DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT NEDC-33178P

"GENERAL ELECTRIC METHODOLOGY FOR DEVELOPMENT OF REACTOR PRESSURE

VESSEL PRESSURE-TEMPERATURE CURVES"

BOILING WATER REACTORS OWNERS' GROUP

PROJECT NO. 691

1.0 INTRODUCTION AND BACKGROUND

By letter dated July 28, 2006, the Boiling Water Reactor Owners' Group (BWROG) submitted Licensing Topical Report (LTR) NEDC-33178P, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 0 (Reference 1), for the Nuclear Regulatory Commission (NRC) review and acceptance for referencing in subsequent licensing actions. The BWROG provided this LTR to support applications by BWR licensees to relocate their pressure-temperature (P-T) curves 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 2). Responses to NRC staff's requests for additional information (RAIs) were provided in a BWROG letter dated July 31, 2007 (Reference 3), which was later superseded by a revised version of the LTR. LTR NEDC-33178P, Revision 1, incorporating the proposed changes was provided to the NRC in a letter dated January 19, 2009 (Reference 4).

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 Title 10, *Code of Federal Regulations* Part 50 (10 CFR Part 50, Appendix G; Reference 5), 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 Boiler and Pressure Vessel Code (ASME Code, Section XI, Appendix G; Reference 6), 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 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, 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 7), and NUREG-0800, Standard Review Plan (SRP), Section 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock" (Reference 8).

The regulation at 10 CFR Part 50, Appendix H (Reference 9), 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 issued RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Reference 10). Neutron fluence calculations are acceptable if they are performed 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 11). This 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:

> Criterion 2: 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 BWRs fall within the scope of Criterion 2 of 10 CFR 50.36(c)(2)(ii) and were therefore 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 and/or low temperature over-pressure protection (LTOP) limit setpoint values 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 limit curves and LTOP limits for U.S. licensed light-water reactors would need to be generated in accordance with an NRC-approved methodology and that the methodology to generate the P-T limit curves and LTOP 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.

Technical Specification Task Force (TSTF) Traveler No. TSTF-419, “Revise PTLR Definition and References in ISTS [Improved Standard Technical Specifications] 5.6.6, RCS PTLR” (Reference 12) amended the Standard Technical Specifications (STS) (NUREGs-1430, -1431, -1432, -1433, and -1434) by: (1) deleting references to the TS LCO specifications for the P-T limits and LTOP system limits in the TS definition of the PTLR, and (2) revising STS 5.6.6 to identify, by number and title, NRC-approved topical reports that document PTLR methodologies, or the NRC SE 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 and LTOP system limits are relocated to the PTLR. The methodology for their development must be reviewed and approved by the NRC. 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 13).

3.0 TECHNICAL EVALUATION

As stated in Section 2.1 of this SE, 10 CFR Part 50, Appendix G requires that licensees establish limits on the pressure and temperature of the RCPB to protect it against brittle failure. These limits are defined by P-T limit curves for normal operations, including heatup and cooldown operations of the reactor coolant system (RCS) and system hydrostatic tests.

BWROG LTR NEDC-33178P, Revision 0 has six sections, nine appendices and two attachments. Section 1.0 provides the introduction and the purpose for the LTR. Section 2.0 provides the scope of the analysis, and Section 3.0 refers to Attachment 1 for the assumptions for the plant-specific P-T analysis. Section 4.0 describes the analysis methods for developing P-T limits. Section 5.0 provides conclusions and recommendations and Section 6.0 provides references. Attachment 1 provides an example of a P-T curve report template. Attachment 2 provides an example of a PTLR. Appendices A through H provide background information used for performing the analyses described in Section 4.0 of the LTR. Appendix I provides guidance for evaluating surveillance data.

3.1 Evaluation of Section 4.0 of the LTR

The NRC staff’s evaluation of Sections 4.1 through 4.3 of the LTR 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 for PTLRs acceptable to the NRC staff. The NRC staff’s evaluations of the contents of BWROG methodology against the seven criteria in Attachment 1 of GL 96-03 are given below.

