

November 14, 2006

Mr. Randy C. Bunt  
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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)

Dear Mr. Bunt:

By letter dated December 20, 2005, and supplement dated August 29, 2006, the BWROG submitted SIR-05-044, "Pressure Temperature Report Methodology for Boiling Water Reactors," Revision 0 to the U.S. Nuclear Regulatory Commission (NRC) staff for review. Enclosed for the BWROG review and comment is a copy of the NRC staff's draft safety evaluation (SE) for the LTR.

Twenty working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The NRC staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Michelle Honcharik at 301-415-1774.

Sincerely,

**/RA/**

Stacey L. Rosenberg, Chief  
Special Projects Branch  
Division of Policy and Rulemaking  
Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: Draft SE

cc w/encl: See next page

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Project No. 691

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ADAMS ACCESSION NO.: **ML062910417** \*No major changes to SE input.      NRR-106

OFFICE	PSPB/PM	PSPB/LA	Tech Branch*	PSPB/BC
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DATE	11/9/06	11/9/06	10/10/06	11/14/06

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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       1.0    INTRODUCTION

2  
3       In a letter dated December 20, 2005, the Boiling Water Reactor Owners' Group (BWROG)  
4       submitted LTR SIR-05-044, "Pressure Temperature Limits Report Methodology for Boiling  
5       Water Reactors", Revision 0, dated December 2005 (Agencywide Documents Access and  
6       Management System (ADAMS) Package Accession No. ML053560336) to the U.S. Nuclear  
7       Regulatory Commission (NRC) for review and acceptance for referencing in subsequent  
8       licensing actions. The BWROG provided this LTR to support the ability of boiling water reactor  
9       (BWR) licensees to relocate their pressure-temperature (P/T) curves and associated numerical  
10       values (such as heatup/cooldown rates) from facility Technical Specifications (TS) to a  
11       Pressure Temperature Limits Report (PTLR), a licensee-controlled document, using the  
12       guidelines provided in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature  
13       Limit Curves and Low Temperature Overpressure Protection System Limits," (Reference 1).  
14       Proposed revisions to this LTR and responses to NRC staff requests for additional information  
15       (RAIs) were provided in letter from the BWROG dated August 29, 2006 (ADAMS Accession No.  
16       ML062440387).

17  
18       2.0    REGULATORY EVALUATION

19  
20       2.1    Requirements for Generating P/T Limits for Light-Water Reactors

21  
22       The NRC has established requirements in Appendix G of Part 50 to Title 10 of the *Code of*  
23       *Federal Regulations* (10 CFR Part 50, Appendix G; Reference 2), in order to protect the  
24       integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. The  
25       regulation at 10 CFR Part 50, Appendix G requires that the P/T limits for an operating  
26       light-water nuclear reactor be at least as conservative as those that would be generated if the  
27       methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME)  
28       Boiler and Pressure Vessel Code (Reference 3, ASME Code, Section XI, Appendix G) were  
29       used to generate the P/T limits. The regulation at 10 CFR Part 50, Appendix G, also requires  
30       that applicable surveillance data from reactor pressure vessel (RPV) material surveillance  
31       programs be incorporated into the calculations of plant-specific P/T limits, and that the P/T

ENCLOSURE

1 limits for operating reactors be generated using a method that accounts for the effects of  
2 neutron irradiation on the material properties of the RPV beltline materials.  
3

4 Table 1 to 10 CFR Part 50, Appendix G provides the NRC staff's criteria for meeting the P/T  
5 limit requirements of ASME Code, Section XI, Appendix G, as well as the minimum temperature  
6 requirements of the rule for bolting up the vessel during normal and pressure testing  
7 operations. In addition, NRC staff regulatory guidance related to P/T limit curves is found in  
8 Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials,"  
9 (Reference 4), and Standard Review Plan Chapter 5.3.2, "Pressure-Temperature Limits and  
10 Pressurized Thermal Shock," (Reference 5).  
11

12 The regulation at 10 CFR Part 50, Appendix H (Reference 6), provides the NRC staff's criteria  
13 for the design and implementation of RPV material surveillance programs for operating light-  
14 water reactors.  
15

