared by Structural Integrity Associates and has three sections and two appendices. Section 1.0 describes the background and purpose for the LTR. Section 2.0 provides the fracture mechanics methodology and its basis for developing P/T limits. Section 3.0 provides a step-by-step procedure for calculating P/T limits. Appendix A provides guidance for evaluating surveillance data. Appendix B provides a template PTLR.

3.1 Evaluation of Section 2.0 of the LTR

Section 2.0 provides the fracture mechanics methodology and its basis for developing P/T limits. The NRC staff evaluation of this section is based on the criteria contained in Attachment 1 of GL 96-03. Attachment 1 of GL 96-03 contains seven technical criteria that the contents of proposed methodology should conform to if license amendments requesting PTLRs are to be approved by the NRC staff. The NRC staff's evaluations of the contents of the BWROG methodology against the seven criteria in Attachment 1 of GL 96-03 are provided in the subsections that follow.

GL 96-03, Attachment 1, Criterion 1:

Criterion 1 requires that the methodology describe the transport calculation methods including computer codes and formulas used to calculate neutron fluences.

Table 1-1 of the BWROG RAI responses, dated August 29, 2006, indicates that this LTR does not describe the transport calculation methods including computer codes and formulas used to calculate neutron fluences. It indicates that fluence methods and results must comply with RG 1.190 and have NRC staff approval for use with this LTR. Table 1-1 will be included in the LTR. Therefore, as stated in the LTR this will be a plant-specific action item to be addressed by licensees. Since Table 1-1 in the proposed LTR methodology will indicate that the fluence methods and results must comply with RG 1.190 and have NRC staff approval, this criterion has been satisfied.

GL 96-03, Attachment 1, Criterion 2:

Criterion 2 requires that the methodology describe the surveillance program and indicates that the topical report should contain a place holder for the requested information.

Table 1-1 of the BWROG RAI responses, dated August 29, 2006, indicates that this information is in Appendix A of the template PTLR, which is in Appendix B of the LTR. Therefore, as stated in the LTR this will be a plant-specific action item to be addressed by licensees. Since Table 1-1 indicates that the information will be included in the PTLR, this criterion has been satisfied.

GL 96-03, Attachment 1, Criterion 3:

Criterion 3 requires that the methodology describe how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics.

This LTR does not need to address this criterion since Criterion 3 only applies to pressurized water reactors (PWRs) and this LTR applies to BWRs.

GL 96-03, Attachment 1, Criterion 4:

Criterion 4 requires that the methodology describe the method for calculating the Adjusted Reference Temperature (ART) using RG 1.99, Revision 2.

Table 1-1 of the BWROG RAI responses, dated August 29, 2006, indicates that this information is in Section 2.3 of the LTR. Section 2.3 of the LTR describes the methodology documented in RG 1.99, Revision 2, for calculating ART. Therefore this criterion has been satisfied.

GL 96-03, Attachment 1, Criterion 5:

Criterion 5 requires that the methodology describe the application of fracture mechanics in the construction of P/T curves based on ASME Code, Section XI, Appendix G, and SRP, Section 5.3.2.

Table 1-1 of the BWROG RAI responses, dated August 29, 2006, indicates that this information is in Sections 2.3 and 2.4 of the LTR. However, the information is in Sections 2.4 and 2.5 of the LTR (Table 1-1 needs to be revised to include Section 2.5). Section 2.4 describes the general fracture mechanics methodology for calculating P/T limit curves. Section 2.5 describes the methodology for calculating P/T limits for the RPV beltline, bottom head region, and non-beltline region. The non-beltline region includes all regions outside the beltline, excluding the bottom head.

Section 2.4 provides fracture mechanics criteria based on ASME Code, Section XI, Appendix G, and ASME Code Cases N–640 and N–641. These code cases allow the use of the reference stress intensity factor, K_{Ic}, for calculating P/T limit curves. NRC Regulatory Issue Summary 2004-04 (Reference 11) indicates that the use of NRC-approved ASME Code cases in conjunction with earlier versions of the ASME Code endorsed in 10 CFR 50.55a may also be used for development of P/T limit curves without the need for an exemption. NRC RG 1.147, Revision 14 (Reference 12) approves these ASME Code cases. The use of the reference stress intensity factor, K_{Ic}, for calculating P/T limit curves was first endorsed by the 1999 Addenda of the ASME Code. Therefore, licensees utilizing this methodology and versions of ASME Code, Section XI, Appendix G, that require P/T limit curves to be calculated using K_{Ic} do not need to request an exemption.

Section 2.5 describes the methodology for calculating P/T limits for the RPV beltline, bottom head region, and non-beltline region. For the beltline shell region, this section describes three methods for calculating the thermal stress intensity factor, K_{It}: a) a stress linearizing technique presented in ASME Code, Section XI, Nonmandatory Appendix A; b) a technique based on Section XI, Appendix G; and c) a technique based on Welding Research Council (WRC) Bulletin Number 175 (Reference 13). In response to NRC staff RAI No. 3, the BWROG changed the stress linearizing technique to the method in ASME Code, Section XI, Appendix G. The allowable pressure is calculated using the methodology and structural factors in ASME Code, Section XI, Appendix G. Since these techniques are based on methodologies endorsed by the NRC, they are acceptable. The NRC staff requires that this change be incorporated into the -A version of the LTR.

The LTR indicates that the methodology for calculating bottom head P/T limit curves should follow the methodologies for the shell region except that a stress concentration factor is applied to bottom head membrane stresses to account for the stress concentration resulting from nozzles in the lower head. In addition, the pressure stress is considered entirely as a membrane stress. Appendix 5 in WRC Bulletin Number 175 describes methods for calculating the stress intensity factors at the inside corner of a nozzle. The stress concentration factors described in these analyses are less than those utilized by the BWROG for the development of bottom head P/T limits. The methodology proposed by the BWROG for the bottom head has been previously reviewed by the NRC staff in a letter from D. S. Collins (NRC) to R. G. Byram (Senior Vice President and Chief Nuclear Officer for Susquehanna Steam Electric Station, Units 1 and 2) dated February 7, 2002 (ADAMS Accession No. ML013520605). The NRC staff performed independent calculations and concluded that the method is consistent with the methods in the 1995 Edition of Appendix G to Section XI of the ASME Code. Based on the use of a conservative concentration factor and the NRC staff's previous evaluation of this methodology, the NRC staff concludes that the methodology proposed by the BWROG for the calculating bottom head P/T limit curves is acceptable.

The non-beltline region analysis method that was contained in Section 2.5 has been deleted and replaced with a methodology that is described in the BWROG response to RAI No. 3. In this methodology the location to be analyzed for determining the highest stresses in the non-beltline region is the feedwater nozzle. The reference temperature, RT_{NDT}, used in the analysis is the limiting value for all non-beltline nozzles. The stress intensity factors for the feedwater nozzle may be calculated from a detailed finite element model analysis of the nozzle. The stress distribution from the finite element analysis is fit with a third order polynomial. The stress intensity factors for the nozzle corner use the coefficients from the stress distribution polynomial and a method proposed in General Electric (GE) Topical Report NEDE-21821-02 (Reference 14) for calculating stress intensity factors for nozzle corner cracks. The stress intensity factor solutions documented in Reference 14 were verified by independent analysis and experiment. Reference 14 was approved by the NRC staff in a letter from D. G. Eisenhut (NRC) to R. Gridley (GE) dated January 14, 1980 (ADAMS Legacy Library Accession No. 8002070141). The NRC staff concluded that each step in the GE analysis is acceptable, but had specific comments. Since none of the comments were directed at the stress intensity solutions for the nozzle corner crack, the stress intensity solutions proposed were considered acceptable for evaluating nozzle corner cracks. The proposed methodology uses the stress intensity factors from both thermal and pressure stress to develop P/T limits based on the structural factors described in Appendix G to Section XI of the ASME Code. The NRC staff finds the non-beltline methodology acceptable since it meets the requirements of ASME Code, Section XI, Appendix G and the stress intensity factors are determined using a previously approved methodology. However, the NRC staff requires that the information provided in response to RAI No. 3 be incorporated into the -A version of the LTR.

Section 2.5 of the LTR and the methodology proposed in response to RAI No. 3 to describe methodologies for calculating bending and membrane stresses using computer code finite element analyses. If these finite element analyses are to be utilized by licensees to develop P/T limits, the NRC staff requested, in RAI No. 2, that the BWROG provide the following:

a) Identify the computer codes that were used in the finite element stress analysis. How were the codes benchmarked?

b) Discuss briefly the assumptions [initial RT_{NDT}] and the inputs to the stress analysis.

c) Update the topical report methodology to require licensees to identify the finite element codes used in the PTLR.

d) Verify that this process for determining bending and membrane stresses will result in the generation of P/T limits that are at least as conservative as those which would be generated using the procedures of ASME Code, Section XI, Appendix G.

In response to the NRC staff request to items a), b), and c), the BWROG proposed to add the following text to the Section 2.5 of the LTR:

> In the subsections that follow, finite element analysis is discussed as a possible approach for providing the necessary stress analysis for RPV regions. If finite element analysis is utilized to develop P-T limits for any RPV region, the following information shall be provided in the PTLR:

a. Identify the computer code(s) that were used in the finite element stress analysis.

b. For any computer codes used, describe how the code(s) were verified or benchmarked. Computer code verification shall be in accordance with a qualified 10 CFR 50 Appendix B Quality Assurance Program. As a part of computer code verification, benchmarking consistent with NRC GL 83-11, Supplement 1 [17] shall be included.

c. Identify the assumptions and the inputs to the finite element analysis. Necessary inputs to the analysis include any or all of the following:

- A description of plant operating conditions used (e.g., pressure and temperature). The conditions used must represent current plant operating conditions.
- A description of the heat transfer coefficients used and the methodology used to calculate them.
- A description of the model developed, including materials, material properties, finite element mesh pattern, and geometry.

New Reference 17 will be added to Section 4.0 of the LTR as follows:

17. U. S. Nuclear Regulatory Commission, Generic Letter 88-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.

Since the LTR will require licensees to provide the requested information in the PTLR, the response is acceptable.

For item d), the BWROG proposed that the linearization techniques discussed in the LTR be removed and replaced with polynomial fit techniques that are consistent with the current ASME Code, Section XI, Appendix G, methodology. The proposed technique is described in the BWROG response to RAI No. 3. Since the linearization technique will be replaced with a technique which is consistent with the current ASME Code, Section XI, Appendix G, methodology, the change is acceptable. Since Sections 2.4 and 2.5 identify fracture mechanics methods for the construction of P/T curves based on ASME Code, Section XI, Appendix G, this criterion has been satisfied.

GL 96-03, Attachment 1, Criterion 6:

Criterion 6 requires that the methodology describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T curves for boltup temperature and hydrotest temperature.

Table 1-1 of the BWROG RAI responses, dated August 29, 2006, indicates that this information is in Sections 2.7 and 2.8 of the LTR. Section 2.7 identifies the minimum metal temperature of the RPV closure head flange and the RPV shell flange regions. This section also describes the

minimum requirements for hydrotest (hydrostatic pressure and leak tests). Section 2.8 identifies the minimum boltup temperatures. Both of these sections identify minimum temperature requirements that are contained in Appendix G to 10 CFR Part 50. Since the information in these sections comply with the requirements in Appendix G to 10 CFR Part 50, this criterion has been satisfied.

GL96-03, Attachment 1, Criterion 7:

Criterion 7 requires that the methodology describe how the data from multiple surveillance capsules are used in the ART calculation.

Table 1-1 of the BWROG RAI responses, dated August 29, 2006, indicates that this information is in Sections 2.3 of the LTR. Criteria for evaluating surveillance data are contained in Appendix A to the LTR. (Table 1-1 needs to be revised to include Appendix A when it is added to the -A version of the LTR). Appendix A documents two procedures for calculating the ART. One procedure is applicable when RPV material and surveillance material have identical heat numbers. This method follows the methodology documented in Position 2.1 of RG 1.99, Revision 2 and the NRC staff guidance presented in an NRC/Industry Workshop (Reference 15). Position 2.1 in RG 1.99, Revision 2 contains NRC staff guidance for evaluating surveillance data when there are two or more credible surveillance data points. Credibility is determined by following the guidance in RG 1.99, Revision 2.

The second procedure is applicable when the heat number for the surveillance material does not match the heat number for the RPV material. In this case the ART is determined using the guidance in Position 1.1 of RG 1.99, Revision 2. Position 1.1 in RG 1.99, Revision 2 contains NRC staff guidance for determining the ART from the chemical composition (weight-percent copper and nickel) of the RPV material.

The NRC staff recommended changes to these procedures in RAIs sent to the BWROG. These changes are discussed in the evaluation of Appendix A, which is discussed in Section 3.3 of this SE. The changes to Appendix A are acceptable, because they provide additional guidance to the licensees and the guidance has been previously approved by the NRC staff. Based on the changes documented in Section 3.3 and that the procedures follow guidance recommended by the NRC staff, this criterion has been satisfied.

3.2 Evaluation of Section 3.0 of the LTR

Section 3.0 of the LTR provides a step-by-step procedure for calculating P/T limit curves. This section indicates that P/T limits may also be developed for other RPV regions to provide additional operating flexibility. In response to RAI No. 5, the BWROG indicated that a sentence in the LTR will be revised to state:

> P-T limit curves may also be developed for other RPV regions to provide additional operating flexibility; however, for RPV regions other than those defined in Section 2.0 of this report, licensees are required to submit methodologies to the NRC for review and approval prior to use.

Since methods of evaluating other regions will be submitted to the NRC for review and approval prior to use, the proposed change is acceptable. The NRC staff requires that this modification be incorportated into the -A version of the LTR.

The guidance given in Section 3.0 does not indicate surveillance data is to be evaluated. In response to RAI No. 6, the BWROG indicated a new Step (a) will be added to Section 3.0 of the LTR and the previously defined steps will be re-labeled as Steps (b) through (i). The proposed new Step (a) follows:

a. Evaluate surveillance data in accordance with Appendix A of this report.

Appendix A provides guidance for the use of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) surveillance data. The BWRVIP ISP replaces individual plant RPV surveillance capsule programs with representative weld and base materials data from host reactors. 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) 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 RPV weld and base materials. Plant owners, in turn are responsible to evaluate the data using the methods in RG 1.99, Revision 2, in accordance with 10 CFR Part 50, Appendix G, for determination of ART values.

Since the LTR will be revised to indicate surveillance data is to be evaluated in accordance with Appendix A and Appendix A contains criteria for processing and reporting surveillance data, the proposed change is acceptable. The NRC staff requires that this change be incorporated into the -A version of the LTR.

3.3 Evaluation of Appendix A of the LTR

Appendix A provides guidance for evaluating surveillance data. In response to NRC staff RAI No. 7, Appendix A will be revised to identify the source for the best estimate chemistries for the BWR vessel and surveillance capsule materials and to identify that the best estimate chemistries will be documented in the PTLR. The BWROG response adds the following note and reference to Appendix A:

> Note: Revised best estimate chemistries for selected BWR vessel and surveillance capsule materials have been calculated by the BWRVIP, as documented in BWRVIP-86-A [A-1]. Calculation of the best estimate chemistries for all other vessel materials should be determined in accordance with the NRC practice documented in Reference [A-7]. The suggested practice is documented in guidelines contained in BWRVIP-135. This evaluation is the responsibility of the plant, must be described in the PTLR, and must utilize NRC-approved methods.

New Reference A-7 will be added to Section A.5 of the LTR as follows:

A-7. "Generic Letter 92-01 and RPV Integrity Assessment - Status, Schedule, and Issues," Presentation by K. Wichman, M. Mitchell, and A. Hiser at NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998.

In response to NRC staff RAI No. 8, Appendix A will be revised to describe the temperature adjustment to the surveillance data if the temperature of the surveillance capsule is different than that of the vessel. Appendix A, Procedure 1, Procedural Step 3(b) of the LTR will be revised as follows:

b. If the vessel wall temperature is an outlier, appropriate temperature adjustments to the surveillance data may be required. An appropriate temperature adjustment is a 1 °F degree increase in ΔRT_{NDT} per 1°F decrease in irradiation temperature [A-7]. Alternatively, the temperature adjustment can be determined using appropriate NRC guidance. Any temperature adjustments shall be identified and described in the PTLR.

In response to NRC staff RAI No. 9, Appendix A will be revised to define the initial RT_{NDT}, as follows:

> 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: 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]. Other generic mean values may be used, provided they are justified and have NRC review and approval. The generic mean values used shall be identified in the PTLR.