GL 96-03, Attachment 1 Methodology Criterion 1

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

The GL 96-03 conformance table in the BWROG's November 15, 2007, letter indicates this LTR does not describe the transport calculation methods including computer codes and formulas used to calculate neutron fluence. However, Section 4.2.1.2 of the LTR indicates that the neutron fluence will be determined using an approved methodology consistent with RG 1.190. Further, this section indicates the neutron fluence is defined in Section 4.2.1.2 of Attachment 1 and Appendix B of the PTLR. Section 4.2.1.2 of Attachment 1 requires the licensee to identify the report used to calculate the neutron fluence and to document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology. Therefore, this will be a plant-specific action item to be addressed by licensees. Since the LTR methodology indicates that the neutron fluence calculation methodology must comply with RG 1.190 and have been approved by the NRC, this criterion has been satisfied.

GL 96-03, Attachment 1 Methodology Criterion 2

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

The GL 96-03 conformance table in the BWROG's November 15, 2007, letter indicates this information is in Section 4.2.2 of the LTR. This section indicates that the BWR integrated surveillance program is applicable to each BWR reactor vessel and is described in the BWRVIP-102 report, "BWR Vessel and Internals Project Integrated Surveillance Program Implementation Guidelines," and the BWRVIP-135 report, "BWR Vessels and Internals Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations." Since these BWRVIP reports describe the BWR integrated surveillance program, this criterion has been satisfied.

GL 96-03, Attachment 1 Methodology Criterion 3

Methodology Criterion 3 requires that the methodology describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics.

This methodology does not need to address this criterion since it only applies to pressurized water reactors (PWRs) and the methodology applies to BWRs.

GL 96-03, Attachment 1 Methodology Criterion 4

Methodology Criterion 4 requires that the methodology describe the method for calculating the adjusted reference temperature (ART) using RG 1.99, Revision 2.

Sections 4.1 and 4.2 describe the method for determining the material properties for reactor vessel beltline and non-beltline region materials. The unirradiated reference temperature (initial RT_{NDT}) is determined using the method described in ASME Code, Section III, Subsection NB-2300, where sufficient data is available. If insufficient data is available, the initial RT_{NDT} is determined using the methodology described in GENE NEDC-32399-P (Reference 14). This methodology was approved by the NRC on December 16, 1994 (Reference 15). The ART, an indirect measure of the RPV material fracture toughness, is determined using the methodology described in RG 1.99, Revision 2. The ART is defined in the RG as the sum of the initial RT_{NDT}, the shift in reference temperature caused by irradiation (ΔRT_{NDT}), and the margin term.

Section 4.1.2, "Values of Initial RT_{NDT} and Lowest Service Temperature (LST)," indicates:

> Where the lowest energy Charpy value is less than 50 ft-lb, it is adjusted by adding 2 °F per ft-lb energy difference from 50 ft-lb. If the test specimens are transverse and the lowest energy Charpy value is less than 50 ft-lb, it is adjusted by adding 3 °F per ft-lb energy difference from 50 ft-lbs.

The NRC staff noted that the second sentence in the above statement is inconsistent with the example that follows in this section.

The response to NRC staff RAI 10 in the BWROG's November 15, 2007, letter indicated the following:

> The example presented represents only the longitudinal specimen method. To further clarify, a second example for the plate material will be added to this section to demonstrate the methodology for a transverse specimen. The methodology used for the transverse specimens is consistent with that methodology defined in NEDC-32399P; this additional process was added to account for older plants, where all of the ASME Code requirements were not met.