16 In March 2001, the NRC staff issued RG 1.190, "Calculational and Dosimetry Methods for  
17 Determining Pressure Vessel Neutron Fluence" (Reference 7). Fluence calculations are  
18 acceptable if they are done with approved methodologies or with methods which are shown to  
19 conform to the guidance in RG 1.190.  
20

## 21 2.2 Technical Specification Requirements for P/T Limits 22

23 Section 182a of the Atomic Energy Act of 1954 requires applicants for nuclear power plant  
24 operating licenses to include TS as part of the license. The Commission's regulatory  
25 requirements related to the content of TS are set forth in 10 CFR 50.36 (Reference 8). That  
26 regulation requires that the TS include items in five specific categories: (1) safety limits, limiting  
27 safety system settings and limiting control settings; (2) limiting conditions for operation (LCOs);  
28 (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls.  
29

30 The regulation at 10 CFR 50.36(c)(2)(ii) requires that LCOs be established for the P/T limits,  
31 because the parameters fall within the scope of the Criterion 2 identified in the rule:  
32

33 A process variable, design feature, or operating restriction that is an initial  
34 condition of a design basis accident or transient analysis that either assumes the  
35 failure of or presents a challenge to the integrity of a fission product barrier.  
36

37 The P/T limits for BWR-designed light-water reactors fall within the scope of Criterion 2 of  
38 10 CFR 50.36(c)(2)(ii) and are therefore ordinarily required to be included within the TS LCOs  
39 for a plant-specific facility operating license. On January 31, 1996, the NRC staff issued  
40 GL 96-03 to inform licensees that they may request a license amendment to relocate the P/T  
41 limit curves and/or low temperature over-pressure protection (LTOP) limit setpoint values from  
42 the TS LCOs into a PTLR or other licensee-controlled document that would be controlled  
43 through the Administrative Controls Section of the TS. In GL 96-03, the NRC staff informed  
44 licensees that in order to implement a PTLR, the P/T limits and LTOP limits for light-water  
45 reactors would need to be generated in accordance with an NRC-approved methodology and  
46 that the methodology to generate the P/T limits and LTOP limits would need to comply with the  
47 requirements of 10 CFR Part 50, Appendices G and H; be documented in an NRC-approved  
48 topical report or plant-specific submittal; and be incorporated by reference in the Administrative  
49 Controls Section of the TS. The GL also mandated that the TS Administrative Controls Section

1 would need to reference the NRC staff's safety evaluation (SE) issued on the PTLR request  
2 and that the PTLR be defined in Section 1.0 of the TS. Attachment 1 to GL 96-03 provided a  
3 list of the criteria that the approved methodology and PTLR would be required to meet.  
4

5 TS Task Force (TSTF) Traveler No. TSTF-419 (Reference 9) amended the Standard TS (STS)  
6 (NUREGs-1430, -1431, -1432, -1433, and -1434) to: (1) delete references to the TS LCO  
7 specifications for the P/T limits and LTOP system limits in the TS definition of the PTLR, and  
8 (2) revised STS 5.6.6 to identify, by number and title, NRC-approved topical reports that  
9 document PTLR methodologies, or the NRC safety evaluation for a plant-specific methodology  
10 by NRC letter and date. A requirement was added to the reviewers note to specify the  
11 complete citation of the PTLR methodology in the plant-specific PTLR, including the report  
12 number, title, revision, date, and any supplements. Only the figures, values, and parameters  
13 associated with the P/T limits and LTOP system limits are relocated to the PTLR. The  
14 methodology for their development must be reviewed and approved by the NRC. TSTF-419 did  
15 not change the requirements associated with the review and approval of the methodology or the  
16 requirement to operate within the limits specified in the PTLR. Any changes to a methodology  
17 that had not been approved by the NRC staff would continue to require NRC staff review and  
18 approval pursuant to the license amendment request provisions and requirements of  
19 10 CFR 50.90 (Reference 10).  
20

### 21 3.0 TECHNICAL EVALUATION

22

23 As stated in Section 2.1 of this SE, the NRC staff has established a rule, 10 CFR Part 50,  
24 Appendix G, that requires licensees to establish limits on the pressure and temperature of the  
25 RCPB in order to protect the RCPB against brittle failure (i.e., against brittle "fast-fracture").  
26 These limits are defined by P/T limit curves for normal operations (including heatup and  
27 cooldown operations of the reactor coolant system (RCS), normal operation of the RCS with the  
28 reactor being in a critical condition, and transient operating conditions) and during pressure  
29 testing conditions (i.e., either inservice leak rate testing and/or hydrostatic testing conditions).  
30