Reference A-6 is the Pressurized Thermal Shock rule, 10 CFR 50.61. The rule provides generic initial RT_{NDT} values for welds made with Linde 80, 0091, 1092, and 124 and ARCOS B-5 weld fluxes. These values have been reviewed and approved by the NRC staff. Therefore, they are also applicable for BWR RPVs.

In response to NRC staff RAI No. 10, Appendix A will be revised to identify information that the licensee should review to determine whether the data is "credible" or "non-credible" in accordance with RG 1.99, Revision 2. The following two steps will be added to Appendix A, Procedure 1, Procedural Step 3 of the LTR:

d. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 foot-pound temperature and the upper shelf energy unambiguously.

e. When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Reg. Guide 1.99 Rev. 2, Regulatory Position 2.1, normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

The changes to Appendix A are acceptable, because they provide additional guidance to the licensees and the guidance has been previously approved by the NRC staff. The NRC staff requires that these changes to Appendix A of the LTR be incorporated into the -A version of the report.

3.4 Evaluation of Appendix B of the LTR

Appendix B provides a template PTLR. To ensure that the P/T limits were developed using the LTR methodology, the NRC staff in RAI No. 11 requested that the following information be included in the PTLR:

a) The initial RT_{NDT} [IRT_{NDT}] for all reactor pressure vessel materials and the method of determining the initial RT_{NDT} (i.e., ASME Code, Generic Communication, Branch Technical Position - MTEB 5-2 in SRP 5.3.2 in NUREG-0800, or other NRC-approved methodologies),

b) The chemistry (weight-percent copper and nickel) and ART at the 1/4 thickness location for all beltline materials, and

c) The computer codes used in the finite element analysis to determine for calculating bending and membrane stresses from Section 2.5 of the methodology.

d) Identify whether "Procedure #1" or "Procedure #2" was utilized to evaluate the surveillance data. If surveillance data was utilized, provide the surveillance data and the analysis of the surveillance data that was used to determine the ART. If surveillance data was not utilized, state why it was not utilized.

In response to NRC staff RAI No. 11 items (a), (b), and (d), the BWROG proposed that the following be added to Section 2.3 of the LTR:

> The following information should be included in the PTLR with respect to the ART calculations:
>
> a. The IRT_{NDT} for all RPV materials and the method of determining the IRT_{NDT} (i.e., ASME Code, Generic Communication, Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, or other NRC-approved methodologies).
>
> b. The chemistry (weight-percent copper and nickel) and ART at the 1/4t location for all beltline materials.
>
> c. Identify whether "Procedure 1" or "Procedure 2" from Appendix A was utilized to evaluate the surveillance data. If surveillance data was utilized, provide the surveillance [data] and the analysis of the surveillance data that was used to determine the ART values. If surveillance data was not utilized, state why it was not utilized.

The changes are acceptable, because they provide additional guidance for licensees and provide information that the NRC staff needs to evaluate the PTLR. The NRC staff requires that these changes be incorporated into the -A version of the report.

The response to item c) was discussed in the Section 3.1 of this SE (Evaluation of GL 96-03, Attachment 1, Criterion 5). Section 2.5 will be revised to request that the PTLR contain the requested information.

4.0 CONCLUSION

The NRC staff concludes that BWROG LTR SIR-05-044 satisfies the criteria in Attachment 1 to GL 96-03 and provides adequate methodology for BWR licensees to calculate P/T limit curves. By using this methodology and following the PTLR guidance in GL 96-03, as amended by NRC TSTF-419, BWR licensees will be able to relocate the P/T limit curves and the associated heatup/cooldown rates from TS to a PTLR, a licensee-controlled document.

The NRC staff has recommended, as noted in this SE, additional changes to Table 1-1 of the LTR. The BWROG must incorporate the NRC staff recommended changes and the changes proposed by the BWROG in their letter dated August 29, 2006, into the -A version of the report.

5.0 REFERENCES

1. NRC Generic Letter 96-03, "Relocation of the Pressure-Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," January 31, 1996 (ADAMS Legacy Library Accession No. 9601290350).

2. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," 2005 Edition.

3. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 2004 Edition.

4. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 (ADAMS Accession No. ML003740284).

5. NUREG-0800, NRC Standard Review Plan, Section 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock," Draft Revision 2, June 1996.

6. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 2005 Edition.

7. NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession No. ML010890301).

8. 10 CFR 50.36, "Technical specifications," 2005 Edition.

9. NRC Technical Specification Traveler Form TSTF-419, Revision 2, "Pressure Temperature Limits Report [PTLR]," September 16, 2001 (ADAMS Accession No. ML012690234).

10. 10 CFR 50.90, "Application for amendment of license or construction permit," 2005 Edition.

11. NRC Regulatory Issue Summary 2004-04, "Use of Code Cases N-588, N-640 and N-641 in Developing Pressure-Temperature Operating Limits," April 5, 2004 (ADAMS Accession No. ML040920323).

12. NRC Regulatory Guide 1.147, Revision 14, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," August 2005 (ADAMS Accession No. ML052510117).

13. WRC Bulletin No. 175, "Pressure Vessel Research Committee (PVRC) Recommendations on Fracture Toughness Requirements for Ferritic Materials," August 1972.

14. GE Topical Report NEDE-21821-02, "BWR Feedwater Nozzle/Sparger Final Report, Supplement 2," August 1979.

15. NRC, 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.

Attachment: Resolution of Comments

Principle Contributor: B. Elliot

Date: February 6, 2007

RESOLUTION OF COMMENTS

ON DRAFT SAFETY EVALUATION FOR

STRUCTURAL INTEGRITY ASSOCIATES TOPICAL REPORT (TR)

SIR-05-044, "PRESSURE TEMPERATURE REPORT METHODOLOGY FOR

BOILING WATER REACTORS"

By letter dated December 20, 2006 (Agencywide Document Access and Management System Accession No. ML063600123), the Boiling Water Reactor Owners' Group (BWROG) provided comments on the draft safety evaluation (SE) for Structural Integrity Associates Topical Report (TR) SIR-05-044, "Pressure Temperature Report Methodology for Boiling Water Reactors." The following is the NRC staff resolution of those comments.

BWROG Comment:

Delete from Pages 2 and 3 of the SE references to LTOP limit setpoint values, since these do not apply to BWRs.

NRC Resolution:

The NRC staff agreed to change and delete references to the LTOP limit setpoint values on Pages 2 and 3.

ATTACHMENT

FOREWORD

The principle objective of this report is to provide the current Structural Integrity Associates, Inc. methodology for developing reactor coolant system (RCS) pressure test, core not critical, and core critical pressure-temperature (P-T) curves for boiling water reactors (BWRs) at the request of the BWR Owners' Group (BWROG) Pressure-Temperature Curve Committee. This methodology, which has been approved by the NRC, 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 P-T curves, but rather the NRC-approved methodologies that can be referenced by licensees to license the PTLR concept.

This "-A" version of the report incorporates the final NRC Safety Evaluation Review (SER) and all associated actions.

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
FPL Energy	Duane Arnold
NMC	Monticello
NPPD	Cooper
PPL	Susquehanna
Progress Energy	Brunswick
PSEG Nuclear	Hope Creek
SNC	Hatch
TVA	Browns Ferry

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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

that has been approved by the NRC. 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. 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. This NRC-approved 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 NRC-approved methodologies that can be referenced by licensees 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

The "Requirements for Methodology and PTLR" table in GL 96-03 identifies the minimum requirements to be included in the PTLR methodology, and the minimum requirements to be

included in the PTLR. Table 1-1 provides a summary of how the PTLR methodology included in this report satisfies the minimum requirements identified in the GL 96-03 table.

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.

Table 1-1: Summary of GL 96-03 PTLR Methodology Requirements

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	APPLICABLE SECTION OF LTR WHERE REQUIREMENTS ARE ADDRESSED
1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).	Describe transport calculation methods including computer codes and formulas used to calculate neutron fluence. Provide references	Provide the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.	Not covered by this LTR. Fluence methods and results must comply with RG 1.190 and have NRC approval for use with this LTR.
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.	Briefly describe the surveillance program. Licensee transmittal letter should identify by title and number report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. Topical/generic report contains placeholder only. Reference Appendix H to 10 CFR Part 50.	Provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located.	See Appendix A of Template PTLR included in Appendix B of this LTR.
3. Low temperature overpressure protection (LTOP) system limits developed using NRC-approved methodologies may be included in the PTLR.	Describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics. Reference SRP Section 5.2.2; ASME Code Case N-514; ASME Code, Appendix G, Section XI as applied in accordance with 10 CFR 50.55.	Provide setpoint curves or setpoint values.	Not applicable for BWRs.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.	Describe the method for calculating the ART using Regulatory Guide 1.99, Revision 2.	Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (t = vessel beltline thickness).	See Section 2.3 of this LTR.

Table 1-1: Summary of GL 96-03 PTLR Methodology Requirements (concluded)

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	APPLICABLE SECTION OF LTR WHERE REQUIREMENTS ARE ADDRESSED
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800, SRP Section 5.3.2, Pressure-Temperature Limits.	Describe the application of fracture mechanics in constructing P/T curves based on ASME Code, Appendix G, Section XI, and SRP Section 5.3.2.	Provide the P/T curves for heatup, cooldown, criticality, and hydrostatic and leak tests.	See Sections 2.4 and 2.5 of this LTR.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.	Describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T curves.	Identify minimum temperatures on the P/T curves such as minimum boltup temperature and hydrotest temperature.	See Sections 2.7 and 2.8 of this LTR.
7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_\Delta$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma_\Delta$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Describe how the data from multiple surveillance capsules are used in the ART calculation. Describe procedure if measured value exceeds predicted value. <u>WHEN OTHER PLANT DATA ARE USED</u> 1. Identify the source(s) of data when other plant data are used. 2.a Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability. OR 2.b Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation. Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.	See Appendix A of this LTR.

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:

$$\mathrm{Margin} = 2\sqrt{\sigma_I^{\,2} + \sigma_\Delta^{\,2}} \qquad (2.3\text{-}5)$$

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.

The following information should be included in the PTLR with respect to the ART calculations:

a. The IRT_{NDT} for all RPV materials and the method of determining the IRT_{NDT} (i.e., ASME Code, Generic Communication, Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, or other NRC-approved methodologies).

b. The chemistry (weight-percent copper and nickel) and ART at the 1/4t location for all beltline materials.

c. Identify whether "Procedure 1" or "Procedure 2" from Appendix A was utilized to evaluate the surveillance data. If surveillance data was utilized, provide the surveillance data and the analysis of the surveillance data that was used to determine the ART values. If surveillance data was not utilized, state why it was not utilized.

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].

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[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.

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

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

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

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)

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

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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].

"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 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.

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[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.

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

In the subsections that follow, finite element analysis is discussed as a possible approach for providing the necessary stress analysis for RPV regions. If finite element analysis is utilized to develop P-T limits for any RPV region, the following information shall be provided in the PTLR:

a. Identify the computer code(s) that were used in the finite element stress analysis.

b. For any computer codes used, describe how the code(s) were verified or benchmarked. Computer code verification shall be in accordance with a qualified 10 CFR 50 Appendix B Quality Assurance Program. As a part of computer code verification, benchmarking consistent with NRC GL 83-11, Supplement 1 [17] shall be included.

c. Identify the assumptions and the inputs to the finite element analysis. Necessary inputs to the analysis include any or all of the following:
    - A description of plant operating conditions used (e.g., pressure and temperature). The conditions used must represent current plant operating conditions.
    - A description of the heat transfer coefficients used and the methodology used to calculate them.
    - A description of the model developed, including materials, material properties, finite element mesh pattern, and geometry.

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.

<u>*Closed Form Solution Method*</u>

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 [16]:

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

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

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

at $r = r_o$: $$\frac{\partial T}{\partial r} = 0 \qquad (2.5.1\text{-}3)$$

where:

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 $E\alpha$ product in Equation (2.5.1-4) is treated as a constant in the computation of thermal hoop stress.

The thermal stress intensity factor, K_{It}, for the thermal hoop stress distribution calculated from Equation 2.5.1-4 can be calculated at any specified time during the cooldown for a 1/4t inside surface defect using the following relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)\sqrt{\pi a} \qquad (2.5.1\text{-}6)$$

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:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.5.1\text{-}7)$$

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

a = the maximum crack depth (inches)

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:

$$K_{It} = 0.953 \times 10^{-3} \, (CR) \, (t^{2.5}) \tag{2.5.1-8}$$

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:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)\sqrt{\pi a} \tag{2.5.1-9}$$

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:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.5.1\text{-}10)$$

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]:

$$K_{It} = \left[\sigma_{sm} 1.1 M_K + \sigma_{sb} M_B\right]\sqrt{\frac{\pi a}{Q}} \qquad (2.5.1\text{-}11)$$

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:

$$K_{Im} = (K_{Ic} - K_{It}) / SF \qquad (2.5.1\text{-}12)$$

where:

K_{Im} = the allowable stress intensity factor caused by membrane (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 ART for all beltline weld and plate materials from Equation 2.4-2 ($ksi\sqrt{inch}$)

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

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:

$$P_{allow} = (K_{Im}\, t) / (M_m\, R_i) \qquad (2.5.1\text{-}13)$$

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.

Bottom Head Region

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:

$$K_{Im} = (K_{Ic} - K_{It}) / SF \qquad (2.5.1\text{-}14)$$

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:

$$P_{allow} = (2K_{Im}\, t) / (SCF\ M_m\ R_i) \qquad (2.5.1\text{-}15)$$

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:

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

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

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

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 1×10^{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 some cases, such results may already be 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, as shown in Figure 2-7, the thermal and pressure hoop stress distributions should be extracted from the finite element model. Each of the stress distributions should be fit with a third-order polynomial that reasonably fits the calculated stresses in the region of interest.

The thermal stress intensity factor, K_{It}, is computed based on the nozzle corner solution shown in Figure 2-8 for a postulated 1/4t (based on the section thickness) axial defect, as follows:

$$K_{It} = \sqrt{\pi a}\left[0.706\,C_{0t} + 0.537\left(\frac{2a}{\pi}\right)C_{1t} + 0.448\left(\frac{a^2}{2}\right)C_{2t} + 0.393\left(\frac{4a^3}{3\pi}\right)C_{3t}\right] \qquad (2.5.1\text{-}16)$$

where:

K_{It} = the thermal stress intensity factor for the limiting normal/upset transient (ksi $\sqrt{\text{inch}}$)

a = 1/4t postulated flaw depth (inches)

t = thickness of the cross-section through the limiting nozzle inner blend radius corner, as shown in Figure 2-7.

C_{0t}, C_{1t}, C_{2t}, C_{3t} = thermal stress polynomial coefficients based on fits to finite element analysis.

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:

$$K_{Ip} = (K_{Ic} - K_{It}) / SF \qquad (2.5.1\text{-}17)$$

where:

K_{Ip} = the allowable stress intensity factor caused by 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 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 based on the nozzle corner solution shown in Figure 2-8 for a postulated 1/4t (based on the section thickness) axial defect, as follows:

$$K_{Ip-applied} = \sqrt{\pi a}\left[0.706\, C_{0p} + 0.537\left(\frac{2a}{\pi}\right)C_{1p} + 0.448\left(\frac{a^2}{2}\right)C_{2p} + 0.393\left(\frac{4a^3}{3\pi}\right)C_{3p}\right] \quad (2.5.1\text{-}18)$$

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

a = 1/4t postulated flaw depth (inches)

t = thickness of the cross-section through the limiting nozzle inner blend radius corner, as shown in Figure 2-7.

C_{0p}, C_{1p}, C_{2p}, C_{3p} = pressure stress polynomial coefficients based on fits to finite element analysis.

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

$$P_{allow} = (K_{Ip}\, P) / K_{Ip\text{-}applied} \quad (2.5.1\text{-}19)$$

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:

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

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

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.