In the same response, the BWROG proposed to add the following to clarify this section of the LTR:

> A second example, for a plate material based upon transverse specimens, is seen below.
>
> The lowest Charpy energy and test temperature from the CMTRs [Certified Material Test Reports] are 47 ft-lb and 13 °F. The estimated transverse 50 ft-lb test temperature is:
>
> $$T_{50T} = 10\ °F + [(50 - 47)\ \text{ft-lb}^*\ 3\ °F/\text{ft-lb}] = 19\ °F$$
>
> The initial RT_{NDT} is the greater of NDT [nil-ductility temperature] or (T_{50T} - 60 °F).
>
> $$T_{50T} - 60\ °F = 19\ °F - 60\ °F = -41\ °F$$
>
> Dropweight testing to establish NDT for plate material is listed in the CMTR; the NDT for this material is -20 °F. Therefore, the initial RT_{NDT} for this plate heat is -20 °F.

Since the proposed change to the LTR is consistent with the methodology approved by the NRC staff, it is acceptable. The NRC staff verified that the change has been implemented in the revised LTR dated December 2008.

The margin term that is defined in RG 1.99, Revision 2, is dependent upon the standard deviation for the initial RT_{NDT} (σ_I) and the standard deviation for the ΔRT_{NDT} (σ_Δ). Section 4.2.1 of the LTR indicates:

> The margin term σ_Δ as described above, is defined in RG 1.99: this methodology is used except when Integrated Surveillance Program data from BWRVIP-135…is available, and BWRVIP-102…methods are applied.

In response to NRC staff RAI 11, the BWROG indicated the following in its November 15, 2007, letter:

> This statement was intended to indicate that RG 1.99 is to be used to determine the margin term, and that the procedures of BWRVIP-102 should be followed to incorporate surveillance data from the ISP.

In the same response, the BWROG proposed to add the following to clarify this section of the LTR:

> The margin term σ_Δ, as described above, is defined in RG 1.99. When Integrated Surveillance Program data from BWRVIP-135…are available, BWRVIP-102…provides guidance with respect to applying the requirements of RG 1.99 to this data. Appendix I of this report also contains guidance regarding the application of surveillance data.

The proposed change and explanation provide the necessary clarification to ensure that RG 1.99, Revision 2, is properly applied by licensees. The NRC staff verified that the change has been implemented in the revised LTR dated December 2008.

Since the GL 96-03 conformance table in the BWROG's November 15, 2007, letter indicates the information for calculating the ART is in Section 4.2 of the LTR, and this section describes the methodology documented in RG 1.99, Revision 2, this criterion has been satisfied.

GL 96-03, Attachment 1 Methodology Criterion 5

Methodology 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.

The GL 96-03 conformance table in the BWROG's November 15, 2007, letter indicates this information is in Section 4.3 of the LTR. This section of the report describes the methodology for developing P-T curves for the lower vessel region, the upper vessel region, the core beltline region, and the closure flange region of the RPV.

The lower vessel region analyses evaluate the materials in the bottom head, control rod drive (CRD) penetrations and the nozzles, skirt, attachments to the bottom head (Tables 4-5a and 4-5b in the LTR). The bottom head analysis for pressure and leak test conditions is described in Section 4.3.2.1.1 of the LTR. The limit for the coolant temperature rate change for the pressure and leak test is 20 °F/hour. The bottom head analysis for core not critical heatup/cooldown conditions is described in Section 4.3.2.1.2 of the LTR. The core not critical P-T limit curves were developed based on assumed 100 °F/hour heatup/cooldown rates and bounding bottom head transients defined on the plant-specific RPV thermal cycle and nozzle thermal cycle diagrams.

The upper vessel region analyses evaluate materials in the upper shell, closure flange and the nozzles, skirts and attachments to the upper shell (Tables 4-4a and 4-4b in the LTR). The upper vessel region analysis for pressure and leak test conditions is described in Section 4.3.2.1.3 of the LTR. The upper shell region analysis for core not critical heatup/cooldown conditions is described in Section 4.3.2.1.4 of the LTR. The core not critical P-T limit curves were developed from the bounding feedwater transients defined on the plant-specific RPV thermal cycle and nozzle thermal cycle diagrams.

