March 24, 2008

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TOPICAL REPORT (TR) BAW-2308, REVISION 2, "INITIAL RT_{NDT} OF LINDE 80 WELD MATERIALS" (TAC NO. MD4241)

Dear Mr. Bischoff:

By letter dated February 5, 2007, the PWROG submitted TR BAW-2308, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. TR BAW-2308, Revision 1-A, was published on November 17, 2005, with the NRC safety evaluation (SE) containing, among others, two NRC imposed conditions to reevaluate the conclusions of Revision 1-A, considering anticipated additional test data and to reevaluate the conclusions of Revision 1-A, using the most recent consensus approach in the American Society for Testing and Materials for reference temperature determination. The intent of TR BAW-2308, Revision 2, was to supplement TR BAW-2308, Revision 1-A, by addressing these two NRC imposed conditions.

By letter dated February 20, 2008, an NRC draft SE regarding our approval of TR BAW-2308, Revision 2, was provided for your review and comments. By letter dated March 11, 2008, the PWROG commented on the draft SE. The NRC staff's disposition of PWROG's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR BAW-2308, Revision 2, is acceptable for referencing in licensing applications for: Arkansas Nuclear One, Unit 1; Crystal River Nuclear Plant, Unit 3; Davis-Besse Nuclear Power Station; Oconee Nuclear Station, Units 1, 2, and 3; Three Mile Island Nuclear Station, Unit 1; Point Beach Nuclear Plant, Units 1 and 2; Surry Power Station, Units 1 and 2; and Turkey Point Nuclear Plant, Units 3 and 4 to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

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

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

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

Sincerely,

/RA/

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

Project No. 694

Enclosure: Final SE

cc w/encl: Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
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Pittsburgh, PA 15230-0355
greshaja@westinghouse.com

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ADAMS ACCESSION NO. ML080770349 ***No major changes to SE input. NRR-043**

OFFICE	PSPB/PM	PSPB/PM	PSPB/LA	CVIB/BC*	PSPB/BC	DPR/DD
NAME	HCruz	SPeters	DBaxley	MMitchell	SRosenberg	HNieh
DATE	3/17/08	2/19/08	3/17/08	11/8/07	3/ 21 /08	3/ 24 /08

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT BAW-2308, REVISION 2

"INITIAL RT_{NDT} OF LINDE 80 WELD MATERIALS"

PRESSURIZED WATER REACTOR OWNERS GROUP

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

By letter dated July 26, 2002 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML022200546), Babcock and Wilcox Owners Group (B&WOG), now part of the Pressurized Water Reactor Owners Group (PWROG), submitted Topical Report (TR) BAW-2308, Revision 0, "Initial RT_{NDT} of Linde 80 Weld Materials," to the U.S. Nuclear Regulatory Commission (NRC) staff for review. The intent of the TR was to establish an alternative method for determining initial, unirradiated material reference temperatures for reactor pressure vessel (RPV) welds manufactured using Linde 80 weld flux (Linde 80 welds) and to establish weld wire heat-specific and Linde 80 weld generic values of this reference temperature, which would be used in lieu of the nil-ductility reference temperature (RT_{NDT}) parameter specified in Paragraph NB-2331 of Section III of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

By letter dated August 19, 2003 (ADAMS Accession No. ML032380449), the B&WOG withdrew TR BAW-2308, Revision 0, and submitted for review TR BAW-2308, Revision 1, which incorporated the substantive changes resulting from the NRC staff's request for information (RAI) on TR BAW-2308, Revision 0. The safety evaluation (SE) for BAW-2308, Revision 1, was issued on August 4, 2005, and the approved TR version, BAW-2308, Revision 1-A, was published on November 17, 2005. This SE requested, among other NRC imposed conditions and limitations, that the PWROG: (1) reevaluate the conclusions of Revision 1-A, considering additional test data, and (2) reevaluate the conclusions of Revision 1-A, using the most recent consensus approach in the American Society for Testing and Materials (ASTM) for reference temperature, T_0, determination. By letter dated February 5, 2007, the PWROG submitted TR BAW-2308, Revision 2, as a supplement to Revision 1-A, to address these two NRC imposed conditions. The NRC staff review of TR BAW-2308, Revision 2, includes the PWROG response to the NRC staff's RAI for this submittal dated September 12, 2007.