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[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 [14] 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

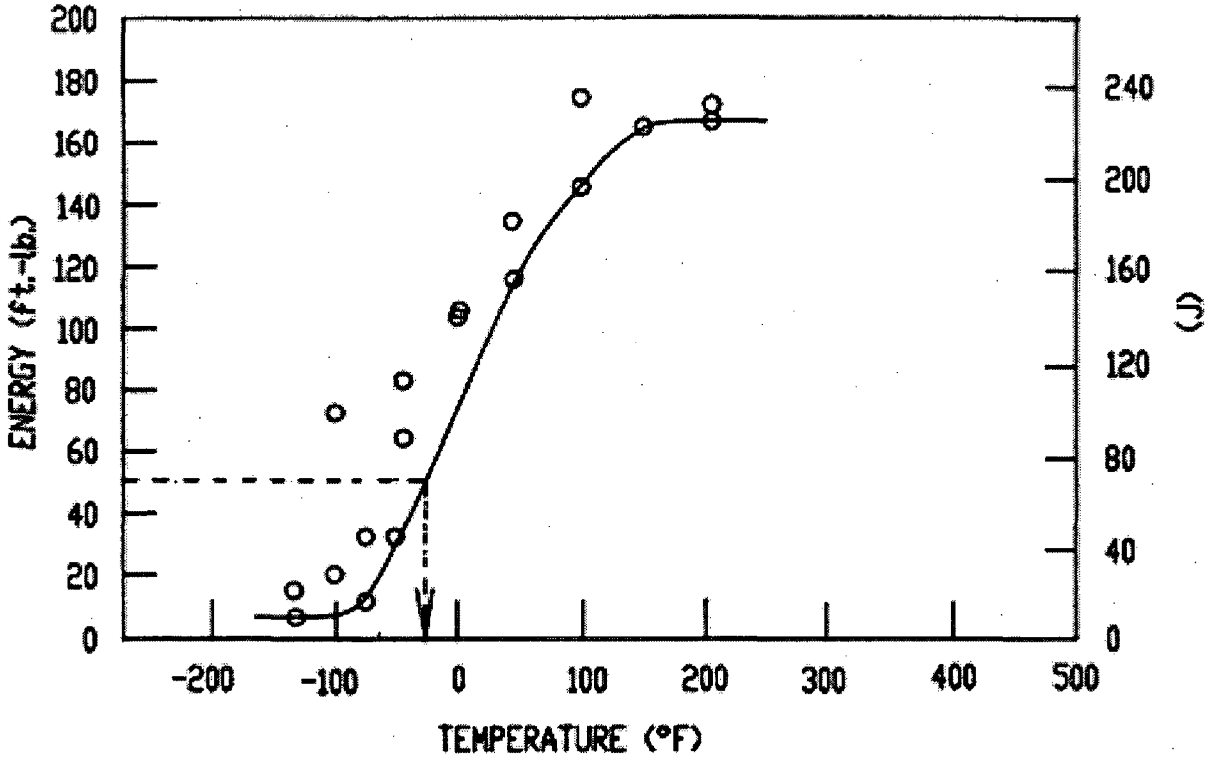
addition, there are no known mechanisms of degradation for this region, other than fatigue. However, the calculated design fatigue usage for this region is typically high for most BWR plants (i.e., greater than 0.8) due to the high bolt preload stresses, so it cannot be necessarily concluded that fatigue 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 [15], 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

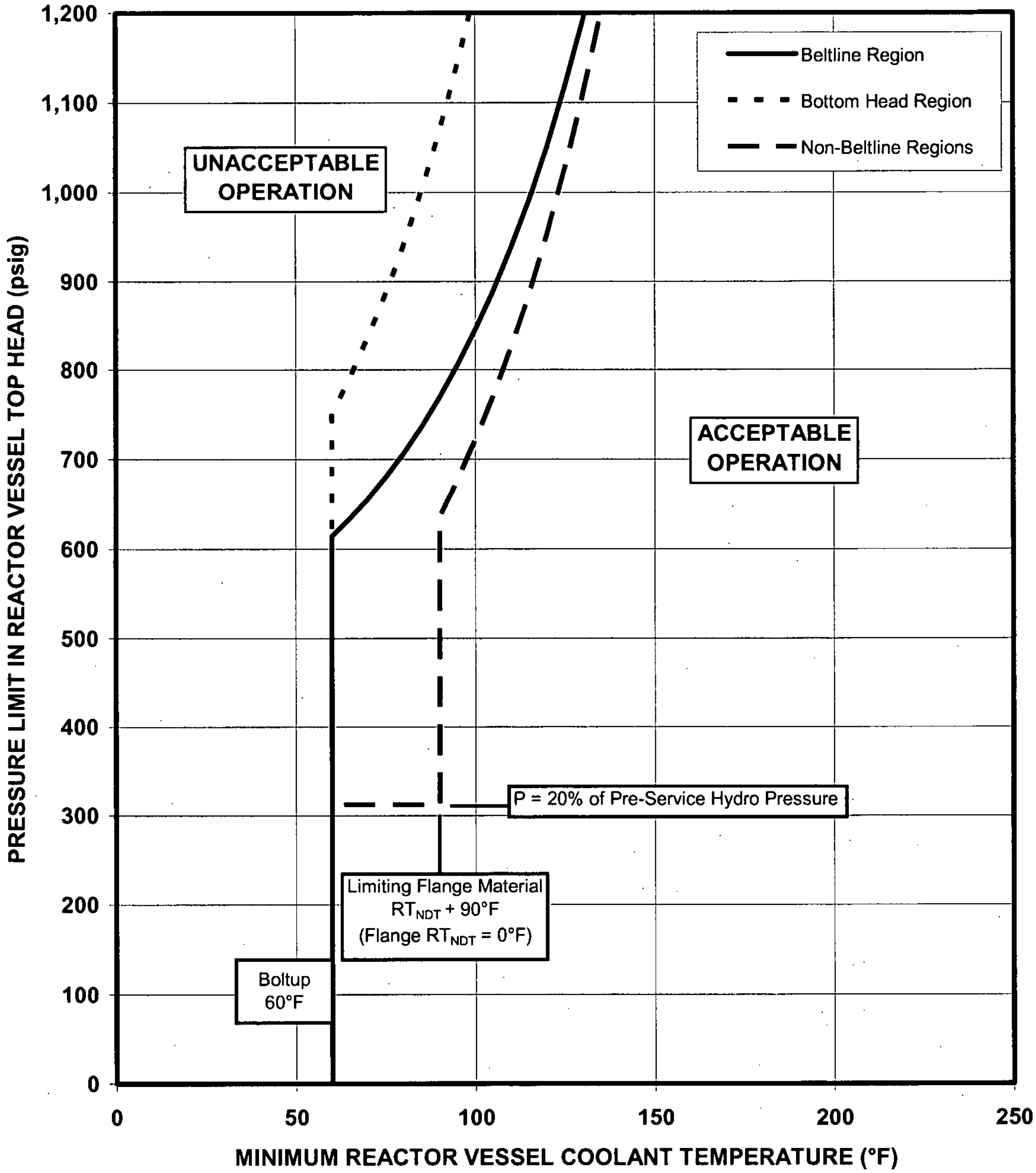


Figure 2-3. Sample Core Not Critical P-T Limit Curves for 32 EFPY

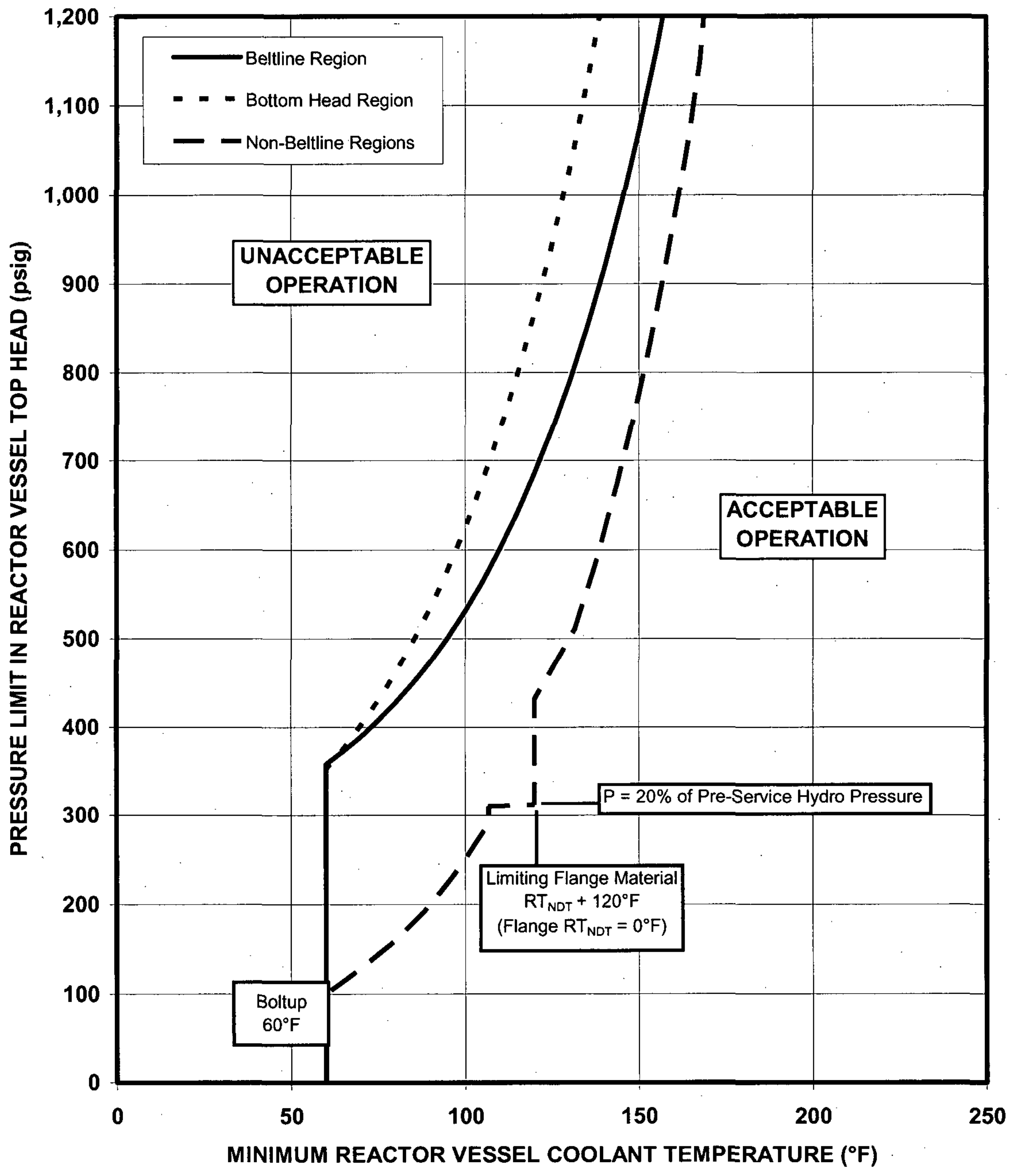


Figure 2-4. Membrane Stress Correction Factor (M_K) (WRC Bulletin No. 175 Method)

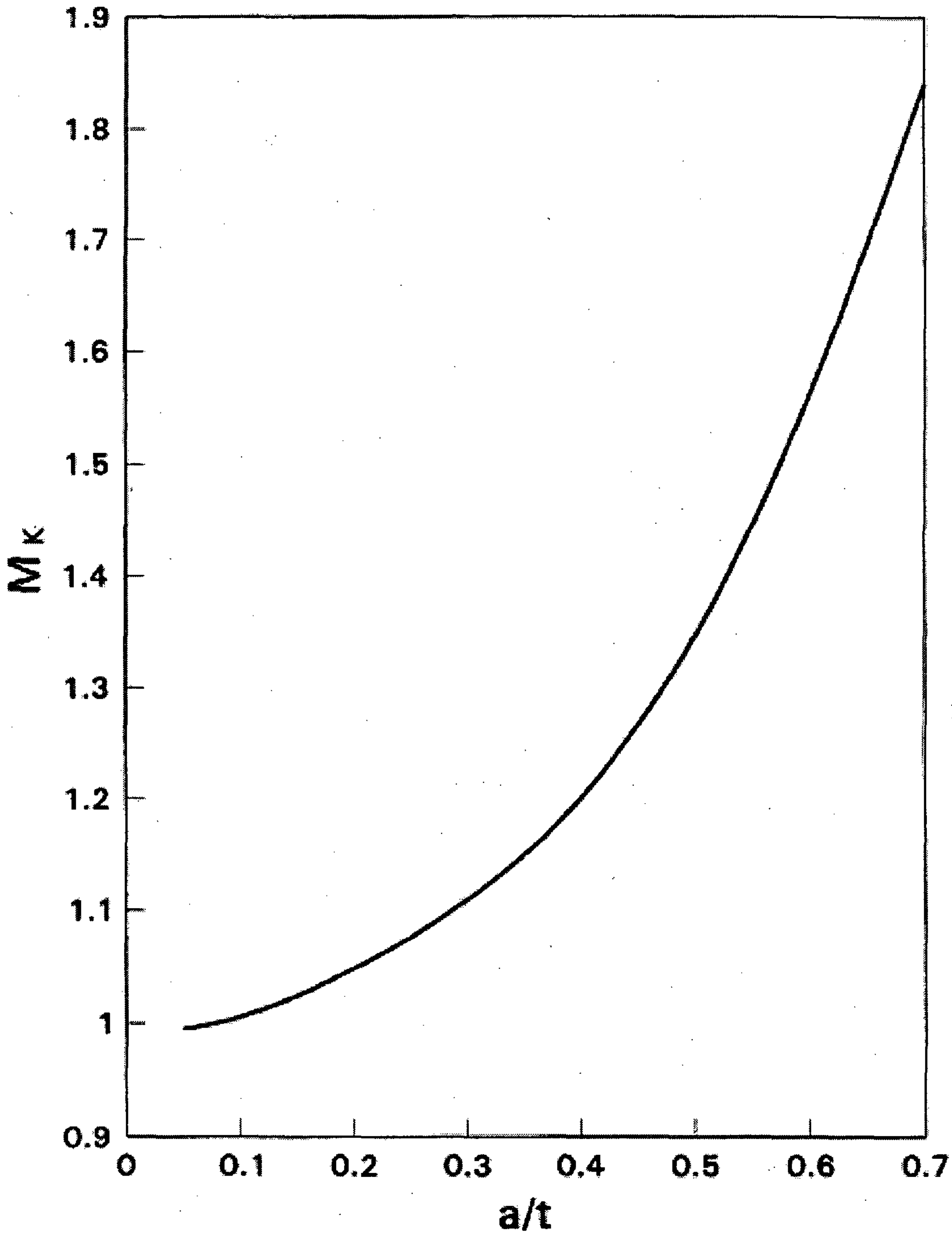


Figure 2-5. Bending Stress Correction Factor (M_B) (WRC Bulletin No. 175 Method)

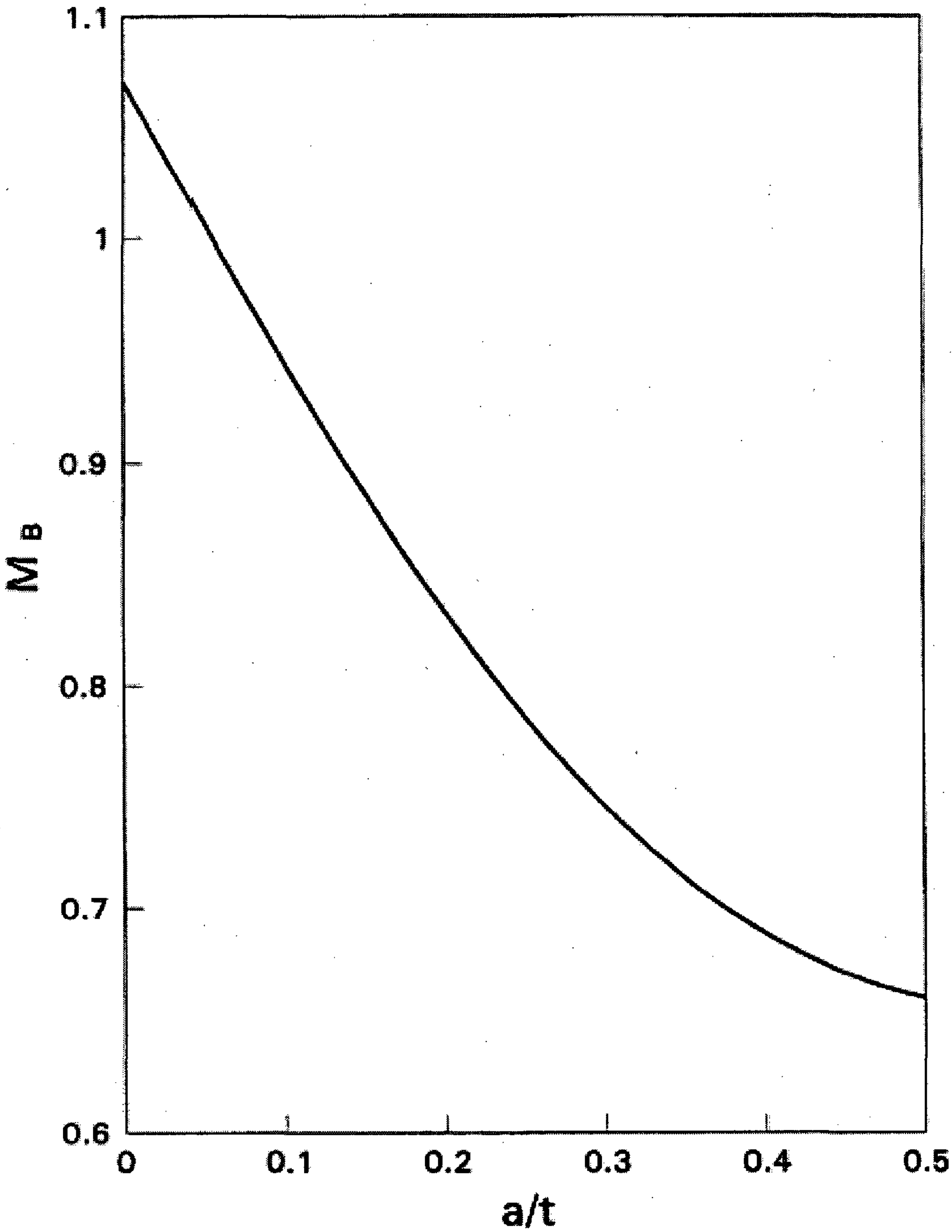


Figure 2-6. Nozzle Stress Intensity Factors (Figure A5-1 of WRC Bulletin No. 175)