31 The BWROG LTR SIR-05-44 was prepared by Structural Integrity Associates and has three  
32 sections and two appendices. Section 1.0 describes the background and purpose for the LTR.  
33 Section 2.0 provides the fracture mechanics methodology and its basis for developing P/T  
34 limits. Section 3.0 provides a step-by-step procedure for calculating P/T limits. Appendix A  
35 provides guidance for evaluating surveillance data. Appendix B provides a template PTLR.  
36

#### 37 3.1 Evaluation of Section 2.0 of the LTR

38

39 Section 2.0 provides the fracture mechanics methodology and its basis for developing P/T  
40 limits. The NRC staff evaluation of this section is based on the criteria contained in  
41 Attachment 1 of GL 96-03. Attachment 1 of GL 96-03 contains seven technical criteria that the  
42 contents of proposed methodology should conform to if license amendments requesting PTLRs  
43 are to be approved by the NRC staff. The NRC staff's evaluations of the contents of BWROG  
44 methodology against the seven criteria in Attachment 1 of GL 96-03 are given in the  
45 subsections that follow.  
46

1 GL 96-03, Attachment 1, Criterion 1:  
2

3 Criterion 1 requires that the methodology describe the transport calculation methods including  
4 computer codes and formulas used to calculate neutron fluences.  
5

6 Table 1-1 in the BWROG's August 29, 2006, letter indicates this LTR does not describe the  
7 transport calculation methods including computer codes and formulas used to calculate neutron  
8 fluences. It indicates fluence methods and results must comply with RG 1.190 and have NRC  
9 approval for use with this LTR. Table 1-1 will be included in the LTR. Therefore, this will be a  
10 plant-specific action item to be addressed by licensees. Since Table 1-1 in the proposed LTR  
11 methodology will indicate that the fluence methods and results must comply with RG 1.190 and  
12 have NRC approval, this criterion has been satisfied.  
13

14 GL 96-03, Attachment 1, Criterion 2:  
15

16 Criterion 2 requires that the methodology describe the surveillance program and indicates that  
17 the topical report should contain a place holder for the requested information.  
18

19 Table 1-1 in the BWROG's August 29, 2006, letter indicates this information is in Appendix A of  
20 the template PTLR, which is in Appendix B of the LTR. Therefore, this will be a plant-specific  
21 action item to be addressed by licensees. Since Table 1-1 indicates the information will be  
22 included in the PTLR, this criterion has been satisfied.  
23

24 GL 96-03, Attachment 1, Criterion 3:  
25

26 Criterion 3 requires that the methodology describe how the LTOP system limits are calculated  
27 applying system/thermal hydraulics and fracture mechanics.  
28

29 This LTR does not need to address this criterion since Criterion 3 only applies to pressurized  
30 water reactors (PWRs) and this LTR applies to BWRs.  
31

32 GL 96-03, Attachment 1, Criterion 4:  
33

34 Criterion 4 requires that the methodology describe the method for calculating the Adjusted  
35 Reference Temperature (ART) using RG 1.99, Revision 2.  
36

37 Table 1-1 in the BWROG's August 29, 2006, letter indicates this information is in Section 2.3 of  
38 the LTR. Section 2.3 of the LTR describes the methodology documented in RG 1.99,  
39 Revision 2 for calculating ART. Therefore this criterion has been satisfied.  
40

41 GL 96-03, Attachment 1, Criterion 5:  
42

43 Criterion 5 requires that the methodology describe the application of fracture mechanics in the  
44 construction of P/T curves based on ASME Code Section XI, Appendix G, and SRP,  
45 Section 5.3.2.  
46

47 Table 1-1 in the BWROG's August 29, 2006, letter indicates this information is in Sections 2.3  
48 and 2.4 of the LTR. However, the information is in Sections 2.4 and 2.5 of the LTR (Table 1-1  
49 needs to be revised to include Section 2.5). Section 2.4 describes the general fracture

1 mechanics methodology for calculating P/T limit curves. Section 2.5 describes the  
2 methodology for calculating P/T limits for the RPV beltline, bottom head region, and non-beltline  
3 region. The non-beltline region includes all regions outside the beltline, excluding the bottom  
4 head.