2.0 REGULATORY EVALUATION

The determination of RPV material properties impacts regulations associated with the protection of the RPV from brittle failure and ductile rupture. These regulations include Appendix G to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) and 10 CFR 50.61, the pressurized thermal shock (PTS) rule. Appendix G to 10 CFR Part 50 and 10 CFR 50.61 require

ENCLOSURE

that the initial, unirradiated material reference temperature, RT_{NDT}, be determined in accordance with the provisions of ASME Code, Section III, Paragraph NB-2331. The determination of RT_{NDT} per ASME Code, Section III, Paragraph NB-2331, requires the performance of drop weight testing in accordance with ASTM Standard Test Method E 208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," and Charpy V-notch impact testing in accordance with ASTM Standard Test Method E 23, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials." Guidance provided in NRC Standard Review Plan Section 5.3.1, “Reactor Vessel Material,” and Branch Technical Position MTEB 5-2, “Fracture Toughness Requirements,” also reflect this dependence on drop weight and Charpy V-notch impact testing. In addition, regarding the implementation of alternatives to the requirements of Appendix G to 10 CFR Part 50, 10 CFR 50.60 states, “Proposed alternatives to the described requirements in Appendices G and H of this part or portions thereof may be used when an exemption is granted by the Commission....”

3.0 TECHNICAL EVALUATION

3.1 PWROG Evaluation

The PWROG submitted TR BAW-2308, Revision 2, as a supplement to TR BAW-2308, Revision 1-A, to address the two NRC conditions stated in the SE for TR BAW-2308, Revision 1-A: (1) reevaluation of the conclusions of Revision 1-A, considering additional test data, and (2) reevaluation of the conclusions of Revision 1-A, using the most recent consensus approach in the ASTM for the T_0 determination. The fracture toughness testing in TR BAW-2308, Revision 1-A, was based on the application of the “Master Curve” evaluation procedure. Using this procedure, the T_0 was derived from data obtained from sample sets tested at different temperatures. The initial, unirradiated nil-ductility reference temperature (IRT_{T0}) was then derived for each Linde 80 weld heat based on its T_0 value. Separately, the associated initial margin (σ_I) was calculated using an approach similar to that in TR BAW-2308, Revision 1-A.

TR BAW-2308, Revision 2, documented in Table 1 (unirradiated data) and Table 2 (irradiated data), new test data for the specimens of weld metal SA-1135, which were manufactured using weld wire heat 61782 (referred to later as weld heat 61782). These two tables listed test temperatures, the J_C values (the J-integral at the onset of cleavage fracture), and the K_{JC} values (the elastic-plastic equivalent stress intensity factor derived from J_C) for new weld heat 61782 test specimens. TR BAW-2308, Revision 2, also considered additional test data for weld heat 299L44 from BAW-2439, “Analysis of the B&W Owners Group Capsule TMI2-LG2.”

To satisfy the first condition specified in the SE for TR BAW-2308, Revision 1-A, this TR repeated the TR BAW-2308, Revision 1-A analyses and recalculated the weld wire heat-specific IRT_{T0} and σ_I values and the Linde 80 weld generic values considering new weld heat 61782 and weld heat 299L44 test data. To satisfy the second SE condition, this TR used the proposed 2007 Edition of the ASTM Standard Test Method E 1921 (ASTM E 1921), “Standard Test Method for Determination of Reference Temperature, T_0, for Ferritic Steels in the Transition Range,” in all relevant IRT_{T0} and σ_I calculations. The recalculated IRT_{T0} and σ_I values, along with those reported in TR BAW-2308, Revision 1-A, were summarized in Table 9 of TR BAW-2308, Revision 2, for various heats of Linde 80 welds. As indicated in the TR, the proposed 2007 Edition of ASTM E 1921 adopted a loading rate adjustment for T_0, which is

different from that of TR BAW-2308, Revision 1-A. Based on the Table 9 summary results, the PWROG concluded that when combining the σ_I values and the shift margin (σ_Δ) of 28 °F with IRT_{T0}, the approved IRT_{T0} and σ_I values in TR BAW-2308, Revision 1-A are conservative relative to the recalculated values presented in this TR, with the exception of the results for weld heat 72105 and weld heat 299L44. The results for weld heat 72105 are non-conservative relative to the approved value by 3.2 °F, and the results for weld heat 299L44 are non-conservative relative to the approved value by 8.5 °F. The σ_Δ of 28 °F is specified in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," for welds.