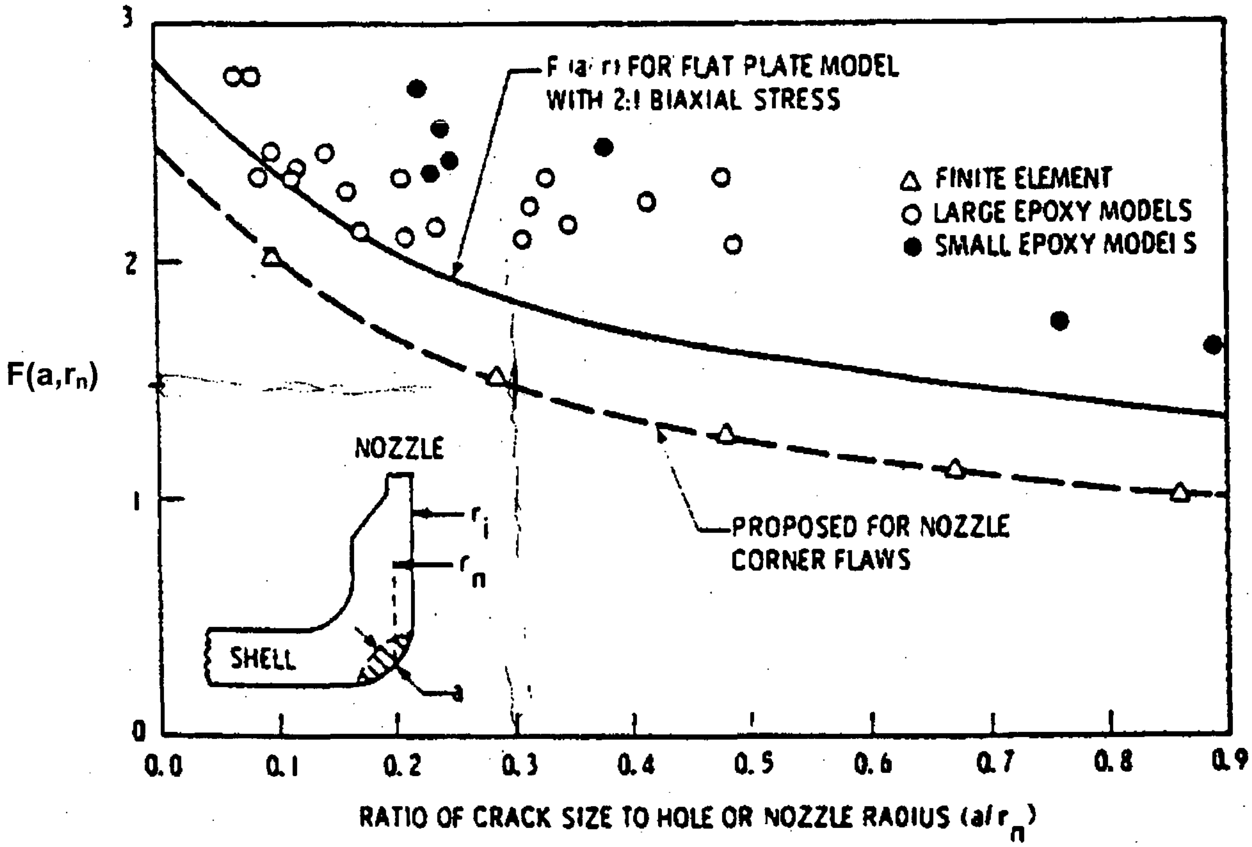


Figure 2-7: Nozzle Thickness Definition

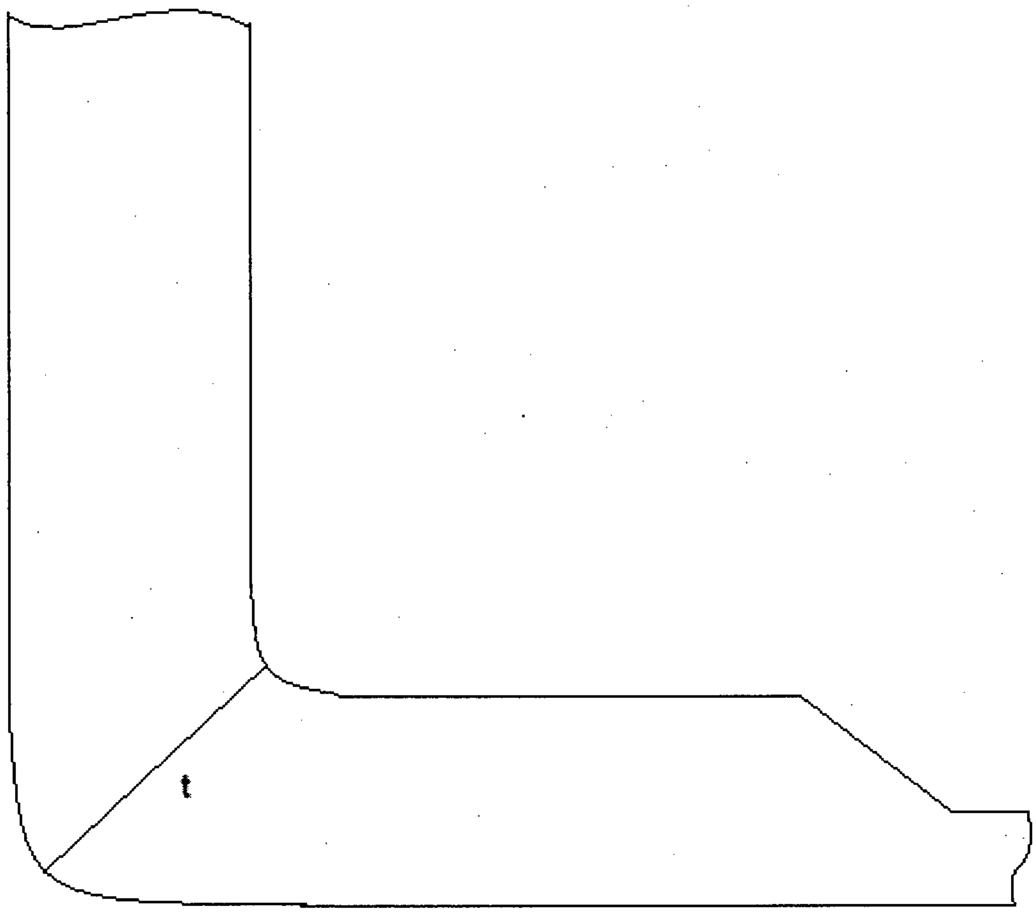
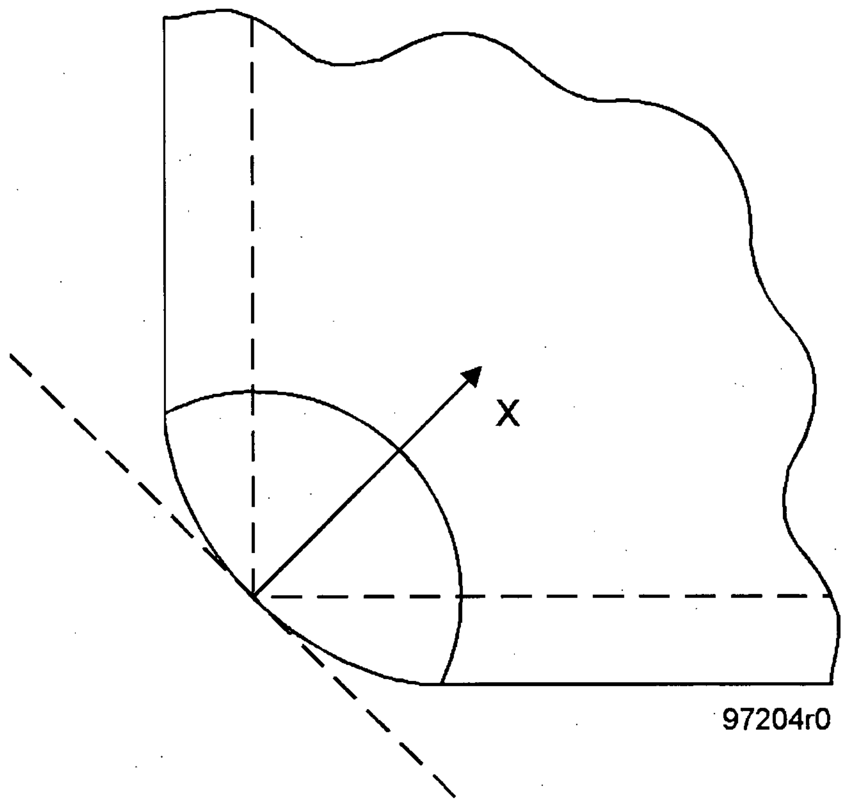


Figure 2-8: Stress Intensity Factor Solution for a Nozzle Corner Crack



SIMULATED 3-D NOZZLE CORNER CRACK

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; however, for RPV regions other than those defined in Section 2.0 of this report, licensees are required to submit methodologies to the NRC for review and approval prior to use.

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. Evaluate surveillance data in accordance with Appendix A of this report.

b. 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.

c. 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.

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

e. 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.

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

g. Repeat steps (b) 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.

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

i. 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. 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.

15. 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.

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

17. U. S. Nuclear Regulatory Commission, Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999.

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:

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

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: 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]. Other generic mean values may be used, provided they are justified and have NRC review and approval. The generic mean values used shall be identified in the PTLR.

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

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

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:

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

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:

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \tag{4}$$

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. An appropriate temperature adjustment is a 1°F degree increase in ΔRT_{NDT} per 1°F decrease in irradiation temperature [A-7]. Alternatively, the temperature adjustment can be determined using appropriate NRC guidance. Any temperature adjustments shall be identified and described in the PTLR.
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 <u>will</u> 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.

d. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 foot-pound temperature and the upper shelf energy unambiguously.
e. When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Reg. Guide 1.99 Rev. 2, Regulatory Position 2.1, normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82.

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 vessel and surveillance capsule materials have been calculated by the BWRVIP, as documented in BWRVIP-86-A [A-1]. Calculation of the best estimate chemistries for all other vessel materials should be determined in accordance with the NRC practice documented in Reference [A-7]. The suggested practice is documented in guidelines contained in BWRVIP-135. This evaluation is the responsibility of the plant, must be described in the PTLR, and must utilize NRC-approved methods.
2) Determine the Table CF for the best estimate chemistry of the surveillance weld (Table $CF_{\text{Surv. Chem}}$).
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 $CF_{\text{Vessel Chem.}}$.
5) If the Adjusted Surveillance CF is higher than the Table $CF_{\text{Vessel Chem}}$, 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 $CF_{\text{Vessel Chem}}$, 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 $CF_{\text{Vessel Chem}}$, and the surveillance data are not credible, then the higher (e.g., Table $CF_{\text{Vessel Chem}}$) 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 vessel and surveillance capsule materials have been calculated by the BWRVIP, as documented in BWRVIP-86-A [A-1]. Calculation of the best estimate chemistries for all other vessel materials should be determined in accordance with the NRC practice documented in Reference [A-7]. The suggested practice is documented in guidelines contained in BWRVIP-135. This evaluation is the responsibility of the plant, must be described in the PTLR, and must utilize NRC-approved methods.*

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):

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

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 not credible, then σ_Δ is the lower of the following:

a) 28°F for welds, 17°F for base metal, or

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:

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

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: 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]. Other generic mean values may be used, provided they are justified and have NRC review and approval. The generic mean values used shall be identified in the PTLR.

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 vessel and surveillance capsule materials have been calculated by the BWRVIP, as documented in BWRVIP-86-A [A-1]. Calculation of the best estimate chemistries for all other vessel materials should be determined in accordance with the NRC practice documented in Reference [A-7]. The suggested practice is documented in guidelines contained in BWRVIP-135. This evaluation is the responsibility of the plant, must be described in the PTLR, and must utilize NRC-approved methods.

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, or

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.

A-7. "Generic Letter 92-01 and RPV Integrity Assessment – Status, Schedule, and Issues," Presentation by K. Wichman, M. Mitchell, and A. Hiser at NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998.

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].

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[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.

The initial RT_{NDT}, the chemistry (weight-percent copper and nickel) and adjusted reference temperature at the 1/4 thickness location for all RPV beltline materials significantly affected by fluence (i.e., fluence > 10^{17} n/cm^2 for E > 1 MeV) are shown in Table 7 for SSES-1 and Table 8 for SSES-2. The initial RT_{NDT} values shown in Tables 7 and 8 were developed using the procedures of Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, and they have been previously approved for use by the NRC [6.6].

For SSES-1, limiting RPV plate C-2433-1, BWRVIP "Procedure 1" was utilized since the heat number of this material is identical to the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. For limiting RPV weld 494K2351, BWRVIP "Procedure 2" was utilized since the heat number of this material is different than the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. Therefore, Regulatory Guide 1.99, Revision 2 chemistry factors were used in the determination of the ART values for all materials for SSES-1.

For SSES-2, limiting RPV plate C-2421-3, BWRVIP "Procedure 2" was utilized since the heat number of this material is different than the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. For limiting RPV

weld 624263, BWRVIP "Procedure 2" was utilized since the heat number of this material is different than the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. Therefore, Regulatory Guide 1.99, Revision 2 chemistry factors were used in the determination of the ART values for all materials for SSES-2.

The only computer code used in the determination of the SSES P-T curves was the ANSYS (Version 4.4) finite element computer program for the feedwater nozzle (non-beltline) stresses. This program was controlled under the vendor's 10 CFR 50 Appendix B Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 88-13, Supplement 1 was performed as a part of the computer program verification by comparing the solutions produced by the computer code to hand calculations for several problems. The following inputs were used as input to the finite element analysis *[Editorial note: The following items must be included on a plant-specific basis]*:

- *Plant operating conditions must be listed here. These conditions represent current plant operating conditions.*
- *Heat transfer coefficients must be listed here. These values were developed using conventional heat transfer methods for forced convection flow on a vertical flat plate.*
- *A description of the finite element model must be listed here, including materials, material properties, finite element mesh pattern, and geometry.*

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.