The core beltline region analyses evaluate materials in the shell region of the RPV that are adjacent to the active fuel, such that the neutron fluence is sufficient to cause a significant shift of the RT_{NDT} (i.e., at a neutron fluence exceeding 1.0E17 n/cm^2 (E>1MeV)). The beltline region analysis for pressure and leak test conditions is described in Sections 4.3.2.2.1 and 4.3.2.2.2 of the LTR. The beltline region analysis for core not critical heatup/cooldown conditions is described in Sections 4.3.2.2.3 and 4.3.2.2.4 of the LTR. Appendix E describes the method to determine whether there are any RPV discontinuities be included in the beltline region.

The closure flange region analyses evaluate materials in the top head and shell closure flanges in the RPV. The closure flange region analysis for pressure and leak test conditions and the closure flange region analysis for core not critical heatup/cooldown conditions are described in Section 4.3.2.3 of the LTR.

In addition to the fracture mechanics analyses documented in Section 4.3, Appendices F, G, and H contain detailed fracture mechanics analyses for various RPV regions. Appendix F describes the supplemental analysis performed for nozzles that are within the beltline region. Appendix G provides supplemental analyses for thickness transition discontinuities between the bottom head, lower torus and the upper torus and thickness transition discontinuities in the beltline region. Appendix H provides a supplemental analysis for the bottom head CRD penetrations.

The analyses described in Section 4.3 and Appendices F, G, and H of the LTR were performed to satisfy the requirements in 10 CFR Part 50, Appendix G and Appendix G to Section XI of the ASME Code. The Edition and Addenda of ASME Code, Section XI used in the plant-specific evaluation will be specified in the plant-specific report provided to the licensee and the PTLR. The methodology includes the following: 1) the use of K_{Ic} from Figure A-4200-1 of Appendix A to ASME Code, Section XI and based on T-$RT_{NDT,}$ and 2) the use of the M_m calculation in ASME Code, Section XI, Paragraph G-2214.1 for a postulated defect normal to the direction of maximum stress. In its November 15, 2007, letter, the BWROG indicated in its response to NRC staff RAI 1a that the methodology described in Section 4.3 and Appendix F has been reported in P-T curve reports for Columbia Generating Station (Reference 16), Duane Arnold Energy Center (Reference 17) and LaSalle County Station, Units 1 and 2 (References 18 and 19, respectively). At the time these reports were prepared, the beltline nozzle methodology was not presented in a separate appendix; however, the methodology discussion was included in the report text. The NRC staff approved the P-T curves for Columbia Generating Station, Duane Arnold Energy Center and LaSalle County Station, Units 1 and 2 in SEs that are documented in References 20, 21 and 22, respectively.

In the November 15, 2007, letter, the BWROG indicated that the methodology described in Appendix G of the LTR has been reported in Appendix G of the P-T curve reports for Columbia Generating Station, Fermi, Unit 2 (Reference 23), and LaSalle County Station, Unit 1. The NRC staff approved the P-T curves for Fermi, Unit 2 in a SE that is documented in Reference 24.

In the November 15, 2007, letter, the BWROG indicated that the methodology described in Appendix H of the LTR has been reported in Appendix F of Dresden Nuclear Power Station, Units 2 and 3 (References 25 and 26, respectively) and Quad Cities Nuclear Power Station, Units 1 and 2 (References 27 and 28, respectively) and Appendix G of the P-T limit curve report for LaSalle County Station, Units 1 and 2. The NRC staff approved the P-T curves for Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2 in an SE that is documented in Reference 29.

Since the methodologies contained in Section 4.3 and Appendices F, G, and H of the LTR have been previously reviewed and approved by the NRC staff in the aforementioned plant-specific reviews, the BWROG's fracture mechanics analyses are acceptable for utilization in calculating P-T limit curves. Hence, this criterion has been satisfied.

<u>GL 96-03, Attachment 1 Methodology Criterion 6</u>

Methodology 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 limit curves.