5  
6 Section 2.4 provides fracture mechanics criteria based on ASME Code, Section XI,  
7 Appendix G, and ASME Code Cases N-640 and N-641. These code cases allow the use of  
8 the reference stress intensity factor,  $K_{IC}$ , for calculating P/T limit curves. NRC Regulatory Issue  
9 Summary 2004-04 (Reference 11) indicates the use of NRC-approved ASME Code cases in  
10 conjunction with earlier versions of the ASME Code endorsed in 10 CFR 50.55a may also be  
11 used for development of P/T limit curves without the need for an exemption. NRC RG 1.147,  
12 Revision 14 (Reference 12) approves these ASME Code cases. The use of the reference  
13 stress intensity factor,  $K_{IC}$ , for calculating P/T limit curves was first endorsed by the 1999  
14 Addenda of the ASME Code. Therefore, licensees utilizing this methodology and versions of  
15 ASME Code, Section XI, Appendix G that require P/T limit curves to be calculated using  $K_{IC}$  do  
16 not need to request an exemption.

17  
18 Section 2.5 describes the methodology for calculating P/T limits for the RPV beltline, bottom  
19 head region, and non-beltline region. For the beltline shell region, this section describes three  
20 methods for calculating the thermal stress intensity factor,  $K_{It}$ : a) a stress linearizing technique  
21 presented in ASME Code, Section XI, Nonmandatory Appendix A; b) a technique based on  
22 Section XI, Appendix G; and c) a technique based on Welding Research Council (WRC)  
23 Bulletin Number 175 (Reference 13). In response to NRC staff RAI No. 3, the BWROG  
24 changed the stress linearizing technique to the method in ASME Code, Section XI, Appendix G.  
25 The allowable pressure is calculated using the methodology and structural factors in ASME  
26 Code, Section XI, Appendix G. Since these techniques are based on methodologies endorsed  
27 by the NRC, they are acceptable. The NRC staff requires that this change be incorporated into  
28 the -A version of the LTR.

29  
30 The LTR indicates that the methodology for the calculating bottom head P/T limit curves should  
31 follow the methodologies for the shell region except that a stress concentration factor is applied  
32 to bottom head membrane stresses to account for the stress concentration resulting from  
33 nozzles in the lower head. In addition, the pressure stress is considered entirely as a  
34 membrane stress. Appendix 5 in WRC Bulletin Number 175 describes methods for calculating  
35 the stress intensity factors at the inside corner of a nozzle. The stress concentration factors  
36 described in these analyses are less than those utilized by the BWROG for the development of  
37 bottom head P/T limits. The methodology proposed by the BWROG for the bottom head has  
38 been previously reviewed by the NRC staff in a letter from D. S. Collins (NRC) to R. G. Byram  
39 (Senior Vice President and Chief Nuclear Officer for Susquehanna Steam Electric Station,  
40 Units 1 and 2) dated February 7, 2002 (ADAMS Accession No. ML013520605). The NRC staff  
41 performed independent calculations and concluded that the method is consistent with the  
42 methods in the 1995 Edition of Appendix G to Section XI of the ASME Code. Based on the use  
43 of a conservative concentration factor and the NRC staff's previous evaluation of this  
44 methodology, the NRC staff concludes that the methodology proposed by the BWROG for the  
45 calculating bottom head P/T limit curves is acceptable.

46  
47 The non-beltline region analysis method that was contained in Section 2.5 has been deleted  
48 and replaced with a methodology that is described in the BWROG response to RAI No. 3. In  
49 this methodology the location to be analyzed for determining the highest stresses in the



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

21  
22 Section 2.5 of the LTR and the methodology proposed in response to RAI No. 3 describe  
23 methodologies for calculating bending and membrane stresses using computer code finite  
24 element analyses. If these finite element analyses are to be utilized by licensees to develop  
25 P/T limits, the NRC staff requested, in RAI No. 2, that the BWROG provide the following:

- 26  
27 a) Identify the computer codes that were used in the finite element stress analysis. How  
28 were the codes benchmarked?  
29  
30 b) Discuss briefly the assumptions [initial  $RT_{NDT}$ ] and the inputs to the stress analysis.  
31  
32 c) Update the topical report methodology to require licensees to identify the finite element  
33 codes used in the PTLR.  
34  
35 d) Verify that this process for determining bending and membrane stresses will result in the  
36 generation of P/T limits that are at least as conservative as those which would be  
37 generated using the procedures of ASME Code, Section XI, Appendix G.  
38

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

41  
42 In the subsections that follow, finite element analysis is discussed as a possible  
43 approach for providing the necessary stress analysis for RPV regions. If finite element  
44 analysis is utilized to develop P-T limits for any RPV region, the following information  
45 shall be provided in the PTLR:

- 46  
47 a. Identify the computer code(s) that were used in the finite element stress  
48 analysis.  
49

- 1           b.     For any computer codes used, describe how the code(s) were verified or  
2           benchmarked. Computer code verification shall be in accordance with a  
3           qualified 10 CFR 50 Appendix B Quality Assurance Program. As a part of  
4           computer code verification, benchmarking consistent with NRC GL 83-11,  
5           Supplement 1 [17] shall be included.  
6
- 7           c.     Identify the assumptions and the inputs to the finite element analysis.  
8           Necessary inputs to the analysis include any or all of the following:  
9
- 10           •     A description of plant operating conditions used (e.g., pressure and  
11           temperature). The conditions used must represent current plant  
12           operating conditions.
  - 13           •     A description of the heat transfer coefficients used and the methodology  
14           used to calculate them.
  - 15           •     A description of the model developed, including materials, material  
16           properties, finite element mesh pattern, and geometry.

17

18

19

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

- 21
- 22           17.     U. S. Nuclear Regulatory Commission, Generic Letter 88-11,  
23           Supplement 1, "Licensee Qualification for Performing Safety Analyses,"  
24           June 24, 1999.

25

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

28

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

37

38   GL 96-03, Attachment 1, Criterion 6:

39

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

43

44   Table 1-1 in the BWROG's August 29, 2006, letter indicates this information is in Sections 2.7  
45   and 2.8 of the LTR. Section 2.7 identifies the minimum metal temperature of the RPV closure  
46   head flange and the RPV shell flange regions. This section also describes the minimum  
47   requirements for hydrotest (hydrostatic pressure and leak tests). Section 2.8 identifies the  
48   minimum boltup temperatures. Both of these sections identify minimum temperature  
49   requirements that are contained in Appendix G to 10 CFR Part 50. Since the information in

1 these sections comply with the requirements in Appendix G to 10 CFR Part 50, this criterion  
2 has been satisfied.

3  
4 GL96-03, Attachment 1, Criterion 7:

5  
6 Criterion 7 requires that the methodology describe how the data from multiple surveillance  
7 capsules are used in the ART calculation.

8  
9 Table 1-1 of the BWROG's August 29, 2006, letter indicates this information is in Sections 2.3  
10 of the LTR. Criteria for evaluating surveillance data are contained in Appendix A to the LTR.  
11 (Table 1-1 needs to be revised to include Appendix A). Appendix A documents two procedures  
12 for calculating the ART. One procedure is applicable when RPV material and surveillance  
13 material have identical heat numbers. This method follows the methodology documented in  
14 Position 2.1 of RG 1.99, Revision 2 and the NRC staff guidance presented in an NRC/Industry  
15 Workshop (Reference 15). Position 2.1 in RG 1.99, Revision 2 contains NRC staff guidance for  
16 evaluating surveillance data when there are two or more credible surveillance data points.  
17 Credibility is determined following the guidance in RG 1.99, Revision 2.

18  
19 The second procedure is applicable when the heat number for the surveillance material does  
20 not match the heat number for the RPV material. In this case the ART is determined using the  
21 guidance in Position 1.1 of RG 1.99, Revision 2. Position 1.1 in RG 1.99, Revision 2 contains  
22 NRC staff guidance for determining the ART from the chemical composition (weight-percent  
23 copper and nickel) of the RPV material.