6.6 *NRC approval letter for IRT_{NDT} values. [Editorial note: The appropriate plant-specific reference is to be included here.]*

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

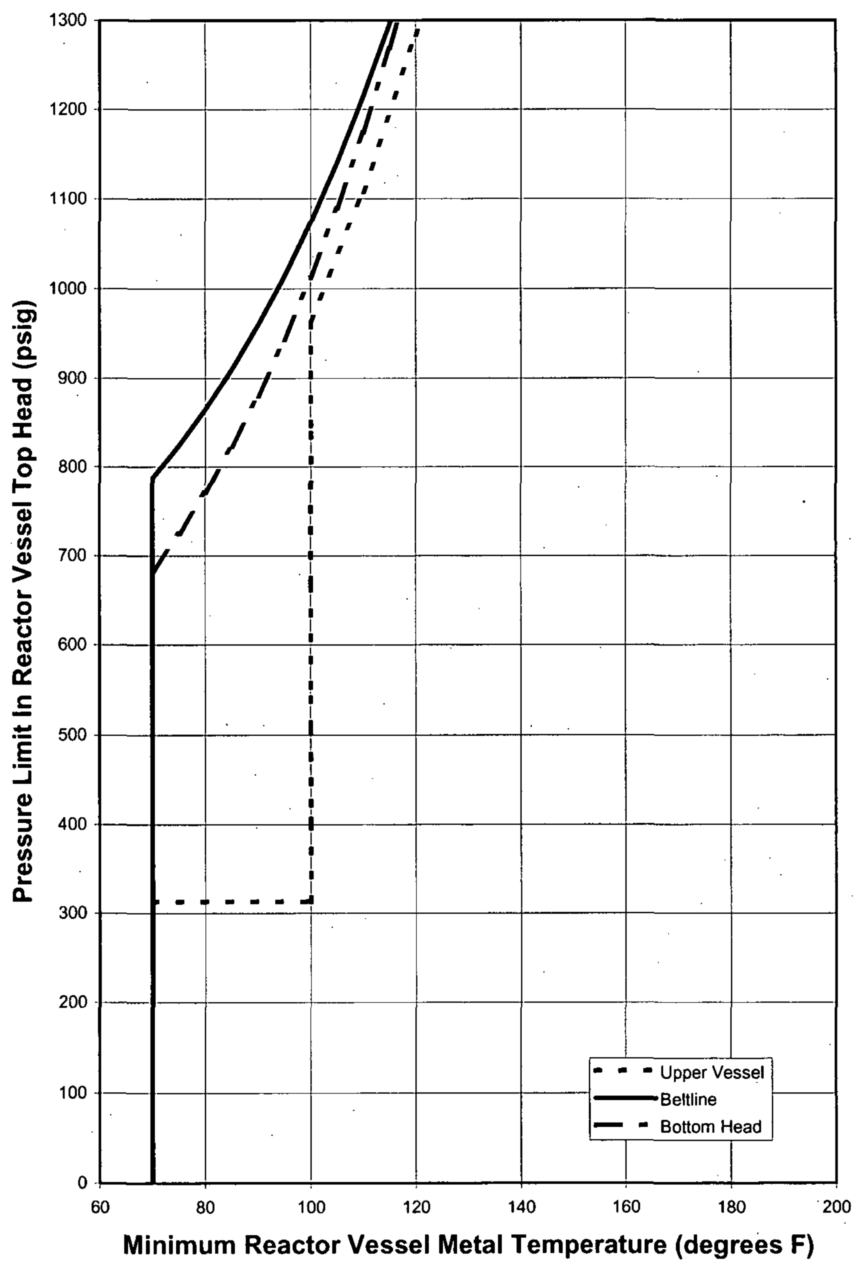

Figure 2: SSES Unit 1 Core Not Critical (Curve B) P-T Curves

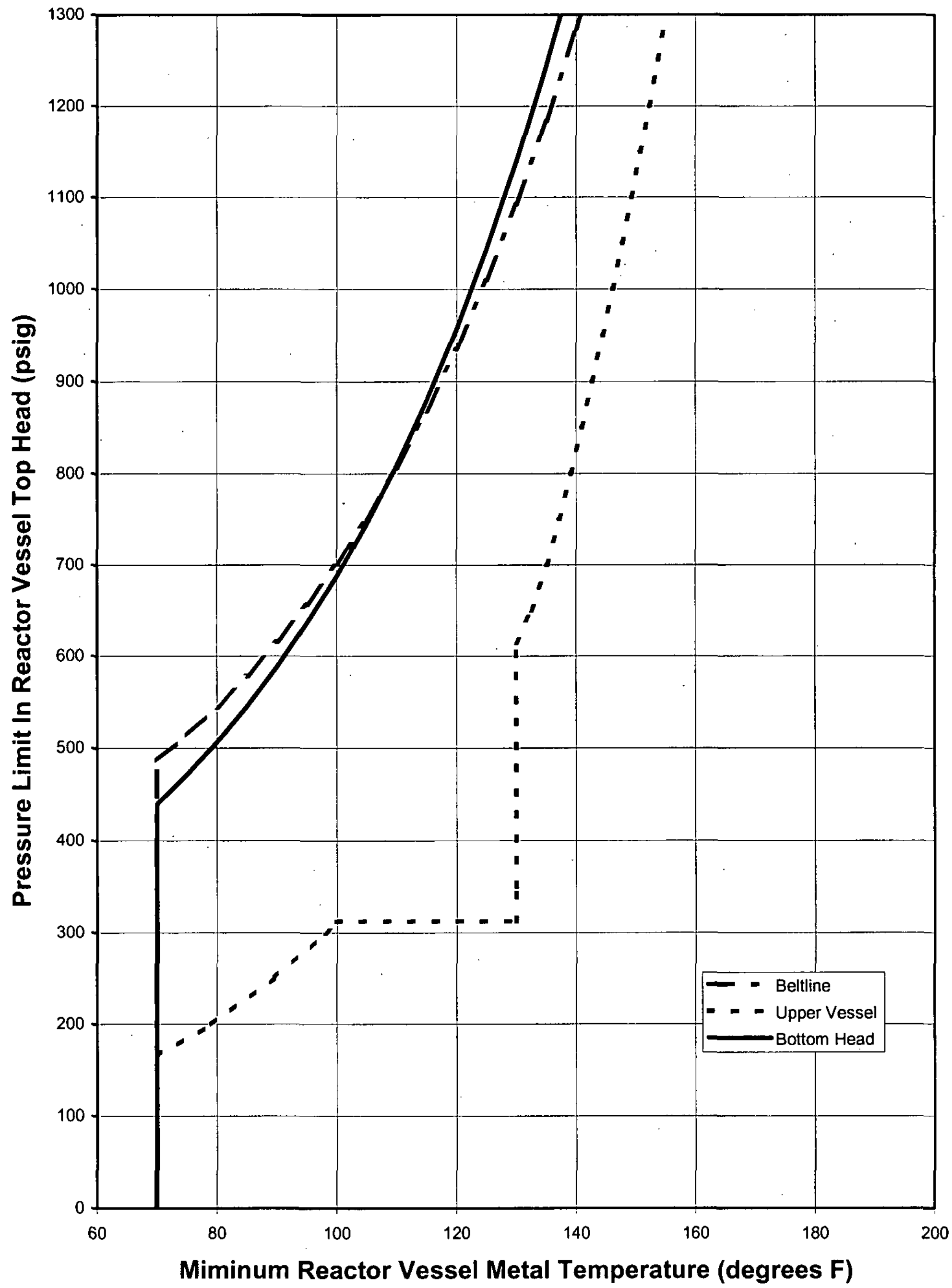

Figure 3: SSES Unit 1 Core Critical (Curve C) P-T Curve

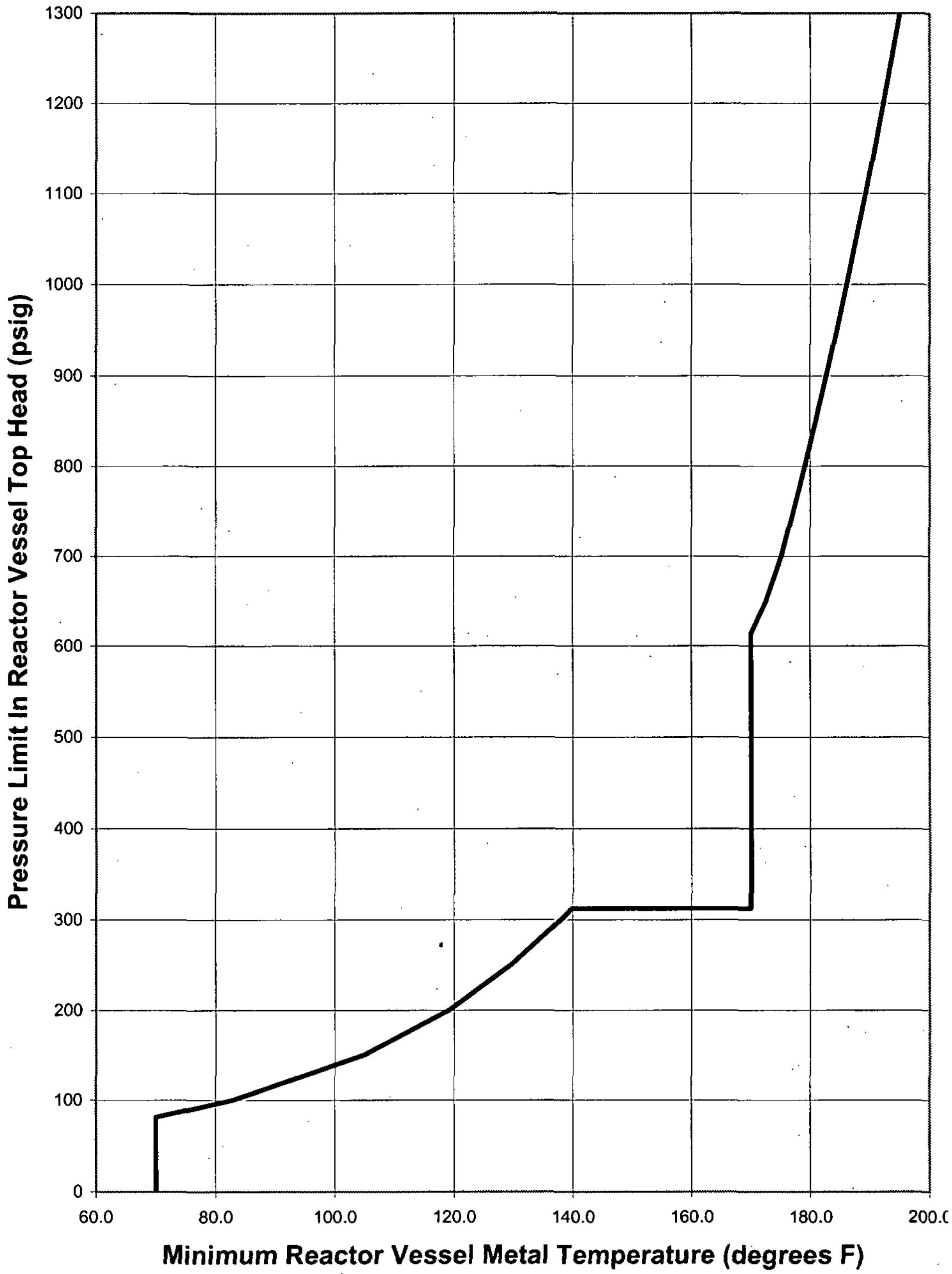

Figure 4: SSES Unit 2 Pressure Test (Curve A) P-T Curves

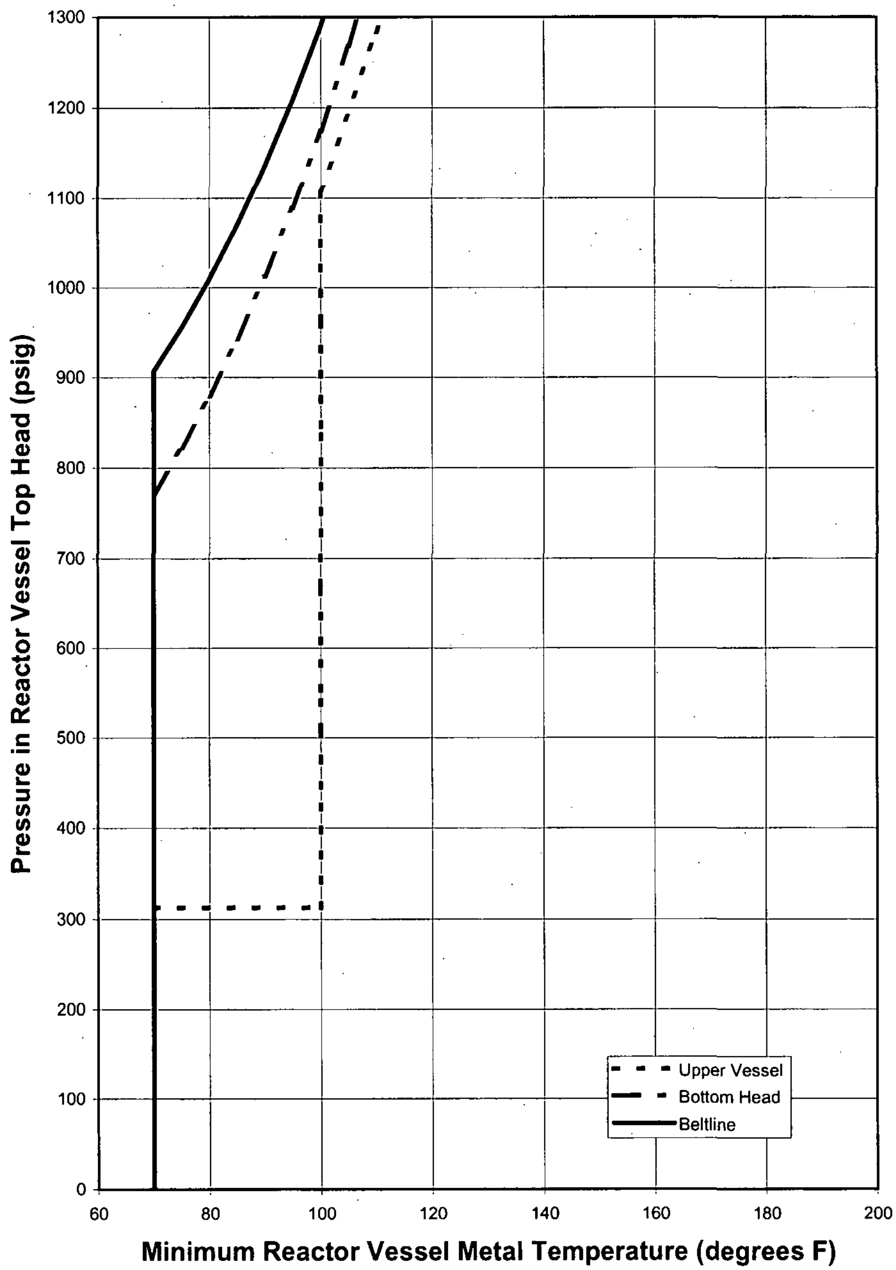

Figure 5: SSES Unit 2 Core Not Critical (Curve B) P-T Curves

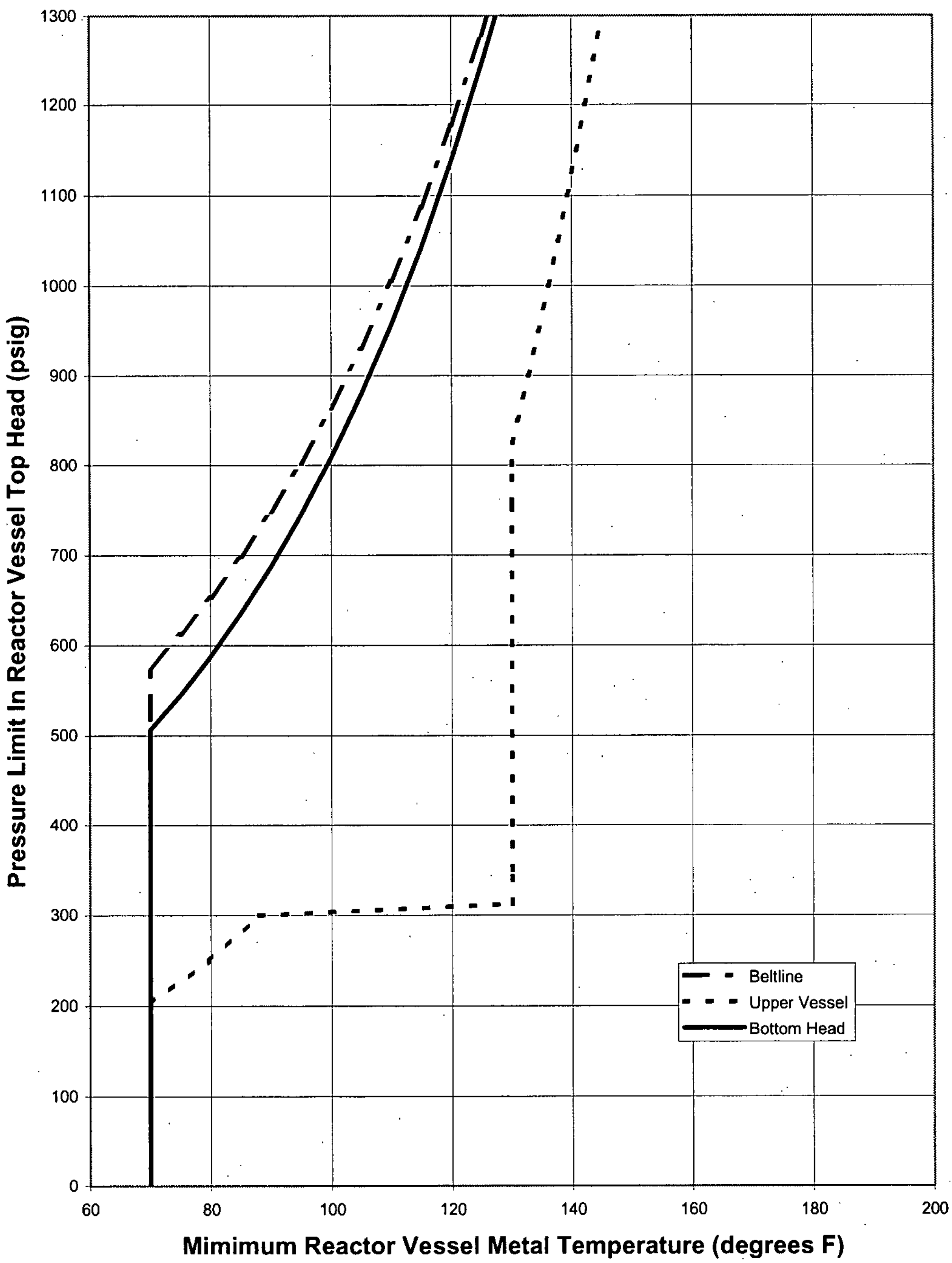

Figure 6: SSES Unit 2 Core Critical (Curve C) P-T Curve

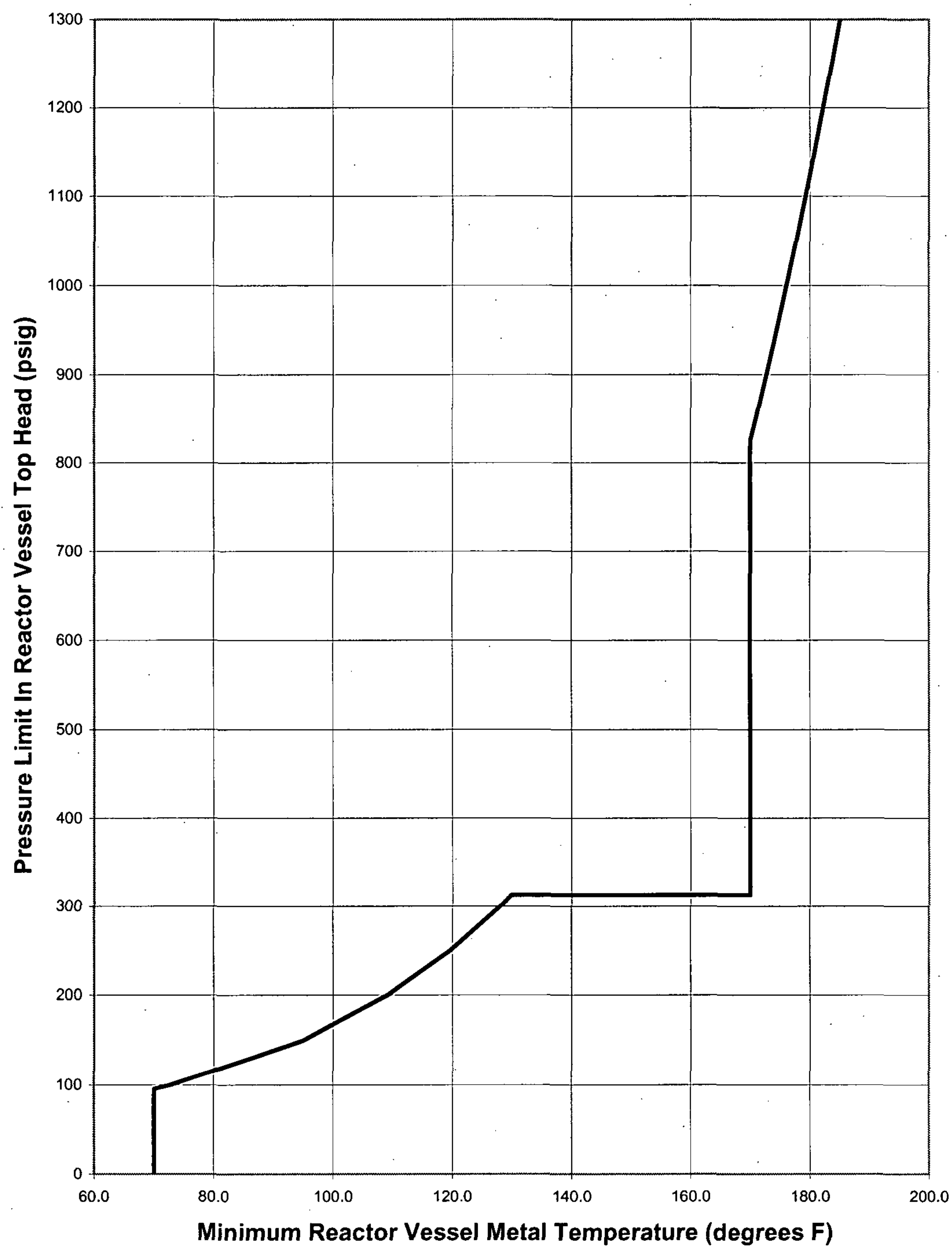

Table 1: SSES Unit 1 Pressure Test (Curve A) P-T Curves

Beltline Region

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

Inputs:

Plant =	**Susquehanna**			
Component =	**Beltline**			
Vessel thickness, t =	**6.1875**	inches, so $\sqrt{t}$ =	2.487	$\sqrt{\text{inch}}$
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)**	**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	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/rn) =	**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	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 $RT_{NDT}{}^{l}$ =	**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	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} =	19.04	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	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	137.8	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 =	**2.54**			
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	-	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	53.9	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 √t = 2.487 √inch
Vessel Radius, R =	**126.6875**	inches
ART_{NDT} =	**46.7**	°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	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/rn) =	**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 $\sqrt{t}$ = 2.487 $\sqrt{\text{inch}}$
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} =	**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 =	**2.54**				
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 ($ksi{\cdot}inch^{1/2}$)	KIp ($ksi{\cdot}inch^{1/2}$)	Total KIc ($ksi{\cdot}inch^{1/2}$)	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_{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	85.23	38.08	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 =	2.54				
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	-	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	178.0	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

Table 7: SSES-1 ART Calculations for 32 EFPY

Part Name & Material	ID No.	Heat No.	Lot No.	Estimated Initial RT_{NDT} (°F)	Chemistry		Chemistry Factor (°F)	Adjustments For 1/4t				
					Cu (wt %)	Ni (wt %)		ΔRT_{NDT} (°F)	Margin Terms: σ_Δ (°F)	Margin Terms: σ_I (°F)	EFPY	ART_{NDT} (°F)
Lower Shell #1	21-1	B5083-1	---	-8	0.14	0.48	94.6	28.2	14.1	0.0	32.0	48.5
Lower Shell #2	21-2	C0770-2	---	-20	0.14	0.50	95.5	28.5	14.2	0.0	32.0	37.0
Lower Shell #3	21-3	C0814-2	---	-20	0.13	0.51	88.3	26.4	13.2	0.0	32.0	32.7
Lower-Int. Shell #1	22-1	C0803-1	---	-10	0.09	0.53	58.0	19.3	9.6	0.0	32.0	28.6
Lower-Int. Shell #2	22-2	C0776-1	---	6	0.12	0.48	80.6	26.8	13.4	0.0	32.0	59.6
Lower-Int. Shell #3	22-3	C2433-1	---	18	0.10	0.63	65.3	21.7	10.9	0.0	32.0	61.4
Weld #1	---	629616	L320A27AG	-50	0.04	0.99	54.0	18.0	9.0	0.0	32.0	-14.1
Weld #2	---	411L3071	L311A27AF	-50	0.03	0.93	41.0	13.6	6.8	0.0	32.0	-22.7
Weld #3	---	494K2351	L307A27AD	-50	0.04	1.10	54.0	18.0	9.0	0.0	32.0	-14.1

Fluence Information:

Location	Wall Thickness (inches): Full	Wall Thickness (inches): 1/4t	EFPY	Fluence at ID (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Lower Shell #1	6.160	1.540	**32.00**	7.50E+17	0.691	5.18E+17	0.298
Lower Shell #2	6.160	1.540	**32.00**	7.50E+17	0.691	5.18E+17	0.298
Lower Shell #3	6.160	1.540	**32.00**	7.50E+17	0.691	5.18E+17	0.298
Lower-Int. Shell #1	6.160	1.540	**32.00**	9.20E+17	0.691	6.36E+17	0.332
Lower-Int. Shell #2	6.160	1.540	**32.00**	9.20E+17	0.691	6.36E+17	0.332
Lower-Int. Shell #3	6.160	1.540	**32.00**	9.20E+17	0.691	6.36E+17	0.332
Weld #1	6.160	1.540	**32.00**	9.20E+17	0.691	6.36E+17	0.332
Weld #2	6.160	1.540	**32.00**	9.20E+17	0.691	6.36E+17	0.332
Weld #3	6.160	1.540	**32.00**	9.20E+17	0.691	6.36E+17	0.332

Notes: 1. Material and fluence information taken from GE Report No. GE-NE-523-169-1292, "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," March 1993, Table 7-3.
(Note that Table 7-3 has a typographical error for Heat No. C0803-1; refer to NRC RVID2 database and Table 3-1 of the GE Report.)
2. The calculations shown in this table are not for design use, as they utilize outdated fluence results. These calculations are for comparison purposes only.

Table 8: SSES-2 ART Calculations for 32 EFPY

Part Name & Material	ID No.	Heat No.	Lot No.	Estimated Initial RT_{NDT} (°F)	Chemistry: Cu (wt %)	Chemistry: Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4t: ΔRT_{NDT} (°F)	Margin Terms: σ_Δ (°F)	Margin Terms: σ_I (°F)	EFPY	ART_{NDT} (°F)
Lower Shell #1	21-1	6C956-1-1	---	-20	0.11	0.55	73.5	20.1	10.1	0.0	32.0	20.2
Lower Shell #2	21-2	6C980-1-1	---	-20	0.10	0.56	65.0	17.8	8.9	0.0	32.0	15.6
Lower Shell #3	21-3	6C1053-1-1	---	10	0.10	0.58	65.0	17.8	8.9	0.0	32.0	45.6
Lower-Int. Shell #1	22-1	C2421-3	---	-10	0.13	0.68	93.0	28.3	14.2	0.0	32.0	46.7