The GL 96-03 conformance table in the BWROG's November 15, 2007, letter indicates this information is in Section 4.3 of the LTR. Table 4-3 in the LTR identifies the 10 CFR Part 50, Appendix G requirements for pressure and leak test, normal operation (heatup and cooldown, including anticipated operational occurrences) with the core not critical, and operation with the core critical conditions. As discussed under Criterion 5, the P-T limits for pressure and leak test and heatup and cooldown, including anticipated operational occurrences with the core not critical conditions have been previously evaluated and have satisfied the minimum temperature requirements in Appendix G to 10 CFR Part 50.

The core critical operation condition evaluation is described in Section 4.3.2.4 of the LTR. This evaluation satisfies the minimum temperature requirements of Appendix G to 10 CFR Part 50. Hence, this criterion has been satisfied.

<u>GL 96-03, Attachment 1 Methodology Criterion 7</u>

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

The GL 96-03 conformance table in the BWROG's November 15, 2007, letter indicates this information is in Section 4.2 and Appendix I of the LTR. Section 4.2 does not indicate that surveillance data is to be evaluated in accordance with Appendix I. In response to NRC staff RAI 4, the BWROG stated that Section 4.2 will be revised to indicate:

> Surveillance material information, where applicable, shall be evaluated in accordance with Section 4.2.2 and Appendix I.

This revision clarified the BWROG's evaluation of surveillance data. The NRC staff confirmed that the revised LTR dated December 2008, has incorporated the stated change.

Appendix I contains the guidance for use of BWRVIP ISP surveillance data. This guidance document indicates:

> If there is new surveillance data for any heat that 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 Adjusted Reference Temperature (ART) should be calculated for the vessel material to determine whether plant vessel integrity evaluations are affected.
>
> 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.

Procedure 1 follows the methodology documented in Position 2.1 of RG 1.99, Revision 2, and the NRC staff guidance presented by the NRC staff in an NRC/Industry workshop (Reference 30). 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 following the guidance in RG 1.99, Revision 2.

Procedure 2 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 based on the chemical composition (weight-percent copper and nickel) of the RPV material.

The NRC staff identified issues with the procedures specified in Appendix I via RAIs sent to the BWROG. The BWROG responded with proposed changes to Appendix I of the LTR. These changes, as discussed below in the evaluation of Appendix I, 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 3.2 of this SE and the fact that the procedures follow guidance recommended by the NRC staff, this criterion has been satisfied.

3.2 Evaluation of Appendix I of the LTR

Appendix I provided guidance for the use of the BWRVIP ISP surveillance data. The BWRVIP ISP replaced 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 for providing 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 for evaluating the data using the methods in RG 1.99, Revision 2, in accordance with 10 CFR Part 50, Appendix G, for the determination of ART values. Procedure 1 in the original LTR, as discussed above did not require that the surveillance data meet all the criteria in RG 1.99, Revision 2, in determining the credibility of the data. In response to NRC staff RAI 5, the BWROG's November 15, 2007, letter indicated that the following two steps will be added to Step 3 in Procedure 1 of Appendix I:

> 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 DRT_{NDT} [(ΔRT_{NDT})] values about a best-fit line drawn as described in Regulatory Guide, Revision 2, Regulatory Position 2.1, normally should be less than 28 °F for welds and 20 °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.

Procedure 1 in the original LTR also did not contain an adequate description of the criteria to be used if the vessel wall temperature is an outlier. In response to NRC staff RAIs 6 and 11, the BWROG proposed to revise Procedure 1, Step 3(b), as follows, and add a reference to available NRC staff guidance [Reference 60] to the LTR:

> 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 increase in ΔRT_{NDT} per 1 °F decrease in irradiation temperature [7]. Any temperature adjustments shall be identified and described in the PTLR.

Note that Reference 30 to this SE is equivalent to the "Reference 20" which was noted in the revised text above and which the BWROG proposed to add to Section 6.0 of the LTR.