24  
25 The NRC staff recommended changes to these procedures in RAIs sent to the BWROG.  
26 These changes are discussed in the evaluation of Appendix A, which is discussed in Section  
27 3.3 of this SE. The changes to Appendix A are acceptable, because they provide additional  
28 guidance to the licensees and the guidance has been previously approved by the NRC staff.  
29 Based on the changes documented in Section 3.3 and that the procedures follow guidance  
30 recommended by the NRC staff, this criterion has been satisfied.

### 31 32 3.2 Evaluation of Section 3.0 of the LTR

33  
34 Section 3.0 of the LTR provides a step-by-step procedure for calculating P/T limit curves. This  
35 section indicates that P/T limits may also be developed for other RPV regions to provide  
36 additional operating flexibility. In response to RAI No. 5, the BWROG indicated that a sentence  
37 in the LTR will be revised to state:

38  
39 P-T limit curves may also be developed for other RPV regions to provide  
40 additional operating flexibility; however, for RPV regions other than those defined  
41 in Section 2.0 of this report, licensees are required to submit methodologies to  
42 the NRC for review and approval prior to use.

43  
44 Since methods of evaluating other regions will be submitted to the NRC for review and approval  
45 prior to use, the proposed change is acceptable. The NRC staff requires that this modification  
46 be incorporated into the -A version of the LTR.

47  
48 The guidance given in Section 3.0 does not indicate surveillance data is to be evaluated. In  
49 response to RAI No. 6, the BWROG indicated a new Step (a) will be added to Section 3.0 of the

1 LTR and the previously defined steps will be re-labeled as Steps (b) through (i). The proposed  
2 new Step (a) follows:  
3

- 4 a. Evaluate surveillance data in accordance with Appendix A of this report.  
5

6 Appendix A provides guidance for the use of the Boiling Water Reactor Vessel and Internals  
7 Project (BWRVIP) Integrated Surveillance Program (ISP) surveillance data. The BWRVIP ISP  
8 replaces individual plant RPV surveillance capsule programs with representative weld and base  
9 materials data from host reactors. A representative material is a plate or weld material that is  
10 selected from among all the existing plant surveillance programs or the Supplemental  
11 Surveillance Program (SSP) to represent one or more limiting plate or weld materials in a plant.  
12 The BWRVIP ISP is responsible to provide each BWR plant with surveillance data for the  
13 materials assigned to represent that plant's limiting RPV weld and base materials. Plant  
14 owners, in turn are responsible to evaluate the data using the methods in RG 1.99, Revision 2,  
15 in accordance with 10 CFR Part 50, Appendix G, for determination of ART values.  
16

17 Since the LTR will be revised to indicate surveillance data is to be evaluated in accordance with  
18 Appendix A and Appendix A contains criteria for processing and reporting surveillance data, the  
19 proposed change is acceptable. The NRC staff requires that this change be incorporated into  
20 the -A version of the LTR.  
21

### 22 3.3 Evaluation of Appendix A of the LTR 23

24 Appendix A provides guidance for evaluating surveillance data. In response to NRC staff RAI  
25 No. 7, Appendix A will be revised to identify the source for the best estimate chemistries for the  
26 BWR vessel and surveillance capsule materials and to identify that the best estimate  
27 chemistries will be documented in the PTLR. The BWROG response adds the following note  
28 and reference to Appendix A:  
29

30 Note: Revised best estimate chemistries for selected BWR vessel and  
31 surveillance capsule materials have been calculated by the BWRVIP, as  
32 documented in BWRVIP-86-A [A-1]. Calculation of the best estimate chemistries  
33 for all other vessel materials should be determined in accordance with the NRC  
34 practice documented in Reference [A-7]. The suggested practice is documented  
35 in guidelines contained in BWRVIP-135. This evaluation is the responsibility of  
36 the plant, must be described in the PTLR, and must utilize NRC-approved  
37 methods.  
38

39 New Reference A-7 will be added to Section A.5 of the LTR as follows:  
40

- 41 A-7. "Generic Letter 92-01 and RPV Integrity Assessment - Status, Schedule,  
42 and Issues," Presentation by K. Wichman, M. Mitchell, and A. Hiser at  
43 NRC/Industry Workshop on RPV Integrity Issues, February 12, 1998.  
44