Lower-Int. Shell #2	22-2	C2929-1	---	-20	0.13	0.64	92.0	28.0	14.0	0.0	32.0	36.1
Lower-Int. Shell #3	22-3	C2433-2	---	2	0.10	0.63	65.3	19.9	10.0	0.0	32.0	41.8
Weld #1	---	629616	L320A27AG	-50	0.04	0.99	54.0	16.5	8.2	0.0	32.0	-17.1
Weld #2	---	624263	E204A27A	-20	0.06	0.89	82.0	25.0	12.5	0.0	32.0	30.0
Weld #3	---	09M057	C109A27A	-36	0.03	0.89	41.0	12.5	6.2	0.0	32.0	-11.0

Fluence Information:

Location	Wall Thickness (inches): Full	Wall Thickness (inches): 1/4t	EFPY	Fluence at ID (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Lower Shell #1	6.160	1.540	**32.00**	6.40E+17	0.691	4.42E+17	0.274
Lower Shell #2	6.160	1.540	**32.00**	6.40E+17	0.691	4.42E+17	0.274
Lower Shell #3	6.160	1.540	**32.00**	6.40E+17	0.691	4.42E+17	0.274
Lower-Int. Shell #1	6.160	1.540	**32.00**	7.80E+17	0.691	5.39E+17	0.305
Lower-Int. Shell #2	6.160	1.540	**32.00**	7.80E+17	0.691	5.39E+17	0.305
Lower-Int. Shell #3	6.160	1.540	**32.00**	7.80E+17	0.691	5.39E+17	0.305
Weld #1	6.160	1.540	**32.00**	7.80E+17	0.691	5.39E+17	0.305
Weld #2	6.160	1.540	**32.00**	7.80E+17	0.691	5.39E+17	0.305
Weld #3	6.160	1.540	**32.00**	7.80E+17	0.691	5.39E+17	0.305

Notes: 1. Material and fluence information taken from GE Report No. GE-NE-523-107-0893, Revision 1, "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," October 1993, Table 7-3.
(Note that Table 7-3 has a typographical error for Heat No. 6C956-1-1; refer to NRC RVID2 database and Table 3-1 of the GE Report.)
2. The calculations shown in this table are not for design use, as they utilize outdated fluence results. These calculations are for comparison purposes only.

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.

APPENDIX C:

NRC REQUEST FOR ADDITIONAL INFORMATION (RAI)

AND

BWROG RESPONSE TO RAI

NRC RAI

-----Original Message-----
From: Michelle Honcharik [mailto:MCH3@nrc.gov]
Sent: Friday, April 28, 2006 8:22 AM
To: Emerson, Frederick (GE Infra, Energy)
Cc: Conen, Joseph E. (GE Infra, Energy, Non-GE)
Subject: Draft RAIs for P-T Limits Report (SIR-05-044)

Please see attached.

Michelle C. Honcharik
NRR/DPR/PSPB
Project Manager
(301) 415-1774
M/S O-7E1A
mch3@nrc.gov

(See attached file: REQUEST FOR ADDITIONAL INFORMATION.doc)

REQUEST FOR ADDITIONAL INFORMATION

BOILING WATER REACTOR (BWR) OWNERS' GROUP

TOPICAL REPORT (TR) SIR-05-044, "PRESSURE TEMPERATURE LIMITS REPORT

METHODOLOGY FOR BOILING WATER REACTORS," REVISION 0

PROJECT NO. 691

All section, page, table, figure, or reference numbers in the questions below refer to items in TR SIR-05-044, unless specified otherwise.

1. The "*Requirements for Methodology and PTLR [Pressure Temperature Limit Report]*" table in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," identifies the minimum requirements to be included in the PTLR methodology and the minimum requirements to be included in the PTLR. Discuss how the proposed PTLR methodology and PTLR satisfy the minimum requirements identified in the GL 96-03 table. If the PTLR methodology or PTLR does not contain all the required information, revise the PTLR methodology and the PTLR to include the required information.

2. Section 2.5, "Pressure-Temperature Curve Generation Methodology," describes methodologies for calculating bending and membrane stresses using computer code finite element analyses (FEA). If these FEA are to be utilized by licensees to develop pressure-temperature (P-T) limits, provide the following:

    a. Identify the computer codes that were used in the finite element stress analysis. How were the codes benchmarked?

    b. Discuss briefly the assumptions and the inputs to the stress analysis.

    c. Update the TR methodology to require licensees to identify the finite element codes used in the PTLR.

    d. Verify that this process for determining bending and membrane stresses will result in the generation of P-T limits that are at least as conservative as those which would be generated using the procedures of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Appendix G.

3. Section 2.5.3, "Thermal and Pressure Stress Intensity Factor Calculations for Discontinuity Regions," indicates that the thermal stress intensity factor, K_{It}, for P-T limits for nozzles is dependent upon the membrane correction factor for an inside surface axial flaw and the thickness (t). The thickness term is not defined. Define the thickness to be used in determining the membrane correction factor for the K_{It} analysis for nozzles.

4. Section 2.5.3 indicates that the thermal stress intensity factor, K_{It}, for P-T limits for nozzles is dependent upon the a correction factor, R. This correction factor is used to correct the nonlinear effects in the plastic region based on the assumptions and recommendations of Welding Research Council (WRC) Bulletin 175, "PVRC [Pressure Vessel Research Committee] Recommendations on Toughness Requirements for Ferritic Materials." Describe how the methodology for analyzing nozzles (Equations 2.5.1-15 through 2.5.1-18) complies with WRC Bulletin 175.

5. Section 3.0, "Step-By-Step Procedure for Calculating P-T Limit Curves," indicates that P-T limits may also be developed for other reactor pressure vessel regions to provide additional operating flexibility. Either delete this statement from the PTLR methodology or provide the methodology for developing curves for the other regions and indicate that licensees will submit for review and approval methodologies for other regions that are not consistent with methodology discussed in the PTLR methodology.

