BWR OWNERS' GROUP

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Project Number 691

BWROG-06024
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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
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D044

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□□) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in RT_{NDT} + 2□□), 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.

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."

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)

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:

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

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.

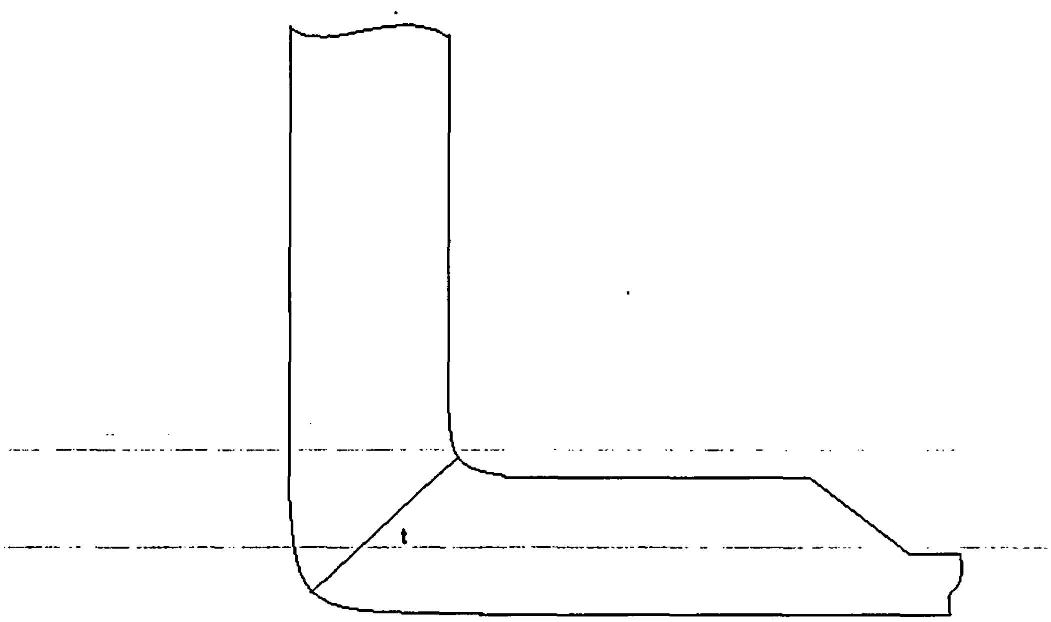
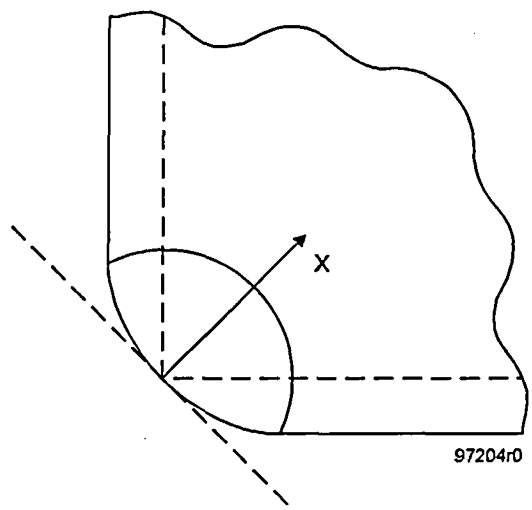


Figure 2-7: Nozzle Thickness Definition



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.

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	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.