BWR OWNERS' GROUP

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Project Number 691

BWROG-06049
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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Draft Safety Evaluation For The Boiling Water Reactor Owners' Group (BWROG) Structural Integrity Associates Licensing Topical Report (LTR) SIR-05-044, "Pressure Temperature Report Methodology For Boiling Water Reactors" (TAC No. MC9694)

ENCLOSURE: Requested Marked-up Copy of Safety Evaluation

Dear Sir:

Please find enclosed our comments on the subject draft Safety Evaluation for the BWROG Structural Integrity Associates Licensing Topical Report (LTR) SIR-05-044, "Pressure Temperature Report Methodology For Boiling Water Reactors" (TAC No. MC9694). As you requested, these comments are provided in a summary table (below) and a marked-up version of the report (enclosed). These comments are editorial in nature, and recommend deletion from pages 2 and 3 of the SE references to LTOP limit setpoint values, since these do not apply to BWRs.

Page	Line	Comment
2	41	Delete the phrase "and/or low temperature over-pressure protection (LTOP) limit setpoint values"
2	44	Delete the phrase "and LTOP limits"
2	46	Delete the phrase "and LTOP limits"
3	7	Delete the phrase "and LTOP system limits"
3	13	Delete the phrase "and LTOP system limits"

We look forward to a timely NRC review of our comments and the final Safety Evaluation for the LTR to support the use of this topical report by the BWROG members that use the SIA methodology

D044

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

Sincerely,

R C Bunt

Randy Bunt
BWR Owners' Group Chair

cc: Ms. Michelle Honcharik, NRC
Mr. Douglas Coleman, BWROG Vice Chair
BWROG Primary Representatives
BWROG PTC-SIA Committee

ENCLOSURE

BWROG COMMENTS ON

DRAFT 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 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 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 ~~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 limits ~~and LTOP 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 ~~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.

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 ~~and LTOP system limits~~ in the TS definition of the PTLR, and (2) revised 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 ~~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 10).

3.0 TECHNICAL EVALUATION

As stated in Section 2.1 of this SE, the NRC staff has established a rule, 10 CFR Part 50, Appendix G, that 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 BWROG methodology against the seven criteria in Attachment 1 of GL 96-03 are given 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 in the BWROG's August 29, 2006, letter indicates this LTR does not describe the transport calculation methods including computer codes and formulas used to calculate neutron fluences. It indicates fluence methods and results must comply with RG 1.190 and have NRC approval for use with this LTR. Table 1-1 will be included in the LTR. Therefore, 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 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 in the BWROG's August 29, 2006, letter indicates this information is in Appendix A of the template PTLR, which is in Appendix B of the LTR. Therefore, this will be a plant-specific action item to be addressed by licensees. Since Table 1-1 indicates 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 LTOP 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 in the BWROG's August 29, 2006, letter indicates 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 in the BWROG's August 29, 2006, letter indicates 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 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 the 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 reviewed 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 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 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 in the BWROG's August 29, 2006, letter indicates 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's August 29, 2006, letter indicates 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). 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 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

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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.

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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.

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

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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.

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