6. Section 3.0 does not indicate surveillance data is to be evaluated in accordance with Appendix A, "Guidance for the Use of BWRVIP [BWR Vessel and Internals Project] ISP [Integrated Surveillance Program] Surveillance Data." Section 3.0 should be revised to indicate surveillance data is to be evaluated in accordance with Appendix A.

7. Pages A-8, A-9, and A-13 of Appendix A, state: "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."

   In order for this procedure to be utilized in the PTLR methodology, the staff must review the procedure for determining the best estimate chemistries for all beltline materials and the results from the data. Therefore, the PTLR methodology must be revised to document the BWRVIP procedure for determining the best estimate chemistries. If the best estimate chemistries are not performed in accordance with the approved procedure, then the PTLR methodology should indicate that the PTLR methodology will not be used in the PTLR process.

8. Appendix A, Procedure 1, Procedural Step 3, "Determine Credibility of Surveillance Data," states: "If the vessel wall temperature is an outlier, appropriate temperature adjustments to the surveillance data may be required."

   In order for this procedure to be utilized in the PTLR methodology, the staff must review the procedure for determining the adjustments to the surveillance data. Therefore, the PTLR methodology must be revised to document a proposed procedure for adjusting the surveillance data if the vessel wall temperature is an outlier. If the adjustments to the surveillance data are not performed in accordance with the approved procedure, then the PTLR methodology should indicate that the PTLR methodology will not be used in the PTLR process.

9. Appendix A, Procedures 1 and 2, "Definitions and Background," states: "For generic values [of Initial RT_{NDT}] of weld metal, the following generic mean values must be used unless justification for different values is provided..."

In order for other generic values of Initial RT_{NDT} to be utilized in the PTLR methodology, the staff must review the procedure for determining the best estimate Initial RT_{NDT}. Therefore, the PTLR methodology must be revised, either explicitly or by referencing a previously approved methodology, to document the BWRVIP procedure for determining the Initial RT_{NDT} . If the Initial RT_{NDT} are not performed in accordance with the approved procedure, then the PTLR methodology should indicate that the PTLR methodology will not be used in the PTLR process.

10. Appendix A, Procedure 1, Procedural Step 3, identifies information that the licensee should review to determine whether the data is "credible" or "not credible".

    In accordance with Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," the following criteria should also be evaluated:

    a. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30-foot-pound temperature and the upper-shelf energy unambiguously.

    b. When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values.

    These criteria should be added to Procedure 1, Procedural Step 3, of Appendix A.

11. To ensure that the P-T limits have been developed using the TR methodology, the following information should be included in the PTLR:

    a. The Initial RT_{NDT} for all reactor pressure vessel materials and the method of determining the Initial RT_{NDT} (i.e., ASME Code, Generic Communication, Branch Technical Position - MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, or other NRC-approved methodologies),

    b. The chemistry (weight-percent copper and nickel) and adjusted reference temperature at the 1/4 thickness location for all beltline materials, and

    c. The computer codes used in the FEA to determine for calculating bending and membrane stresses from Section 2.5.

    d. Identify whether "Procedure 1" or "Procedure 2" was utilized to evaluate the surveillance data. If surveillance data was utilized, provide the surveillance and the analysis of the surveillance data that was used to determine the adjusted reference temperature, ART. If surveillance data was not utilized, state why it was not utilized.

BWR OWNERS' GROUP

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Project Number 691

BWROG-06024
August 29, 2006

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Responses to Requests for Additional Information (RAIs) Dated May 3, 2006, Regarding the BWROG Topical Report SIR-05-044, "Pressure Temperature Limits Report Methodology For Boiling Water Reactors," Revision 0 (TAC NO. MC9694)

ENCLOSURE: Responses to RAIs

Dear Sir:

Enclosed please find the BWROG responses (Enclosure) to the NRC Request for Additional Information on the subject Topical Report (TR) SIR-05-044. NRC provided the RAIs for this report by letter dated May 3, 2006. We look forward to your timely review of these responses, and would be happy to meet with you to discuss any remaining issues.

Should you have additional questions please contact Fred Emerson (BWROG Project Manager) at 910-675-5615 or Steve Williams (BWROG PTC-SIA Committee Chairman) at 910-457-2318.

Sincerely,

RC Bunt

R. C. Bunt
BWR Owners' Group Chair

cc: D. Coleman, BWROG Vice Chair
Michelle Honcharik, NRC
BWROG Primary Representatives
BWROG PTC-SIA Committee
F. Emerson, GE

Enclosure

BWROG Response to NRC Request For Additional Information
Boiling Water Reactor (BWR) Owners' Group
Topical Report (TR) SIR-05-044, "Pressure Temperature Limits Report Methodology For Boiling Water Reactors," Revision 0

All section, page, table, figure, or reference numbers in the questions below refer to items in TR SIR-05-044, unless specified otherwise.

1. *The "Requirements for Methodology and PTLR [Pressure Temperature Limit Report]" table in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," identifies the minimum requirements to be included in the PTLR methodology and the minimum requirements to be included in the PTLR. Discuss how the proposed PTLR methodology and PTLR satisfy the minimum requirements identified in the GL 96-03 table. If the PTLR methodology or PTLR does not contain all the required information, revise the PTLR methodology and the PTLR to include the required information.*

RESPONSE:

The following will be added as a new first paragraph for Section 1.3 of the TR:

"The 'Requirements for Methodology and PTLR' table in GL 96-03 identifies the minimum requirements to be included in the PTLR methodology, and the minimum requirements to be included in the PTLR. Table 1-1 provides a summary of how the PTLR methodology included in this report satisfies the minimum requirements identified in the GL 96-03 table."

The table below will be added as new Table 1-1 of the TR.

Table 1-1: Summary of GL 96-03 PTLR Methodology Requirements

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	APPLICABLE SECTION OF LTR WHERE REQUIREMENTS ARE ADDRESSED
1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).	Describe transport calculation methods including computer codes and formulas used to calculate neutron fluence. Provide references	Provide the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.	Not covered by this LTR. Fluence methods and results must comply with RG 1.190 and have NRC approval for use with this LTR.
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.	Briefly describe the surveillance program. Licensee transmittal letter should identify by title and number report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. Topical/generic report contains placeholder only. Reference Appendix H to 10 CFR Part 50.	Provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located.	See Appendix A of Template PTLR included in Appendix B of this LTR.
3. Low temperature overpressure protection (LTOP) system limits developed using NRC-approved methodologies may be included in the PTLR.	Describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics. Reference SRP Section 5.2.2; ASME Code Case N-514; ASME Code, Appendix G, Section XI as applied in accordance with 10 CFR 50.55.	Provide setpoint curves or setpoint values.	Not applicable for BWRs.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.	Describe the method for calculating the ART using Regulatory Guide 1.99, Revision 2.	Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (t = vessel beltline thickness).	See Section 2.3 of this LTR.

Table 1-1: Summary of GL 96-03 PTLR Methodology Requirements (concluded)

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN STS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	APPLICABLE SECTION OF LTR WHERE REQUIREMENTS ARE ADDRESSED
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG- 0800, SRP Section 5.3.2, Pressure-Temperature Limits.	Describe the application of fracture mechanics in constructing P/T curves based on ASME Code, Appendix G, Section XI, and SRP Section 5.3.2.	Provide the P/T curves for heatup, cooldown, criticality, and hydrostatic and leak tests.	See Sections 2.3 and 2.4 of this LTR.
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.	Describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T curves.	Identify minimum temperatures on the P/T curves such as minimum boltup temperature and hydrotest temperature.	See Sections 2.7 and 2.8 of this LTR.
7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + $2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Describe how the data from multiple surveillance capsules are used in the ART calculation. Describe procedure if measured value exceeds predicted value. <u>WHEN OTHER PLANT DATA ARE USED</u> 1. Identify the source(s) of data when other plant data are used. 2.a Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability. OR 2.b Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation. Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.	See Section 2.3 of this LTR.

3

2. *Section 2.5, "Pressure-Temperature Curve Generation Methodology," describes methodologies for calculating bending and membrane stresses using computer code finite element analyses (FEA). If these FEA are to be utilized by licensees to develop pressure-temperature (P-T) limits, provide the following:*

    a. *Identify the computer codes that were used in the finite element stress analysis. How were the codes benchmarked?*

    b. *Discuss briefly the assumptions and the inputs to the stress analysis.*

    c. *Update the TR methodology to require licensees to identify the finite element codes used in the PTLR.*

    d. *Verify that this process for determining bending and membrane stresses will result in the generation of P-T limits that are at least as conservative as those which would be generated using the procedures of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, Appendix G.*

RESPONSE:

In response to RAI Items (a) through (c), the following text will be added to Section 2.5 of the TR (i.e., prior to Subsection 2.5.1):

"In the subsections that follow, finite element analysis is discussed as a possible approach for providing the necessary stress analysis for RPV regions. If finite element analysis is utilized to develop P-T limits for any RPV region, the following information shall be provided in the PTLR:

a. Identify the computer code(s) that were used in the finite element stress analysis.

b. For any computer codes used, describe how the code(s) were verified or benchmarked. Computer code verification shall be in accordance with a qualified 10 CFR 50 Appendix B Quality Assurance Program. As a part of computer code verification, benchmarking consistent with NRC GL 83-11, Supplement 1 [17] shall be included.

c. Identify the assumptions and the inputs to the finite element analysis. Necessary inputs to the analysis include any or all of the following:
    - A description of plant operating conditions used (e.g., pressure and temperature). The conditions used must represent current plant operating conditions.
    - A description of the heat transfer coefficients used and the methodology used to calculate them.
    - A description of the model developed, including materials, material properties, finite element mesh pattern, and geometry."

New Reference 17 (references will be re-numbered, as identified in the response to RAI No. 3 below) will be added to Section 4.0 of the TR as follows:

"17. U. S. Nuclear Regulatory Commission, Generic Letter 88-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," June 24, 1999."

4

For Item (d), refer to the response to RAI No. 3 below, where the linearization techniques have been removed and replaced with polynomial fit techniques that are consistent with current ASME Code, Section XI, Appendix G methodology.

3. *Section 2.5.3, "Thermal and Pressure Stress Intensity Factor Calculations for Discontinuity Regions," indicates that the thermal stress intensity factor, K_{It}, for P-T limits for nozzles is dependent upon the membrane correction factor for an inside surface axial flaw and the thickness (t). The thickness term is not defined. Define the thickness to be used in determining the membrane correction factor for the K_{It} analysis for nozzles.*

RESPONSE:

Starting on page 2-20 of the TR, replace the entire "Non-Beltline Region" section with the following. This replacement text is considered to provide further detail and clarification that responds to the RAI [quotation below concludes on page 8]:

<u>"Non-Beltline Region</u>

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 $1x10^{17}$ 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 some cases, such results may already be 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, as shown in Figure 2-7, the thermal and pressure hoop stress distributions should be extracted from the finite element model. Each of the stress distributions should be fit with a third-order polynomial that reasonably fits the calculated stresses in the region of interest.

The thermal stress intensity factor, K_{It}, is computed based on the nozzle corner solution shown in Figure 2-8 for a postulated 1/4t (based on the section thickness) axial defect, as follows:

$$K_{It} = \sqrt{\pi a}\left[0.706\,C_{0t} + 0.537\left(\frac{2a}{\pi}\right)C_{1t} + 0.448\left(\frac{a^2}{2}\right)C_{2t} + 0.393\left(\frac{4a^3}{3\pi}\right)C_{3t}\right] \qquad (2.5.1\text{-}15)$$

where:	K_{It}	=	the thermal stress intensity factor for the limiting normal/upset transient (ksi$\sqrt{\text{inch}}$)
	a	=	1/4t postulated flaw depth (inches)

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t	=	thickness of the cross-section through the limiting nozzle inner blend radius corner, as shown in Figure 2-7.
$C_{0t}, C_{1t}, C_{2t}, C_{3t}$	=	thermal stress polynomial coefficients based on fits to finite element analysis.

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 $\sqrt{\text{inch}}$)
K_{Ic}	=	the lower bound of static fracture toughness as a function of the coolant temperature, T, and the limiting RT_{NDT} for all 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 based on the nozzle corner solution shown in Figure 2-8 for a postulated 1/4t (based on the section thickness) axial defect, as follows:

$$K_{Ip-applied} = \sqrt{\pi a}\left[0.706\,C_{0p} + 0.537\left(\frac{2a}{\pi}\right)C_{1p} + 0.448\left(\frac{a^2}{2}\right)C_{2p} + 0.393\left(\frac{4a^3}{3\pi}\right)C_{3p}\right] \qquad (2.5.1\text{-}17)$$

where:

$K_{Ip\text{-}applied}$	=	the applied pressure stress intensity factor (ksi $\sqrt{\text{inch}}$)
a	=	1/4t postulated flaw depth (inches)
t	=	thickness of the cross-section through the limiting nozzle inner blend radius corner, as shown in Figure 2-7.
$C_{0p}, C_{1p}, C_{2p}, C_{3p}$	=	pressure stress polynomial coefficients based on fits to finite element analysis.

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 $\sqrt{\text{inch}}$)"

The figures below will be added as new Figures 2-7 and 2-8 of the TR.

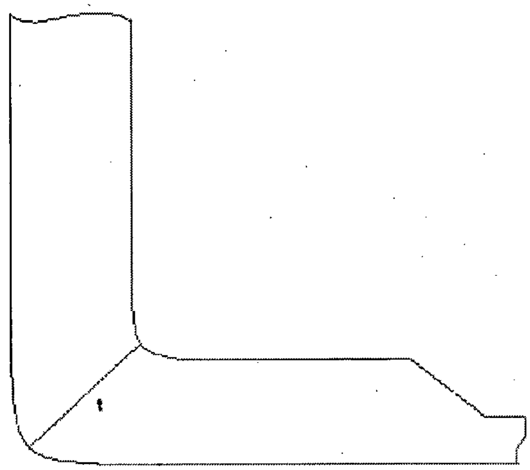

Figure 2-7: Nozzle Thickness Definition

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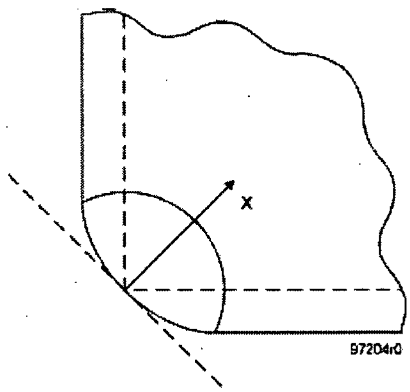

SIMULATED 3-D NOZZLE CORNER CRACK

Figure 2-8: Stress Intensity Factor Solution for a Nozzle Corner Crack

Consistent with the above changes, the following changes will be made to replace the text starting with the paragraph at the bottom of page 2-13 of the TR that begins, "The secondary linear bending ($\square_{sb}$) and constant secondary membrane ($\square_{sm}$) stress...", continuing on page 2-14 through the end of the "Closed Form Solution Method":

"The thermal stress intensity factor, K_{It}, for the thermal hoop stress distribution calculated from Equation 2.5.1-4 can be calculated at any specified time during the cooldown for a 1/4t inside surface defect using the following relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3)\sqrt{\pi a} \qquad (2.5.1\text{-}6)$$

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:

$$\square(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \qquad (2.5.1\text{-}7)$$

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

a = the maximum crack depth (inches)"

Finally, with the above changes, existing Equations 2.5.1-7 through 2.5.1-18 of the TR will be renumbered accordingly, Reference [14] will be deleted, and the remaining references will be renumbered accordingly.

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4. *Section 2.5.3 indicates that the thermal stress intensity factor, KIt, for P-T limits for nozzles is dependent upon the correction factor, R. This correction factor is used to correct the nonlinear effects in the plastic region based on the assumptions and recommendations of Welding Research Council (WRC) Bulletin 175, "PVRC [Pressure Vessel Research Committee] Recommendations on Toughness Requirements for Ferritic Materials." Describe how the methodology for analyzing nozzles (Equations 2.5.1-15 through 2.5.1-18) complies with WRC Bulletin 175.*

RESPONSE:

The following identifies how each of Equations 2.5.1-15 through 2.5.1-18 of the TR complies with WRC Bulletin 175:

Equation 2.5.1-15 is derived from Equation 5-4 in WRC Bulletin 175.