45 In response to NRC staff RAI No. 8, Appendix A will be revised to describe the temperature  
46 adjustment to the surveillance data if the temperature of the surveillance capsule is different  
47 than that of the vessel. Appendix A, Procedure 1, Procedural Step 3(b) of the LTR will be  
48 revised as follows:  
49

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

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

10                   Initial  $RT_{NDT}$  is the reference temperature for the unirradiated material as defined  
11                   in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel  
12                   Code. Some plants have measured values of initial  $RT_{NDT}$ ; other plants use  
13                   generic values. For generic values of weld metal, the following generic mean  
14                   values must be used: 0°F for welds made with Linde 80 flux, and -56°F for welds  
15                   made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes [A-6]. Other  
16                   generic mean values may be used, provided they are justified and have NRC  
17                   review and approval. The generic mean values used shall be identified in the  
18                   PTLR.  
19

20  
21     Reference A-6 is the Pressurized Thermal Shock rule, 10 CFR 50.61. The rule provides  
22     generic initial  $RT_{NDT}$  values for welds made with Linde 80, 0091, 1092, and 124 and ARCOS B-  
23     5 weld fluxes. These values have been reviewed and approved by the NRC staff. Therefore,  
24     they are also applicable for BWR RPVs.  
25

26     In response to NRC staff RAI No. 10, Appendix A will be revised to identify information that the  
27     licensee should review to determine whether the data is “credible” or “non-credible” in  
28     accordance with RG 1.99, Revision 2. The following two steps will be added to Appendix A,  
29     Procedure 1, Procedural Step 3 of the LTR:  
30

- 31           d.     Scatter in the plots of Charpy energy versus temperature for the irradiated and  
32                   unirradiated conditions should be small enough to permit the determination of  
33                   the 30 foot-pound temperature and the upper shelf energy unambiguously.  
34  
35           e.     When there are two or more sets of surveillance data from one reactor, the  
36                   scatter of  $\Delta RT_{NDT}$  values about a best-fit line drawn as described in Reg.  
37                   Guide 1.99 Rev. 2, Regulatory Position 2.1, normally should be less than 28°F  
38                   for welds and 17°F for base metal. Even if the fluence range is large (two or  
39                   more orders of magnitude), the scatter should not exceed twice those values.  
40                   Even if the data fail this criterion for use in shift calculations, they may be  
41                   credible for determining decrease in upper-shelf energy if the upper shelf can be  
42                   clearly determined, following the definition given in ASTM E185-82.  
43

44     The changes to Appendix A are acceptable, because they provide additional guidance to the  
45     licensees and the guidance has been previously approved by the NRC staff. The NRC staff  
46     requires that these changes to Appendix A of the LTR be incorporated into the -A version of the  
47     report.  
48

1 3.4 Evaluation of Appendix B of the LTR  
2

3 Appendix B provides a template PTLR. To ensure that the P/T limits were developed using the  
4 LTR methodology, the NRC staff in RAI No. 11 requested that the following information be  
5 included in the PTLR:  
6

- 7 a) The initial  $RT_{NDT}$  [ $IRT_{NDT}$ ] for all reactor pressure vessel materials and the method of  
8 determining the initial  $RT_{NDT}$  (i.e., ASME Code, Generic Communication, Branch  
9 Technical Position - MTEB 5-2 in SRP 5.3.2 in NUREG-0800, or other NRC-approved  
10 methodologies),  
11  
12 b) The chemistry (weight-percent copper and nickel) and ART at the 1/4 thickness location  
13 for all beltline materials, and  
14  
15 c) The computer codes used in the finite element analysis to determine for calculating  
16 bending and membrane stresses from Section 2.5 of the methodology.  
17  
18 d) Identify whether "Procedure #1" or "Procedure #2" was utilized to evaluate the  
19 surveillance data. If surveillance data was utilized, provide the surveillance data and the  
20 analysis of the surveillance data that was used to determine the ART. If surveillance  
21 data was not utilized, state why it was not utilized.  
22

23 In response to NRC staff RAI No. 11 items (a), (b), and (d), the BWROG proposed that the  
24 following be added to Section 2.3 of the LTR:  
25