Procedures 1 and 2 from the original LTR did not provide an adequate description of the determination of initial RT_{NDT}. In response to NRC staff RAI 7, the BWROG proposed to revise the "Definitions and Background" section of Procedures 1 and 2 as follows:

> Initial RT_{NDT} is the reference temperature for the unirradiated materials 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 154 and ARCOS B-5 weld fluxes [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.

Procedures 1 and 2 from the original LTR did not provide an adequate description of how to determine the best estimate chemistry of a material. In response to NRC staff RAI 8, the BWROG proposed to revise the note in Step 5 of Procedure 1 and Step 3 of Procedure 2 as follows:

> 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 [1]. Calculation of the best estimate chemistries for all other vessel materials should be determined in accordance with the NRC practice documented in [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.

Based on the proposed changes to Procedures 1 and 2 in Appendix I of the LTR, the NRC staff determined that Appendix I of the revised LTR, NEDC-33178, Revision 1, contains sufficient information for licensees to evaluate surveillance data and the ART for the limiting beltline material, in accordance with RG 1.99, Revision 2. The NRC staff verified that the change has been implemented in the revised LTR dated December 2008.

3.3 Evaluation of Attachment 2 of the LTR

Attachment 2 of the LTR provides a template PTLR. To ensure that the P-T limits were developed using the LTR methodology, the NRC staff, in NRC RAI 9, requested that the following information be included in the PTLR:

a) The method of determining the initial RT_{NDT} (i.e., ASME Code, Generic, Branch Technical Position - MTEB 5-2 in SRP 5.3.2 in NUREG-0800, or other NRC approved methodologies),

b) The computer codes used in the finite element analysis to determine bending and membrane stresses,

c) 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, and

d) Identify whether any of the P-T limit curves were adjusted to bound the analyses documented in Section 4.3 of the LTP or in accordance with Attachment 1, Appendix G. Identify the required adjustment in each P-T curve.

In its November 15, 2007, response to NRC staff RAI 9, the BWROG proposed that the following be added to Section 5 of the template PTLR:

> The method for determining the initial RT_{NDT} for all vessel materials is that defined in Section 4.1.2 of Reference 6.2. [Any deviations from this methodology are discussed below.] Initial RT_{NDT} values for all vessel materials considered are presented in tables in this PTLR.
>
> No new computer codes have been used in the development of the P-T curves.
>
> OR

The following computer codes, which are not described in the topical report, have been used in developing the P-T curves for [PLANT NAME].

For [PLANT NAME], the limiting material [HEAT #] considered Procedure [1] defined in Appendix I of Reference 6.2. This procedure was used because [the vessel material and the surveillance material are identical heats]. [If surveillance data was utilized, provide the surveillance data 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.]

For [PLANT NAME], there is a thickness discontinuity in the vessel [between the bottom head torus and dollar plate]. The P-T curves defined in Section 4.3 of Reference 6.2 are based upon an RT_{NDT} of [XXX] °F.

Based on the proposed changes to the template PTLR in Attachment 2, the NRC staff determined that the template PTLR contains sufficient information for the NRC staff to perform an independent evaluation of the P-T curves in accordance with RG 1.99, Revision 2 and the fracture mechanics methodology described in Section 4.3 of the LTR. The NRC staff verified that the change has been implemented in the revised LTR dated December 2008.

4.0 LIMITATIONS AND CONDITIONS

As documented in Section 3.1 of this SE, licensees who chose to implement NEDC-33178, Revision 1 as their facility's PTLR methodology must address one plant-specific action item:

The licensee must identify the report used to calculate the neutron fluence and document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology.

Information to address this licensee action item must be submitted with the licensee's requested license amendment to implement a PTLR for its facility.

5.0 CONCLUSION

The NRC staff concludes that BWROG LTR NEDC-33178P, Revision 1, satisfies the criteria in Attachment 1 in GL 96-03 and provides adequate methodology for BWR licensees to calculate P-T limit curves, given that licensees referencing this LTR comply with the conditions listed in Section 4.0 of this SE. 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 from TS to a PTLR, a licensee-controlled document.

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