Equation 2.5.1-16 is derived from Equation 4-4 in WRC Bulletin 175, with incorporation of the safety factors discussed in Section 4.E of WRC Bulletin 175.

Equation 2.5.1-17 is derived from Equation A5-1 in Appendix 5 of WRC Bulletin 175.

Equation 2.5.1-18 calculates the allowable pressure as a ratio of the previously calculated parameters. Since the operating pressure, P, is directly proportional to $K_{Ip\text{-}applied}$ (from Equation 2.5.1-17), it follows that the allowable pressure, P_{allow}, is directly proportional to the allowable pressure stress intensity, K_{Ip} (as calculated in Equation 2.5.1-16).

5. *Section 3.0, "Step-By-Step Procedure for Calculating P-T Limit Curves," indicates that P-T limits may also be developed for other reactor pressure vessel regions to provide additional operating flexibility. Either delete this statement from the PTLR methodology or provide the methodology for developing curves for the other regions and indicate that licensees will submit for review and approval methodologies for other regions that are not consistent with methodology discussed in the PTLR methodology.*

RESPONSE:

The sentence of the TR in question will be revised to state:

"P-T limit curves may also be developed for other RPV regions to provide additional operating flexibility; however, for RPV regions other than those defined in Section 2.0 of this report, licensees are required to submit methodologies to the NRC for review and approval prior to use."

6. *Section 3.0 does not indicate surveillance data is to be evaluated in accordance with Appendix A, "Guidance for the Use of BWRVIP [BWR Vessel and Internals Project] ISP [Integrated Surveillance Program] Surveillance Data." Section 3.0 should be revised to indicate surveillance data is to be evaluated in accordance with Appendix A.*

RESPONSE:

A new Step (a) will be added to Section 3.0 of the TR as follows:

"a. Evaluate surveillance data in accordance with Appendix A of this report."

The previously defined steps will be re-labeled as Steps (b) through (i).

7. *Pages A-8, A-9, and A-13 of Appendix A, state: "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."*

   In order for this procedure to be utilized in the PTLR methodology, the staff must review the procedure for determining the best estimate chemistries for all beltline materials and the results from the data. Therefore, the PTLR methodology must be revised to document the BWRVIP procedure for determining the best estimate chemistries. If the best estimate chemistries are not performed in accordance with the approved procedure, then the PTLR methodology should indicate that the PTLR methodology will not be used in the PTLR process.

RESPONSE:

The note on pages A-8, A-9, and A-13 of the TR will be revised to the following:

"Note: Revised best estimate chemistries for selected BWR vessel and surveillance capsule materials have been calculated by the BWRVIP, as documented in BWRVIP-86-A [A-1]. Calculation of the best estimate chemistries for all other vessel materials should be determined in accordance with the NRC practice documented in Reference [A-7]. The suggested practice is documented in guidelines contained in BWRVIP-135. This evaluation is the responsibility of the plant, must be described in the PTLR, and must utilize NRC-approved methods."

New Reference A-7 will be added to Section A.5 of the TR as follows:

"A-7. "Generic Letter 92-01 and RPV Integrity Assessment – Status, Schedule, and Issues," Presentation by K. Wichman, M. Mitchell, and A. Hiser at NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998."

8. *Appendix A, Procedure 1, Procedural Step 3, "Determine Credibility of Surveillance Data," states: "If the vessel wall temperature is an outlier, appropriate temperature adjustments to the surveillance data may be required."*

   In order for this procedure to be utilized in the PTLR methodology, the staff must review the procedure for determining the adjustments to the surveillance data. Therefore, the PTLR methodology must be revised to document a proposed procedure for adjusting the surveillance data if the vessel wall temperature is an outlier. If the adjustments to the surveillance data are not performed in accordance with the approved procedure, then the PTLR methodology should indicate that the PTLR methodology will not be used in the PTLR process.

RESPONSE:

Appendix A, Procedure 1, Procedural Step 3(b) of the TR will be revised as follows:

> "b. If the vessel wall temperature is an outlier, appropriate temperature adjustments to the surveillance data may be required. An appropriate temperature adjustment is a 1°F degree increase in □RT_{NDT} per 1°F decrease in irradiation temperature [A-7]. Alternatively, the temperature adjustment can be determined using appropriate NRC guidance. Any temperature adjustments shall be identified and described in the PTLR."

9. *Appendix A, Procedures 1 and 2, "Definitions and Background," states: "For generic values [of Initial RT_{NDT}] of weld metal, the following generic mean values must be used unless justification for different values is provided..."*

   In order for other generic values of Initial RT_{NDT} to be utilized in the PTLR methodology, the staff must review the procedure for determining the best estimate Initial RT_{NDT}. Therefore, the PTLR methodology must be revised, either explicitly or by referencing a previously approved methodology, to document the BWRVIP procedure for determining the Initial RT_{NDT}. If the Initial RT_{NDT} are not performed in accordance with the approved procedure, then the PTLR methodology should indicate that the PTLR methodology will not be used in the PTLR process.

RESPONSE:

The two paragraphs of the TR (pages A-5 and A-11) noted in the RAI will be revised as follows:

"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: 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]. Other generic mean values may be used, provided they are justified and have NRC review and approval. The generic mean values used shall be identified in the PTLR."

10. *Appendix A, Procedure 1, Procedural Step 3, identifies information that the licensee should review to determine whether the data is "credible" or "not credible".*

In accordance with Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," the following criteria should also be evaluated:

a. *Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30-foot-pound temperature and the upper-shelf energy unambiguously.*

b. *When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values.*

These criteria should be added to Procedure 1, Procedural Step 3, of Appendix A.

RESPONSE:

The following two steps will be added to 10. Appendix A, Procedure 1, Procedural Step 3 of the TR:

"d. Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 foot-pound temperature and the upper shelf energy unambiguously.

e. When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Reg. Guide 1.99 Rev. 2, Regulatory Position 2.1, normally should be less than 28°F for welds and 17°F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E185-82."

11. *To ensure that the P-T limits have been developed using the TR methodology, the following information should be included in the PTLR:*

    a. *The Initial RT_{NDT} for all reactor pressure vessel materials and the method of determining the Initial RT_{NDT} (i.e., ASME Code, Generic Communication, Branch Technical Position - MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, or other NRC-approved methodologies),*

    b. *The chemistry (weight-percent copper and nickel) and adjusted reference temperature at the 1/4 thickness location for all beltline materials, and*

    c. *The computer codes used in the FEA to determine for calculating bending and membrane stresses from Section 2.5.*

    d. *Identify whether "Procedure 1" or "Procedure 2" was utilized to evaluate the surveillance data. If surveillance data was utilized, provide the surveillance and the analysis of the surveillance data that was used to determine the adjusted reference temperature, ART. If surveillance data was not utilized, state why it was not utilized.*

RESPONSE:

The following will be added to the end of Section 2.3 of the TR to address items (a), (b), and (d) of the RAI. For Item (c), refer to the response to RAI No. 2:

"The following information should be included in the PTLR with respect to the ART calculations:

a. The IRT_{NDT} for all RPV materials and the method of determining the IRT_{NDT} (i.e., ASME Code, Generic Communication, Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, or other NRC-approved methodologies).

b. The chemistry (weight-percent copper and nickel) and ART at the 1/4t location for all beltline materials.

c. Identify whether "Procedure 1" or "Procedure 2" from Appendix A was utilized to evaluate the surveillance data. If surveillance data was utilized, provide the surveillance and the analysis of the surveillance data that was used to determine the ART values. If surveillance data was not utilized, state why it was not utilized."

The following will be added to Section 5.0, Discussion, of the Template PTLR in Appendix B of the TR to address the four items requested in the RAI:

"The initial RT_{NDT}, the chemistry (weight-percent copper and nickel) and adjusted reference temperature at the 1/4 thickness location for all RPV beltline materials significantly affected by fluence (i.e., fluence > 10^{17} n/cm^2 for E > 1 MeV) are shown in Table 7 for SSES-1 and Table 8 for SSES-2. The initial RT_{NDT} values shown in Tables 7 and 8 were developed using the procedures of Branch Technical Position MTEB 5-2 in Standard Review Plan 5.3.2 in NUREG-0800, and they have been previously approved for use by the NRC [6-6].

For SSES-1, limiting RPV plate C-2433-1, BWRVIP "Procedure 1" was utilized since the heat number of this material is identical to the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two

or more credible data sets available for this material. For limiting RPV weld 494K2351, BWRVIP "Procedure 2" was utilized since the heat number of this material is different than the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. Therefore, Regulatory Guide 1.99, Revision 2 chemistry factors were used in the determination of the ART values for all materials for SSES-1.

For SSES-2, limiting RPV plate C-2421-3, BWRVIP "Procedure 2" was utilized since the heat number of this material is different than the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. For limiting RPV weld 624263, BWRVIP "Procedure 2" was utilized since the heat number of this material is different than the heat number of the BWRVIP ISP Representative Material. Surveillance data was not used in the evaluation procedure since there are not yet two or more credible data sets available for this material. Therefore, Regulatory Guide 1.99, Revision 2 chemistry factors were used in the determination of the ART values for all materials for SSES-2.

The only computer code used in the determination of the SSES P-T curves was the ANSYS (Version 4.4) finite element computer program for the feedwater nozzle (non-beltline) stresses. This program was controlled under the vendor's 10 CFR 50 Appendix B Quality Assurance Program for nuclear quality-related work. Benchmarking consistent with NRC GL 88-13, Supplement 1 was performed as a part of the computer program verification by comparing the solutions produced by the computer code to hand calculations for several problems. The following inputs were used as input to the finite element analysis *[Editorial note: The following items must be included on a plant-specific basis]*:

- *Plant operating conditions must be listed here. These conditions represent current plant operating conditions.*
- *Heat transfer coefficients must be listed here. These values were developed using conventional heat transfer methods for forced convection flow on a vertical flat plate.*
- *A description of the finite element model must be listed here, including materials, material properties, finite element mesh pattern, and geometry."*

The following reference will be added to the Template PTLR:

"6.6 NRC approval letter for IRT_{NDT} values. [Editorial note: The appropriate plant-specific reference is to be included here.]"

The tables below will be added as new Tables 7 and 8 in the Template PTLR.

Table 7: SSES-1 ART Calculations for 32 EFPY

Part Name & Material	ID No.	Heat No.	Lot No.	Estimated Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4t: ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_I (°F)	EFPY	ART_{NDT} (°F)
Lower Shell #1	21-1	B5083-1	—	-8	0.14	0.48	94.6	28.2	14.1	0.0	32.0	48.5
Lower Shell #2	21-2	C0770-2	—	-20	0.14	0.50	95.5	28.5	14.2	0.0	32.0	37.0
Lower Shell #3	21-3	C0314-2	—	-20	0.13	0.51	88.3	26.4	13.2	0.0	32.0	32.7
Lower-Int. Shell #1	22-1	C0803-1	—	-10	0.09	0.53	58.0	19.3	9.6	0.0	32.0	28.6
Lower-Int. Shell #2	22-2	C0776-1	—	6	0.12	0.48	80.6	26.8	13.4	0.0	32.0	59.6
Lower-Int. Shell #3	22-3	C2433-1	—	18	0.10	0.63	65.3	21.7	10.9	0.0	32.0	61.4
Weld #1	—	629516	L320A27AG	-50	0.04	0.99	54.0	18.0	9.0	0.0	32.0	-14.1
Weld #2	—	411L3071	L311A27AF	-50	0.03	0.93	41.0	13.6	6.8	0.0	32.0	-22.7
Weld #3	—	494K2351	L307A27AD	-50	0.04	1.10	54.0	18.0	9.0	0.0	32.0	-14.1

Fluence Information:

Location	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4t	EFPY	Fluence at ID (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10 \log f)}$
Lower Shell #1	6.160	1.540	32.00	7.50E+17	0.691	5.18E+17	0.298
Lower Shell #2	6.160	1.540	32.00	7.50E+17	0.691	5.18E+17	0.298
Lower Shell #3	6.160	1.540	32.00	7.50E+17	0.691	5.18E+17	0.298
Lower-Int. Shell #1	6.160	1.540	32.00	9.20E+17	0.691	6.36E+17	0.332
Lower-Int. Shell #2	6.160	1.540	32.00	9.20E+17	0.691	6.36E+17	0.332
Lower-Int. Shell #3	6.160	1.540	32.00	9.20E+17	0.691	6.36E+17	0.332
Weld #1	6.160	1.540	32.00	9.20E+17	0.691	6.36E+17	0.332
Weld #2	6.160	1.540	32.00	9.20E+17	0.691	6.36E+17	0.332
Weld #3	6.160	1.540	32.00	9.20E+17	0.691	6.36E+17	0.332

Notes: 1. Material and fluence information taken from GE Report No. GE-NE-523-169-1292, "Susquehanna Steam Electric Station Unit 1 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," March 1993, Table 7-3.
(Note that Table 7-3 has a typographical error for Heat No. C0603-1; refer to NRC RVID2 database and Table 3-1 of the GE Report.)
2. The calculations shown in this table are not for design use, as they utilize outdated fluence results. These calculations are for comparison purposes only.

Table 8: SSES-2 ART Calculations for 32 EFPY

Part Name & Material	ID No.	Heat No.	Lot No.	Estimated Initial RT_{NDT} (°F)	Chemistry Cu (wt %)	Chemistry Ni (wt %)	Chemistry Factor (°F)	Adjustments For 1/4t ΔRT_{NDT} (°F)	Margin Terms σ_Δ (°F)	Margin Terms σ_I (°F)	EFPY	ART_{NDT} (°F)
Lower Shell #1	21-1	6C956-1-1	—	-20	0.11	0.55	73.5	20.1	10.1	0.0	32.0	20.2
Lower Shell #2	21-2	6C980-1-1	—	-20	0.10	0.56	65.0	17.8	8.9	0.0	32.0	15.6
Lower Shell #3	21-3	6C1053-1-1	—	10	0.10	0.58	65.0	17.8	8.9	0.0	32.0	45.6
Lower-Int. Shell #1	22-1	C2421-3	—	-10	0.13	0.66	93.0	28.3	14.2	0.0	32.0	46.7
Lower-Int. Shell #2	22-2	C2929-1	—	-20	0.13	0.64	92.0	28.0	14.0	0.0	32.0	36.1
Lower-Int. Shell #3	22-3	C2433-2	—	2	0.10	0.63	65.3	19.9	10.0	0.0	32.0	41.8
Weld #1	—	629616	L320A27AG	-50	0.04	0.59	54.0	16.5	8.2	0.0	32.0	-17.1
Weld #2	—	624263	E204A27A	-20	0.06	0.89	82.0	25.0	12.5	0.0	32.0	30.0
Weld #3	—	09M057	C109A27A	-36	0.03	0.89	41.0	12.5	6.2	0.0	32.0	-11.0

Fluence Information:

Location	Wall Thickness (inches) Full	Wall Thickness (inches) 1/4t	EFPY	Fluence at ID (n/cm^2)	Attenuation, 1/4t $e^{-0.24x}$	Fluence @ 1/4t (n/cm^2)	Fluence Factor, FF $f^{(0.28-0.10\log f)}$
Lower Shell #1	6.160	1.540	32.00	6.40E+17	0.691	4.42E+17	0.274
Lower Shell #2	6.160	1.540	32.00	6.40E+17	0.691	4.42E+17	0.274
Lower Shell #3	6.160	1.540	32.00	6.40E+17	0.691	4.42E+17	0.274
Lower-Int. Shell #1	6.160	1.540	32.00	7.80E+17	0.691	5.39E+17	0.305
Lower-Int. Shell #2	6.160	1.540	32.00	7.80E+17	0.691	5.39E+17	0.305
Lower-Int. Shell #3	6.160	1.540	32.00	7.80E+17	0.691	5.39E+17	0.305
Weld #1	6.160	1.540	32.00	7.80E+17	0.691	5.39E+17	0.305
Weld #2	6.160	1.540	32.00	7.80E+17	0.691	5.39E+17	0.305
Weld #3	6.160	1.540	32.00	7.80E+17	0.691	5.39E+17	0.305

Notes: 1. Material and fluence information taken from GE Report No. GE-NE-523-107-0893, Revision 1, "Susquehanna Steam Electric Station Unit 2 Vessel Surveillance Materials Testing and Fracture Toughness Analysis," October 1993, Table 7-3.
(Note that Table 7-3 has a typographical error for Heat No. 6C065-1-1; refer to NRC RVID2 database and Table 3-1 of the GE Report.)
2. The calculations shown in this table are not for design use, as they utilize outdated fluence results. These calculations are for comparison purposes only.

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