26 The following information should be included in the PTLR with respect to the ART  
27 calculations:  
28

- 29 a. The  $IRT_{NDT}$  for all RPV materials and the method of determining the  $IRT_{NDT}$   
30 (i.e., ASME Code, Generic Communication, Branch Technical Position MTEB 5-2  
31 in Standard Review Plan 5.3.2 in NUREG-0800, or other NRC-approved  
32 methodologies).  
33  
34 b. The chemistry (weight-percent copper and nickel) and ART at the 1/4t location  
35 for all beltline materials.  
36  
37 c. Identify whether "Procedure 1" or "Procedure 2" from Appendix A was utilized to  
38 evaluate the surveillance data. If surveillance data was utilized, provide the  
39 surveillance [data] and the analysis of the surveillance data that was used to  
40 determine the ART values. If surveillance data was not utilized, state why it was  
41 not utilized.  
42

43 The changes are acceptable, because they provide additional guidance for licensees and  
44 provide information that the NRC staff needs to evaluate the PTLR. The NRC staff requires  
45 that these changes be incorporated into the -A version of the report.  
46

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

1       4.0     CONCLUSION  
2

3       The NRC staff concludes that BWROG LTR SIR-05-044 satisfies the criteria in Attachment 1 to  
4       GL 96-03 and provides adequate methodology for BWR licensees to calculate P/T limit curves.  
5       By using this methodology and following the PTLR guidance in GL 96-03, as amended by NRC  
6       TSTF-419, BWR licensees will be able to relocate the P/T limit curves and the associated  
7       heatup/cool-down rates from TS to a PTLR, a licensee-controlled document.  
8

9       The NRC staff has recommended, as noted in this SE, additional changes to Table 1-1 of the  
10       LTR. The BWROG must incorporate the NRC staff recommended changes and the changes  
11       proposed by the BWROG in their letter dated August 29, 2006, into the -A version of the report.  
12

13       5.0     REFERENCES  
14

- 15       1.       NRC Generic Letter 96-03, "Relocation of the Pressure-Temperature Limit Curves and  
16       Low Temperature Overpressure Protection System Limits," January 31, 1996 (ADAMS  
17       Legacy Library Accession No. 9601290350).
  - 18       2.       10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," 2005 Edition.
  - 19       3.       ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness  
20       Criteria for Protection Against Failure," 2004 Edition.
  - 21       4.       NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel  
22       Materials," May 1988 (ADAMS Accession No. ML003740284).
  - 23       5.       NUREG-0800, NRC Standard Review Plan, Section 5.3.2, "Pressure-Temperature  
24       Limits and Pressurized Thermal Shock," Draft Revision 2, June 1996.
  - 25       6.       10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program  
26       Requirements," 2005 Edition.
  - 27       7.       NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining  
28       Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession  
29       No. ML010890301).
  - 30       8.       10 CFR 50.36, "Technical specifications," 2005 Edition.
  - 31       9.       NRC Technical Specification Traveler Form TSTF-419, Revision 2, "Pressure  
32       Temperature Limits Report [PTLR]," September 16, 2001 (ADAMS Accession  
33       No. ML012690234).
  - 34       10.       10 CFR 50.90, "Application for amendment of license or construction permit,"  
35       2005 Edition.
  - 36       11.       NRC Regulatory Issue Summary 2004-04, "Use of Code Cases N-588, N-640 and  
37       N-641 in Developing Pressure-Temperature Operating Limits," April 5, 2004 (ADAMS  
38       Accession No. ML040920323).
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- 1 12. NRC Regulatory Guide 1.147, Revision 14, "Inservice Inspection Code Case  
2 Acceptability, ASME Section XI, Division 1," August 2005 (ADAMS Accession  
3 No. ML052510117).  
4
- 5 13. WRC Bulletin No. 175, "Pressure Vessel Research Committee (PVRC)  
6 Recommendations on Fracture Toughness Requirements for Ferritic Materials,"  
7 August 1972.  
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- 9 14. GE Topical Report NEDE-21821-02, "BWR Feedwater Nozzle/Sparger Final Report,  
10 Supplement 2," August 1979.  
11
- 12 15. NRC, Generic Letter 92-01 and RPV Integrity Workshop Handouts, K. Wichman,  
13 M. Mitchell, and A. Hiser, NRC/Industry Workshop on RPV Integrity Issues,  
14 February 12, 1998.  
15

16 Principle Contributor: B. Elliot

17  
18 Date: November 14, 2006