3.2 NRC Staff Evaluation

As a supplement to TR BAW-2308, Revision 1-A, TR BAW-2308, Revision 2, addressed two of the NRC conditions stated in the SE for TR BAW-2308, Revision 1-A, to reevaluate the conclusions of Revision 1-A, considering additional test data for weld heat 61782, and to reevaluate the conclusions of Revision 1-A, using the most recent consensus approach in the ASTM for the T_0 determination.

Regarding the first condition, the PWROG followed a similar procedure as described in TR BAW-2308, Revision 1-A, to calculate the IRT_{T0} and σ_I values for the new weld heat 61782, the expanded weld heat 299L44 (considering 8 additional data from TMI2-LG2), and the generic Linde 80 weld (considering old and new data for all heats) data. Since all new data was considered, the first condition was adequately addressed by the PWROG. Regarding the second condition, the PWROG used the most recent proposed ASTM E 1921 (June 2007 revision) procedure to calculate the T_0 values. This proposed procedure considered a different approach to assess the effect of loading rate on specimen test results. The 2007 Edition of ASTM E 1921 has been approved by ASTM, but has not yet been published in the ASTM Standards Book. The NRC representative to the ASTM subcommittee responsible for ASTM E 1921 confirmed that the NRC had no objection to the 2007 Edition of ASTM E 1921. Therefore, the second condition was also adequately addressed by the PWROG.

Except for the loading rate effect adjustment applied to the T_0 determination and a standard deviation (related to Monte Carlo analyses) averaging applied to the σ_I determination for the new weld heat 61782, the TRs methodology in determining IRT_{T0} and σ_I values is identical to that of TR BAW-2308, Revision 1-A. Hence, the NRC staff's RAIs are limited to the loading rate effect calculations and the implementation of this slightly modified methodology. The PWROG's September 12, 2007, response to the RAIs provided detailed calculations for the proposed and the existing loading rate effect adjustments for weld heat 71249, and weld heat 299L44, validating the T_0 value in Table 5 of TR BAW-2308, Revision 2, for two typical heats of Linde 80 welds. For the revised σ_I values, the same Monte Carlo analyses were performed for the expanded weld heat 299L44, and the generic Linde 80 weld evaluation, considering all new data. However, since there is insufficient data to properly perform a Monte Carlo analysis for the new weld heat 61782, the PWROG derived the standard deviation for weld heat 61782, based on average standard deviation values from all other heats of Linde 80 welds. The NRC staff determined that averaging was acceptable because insufficient data would make the Monte Carlo results meaningless. Other RAI responses provided satisfactory explanation regarding the sample size uncertainty calculation, and certain specimen data number discrepancies. Therefore, all RAIs are resolved, and the modified methodology for IRT_{T0} and σ_I calculations is acceptable to the NRC staff.

The PWROG applied the modified methodology to all Linde 80 weld heats and found that, after combining the σ_I values and the shift margin (σ_Δ) of 28 °F with IRT_{T0}, the approved IRT_{T0} and σ_I values in TR BAW-2308, Revision 1-A, are not conservative relative to the revised values presented in this TR by 3.2 °F for weld heat 72105, and by 8.5 °F for weld heat 299L44. As a result, the NRC staff determined that for future plant-specific applications for RPVs containing these two heats of Linde 80 welds, the revised IRT_{T0} and σ_I values in TR BAW-2308, Revision 2, must be used.

4.0 LIMITATIONS AND CONDITIONS

As a result of the evaluation in Section 3.2 of this SE, the NRC staff has concluded that the TR BAW-2308, Revision 2, is acceptable for referencing in licensing applications for PWRs with the following condition:

> Future plant-specific applications for RPVs containing weld heat 72105, and weld heat 299L44, of Linde 80 welds must use the revised IRT_{T0} and σ_I values in TR BAW-2308, Revision 2.

5.0 CONCLUSION

Based on the evaluation in Section 3.2 of this SE, the NRC staff concludes that the slightly modified PWROG initial RT_{NDT} methodology and the revised IRT_{T0} and σ_I values in TR BAW-2308, Revision 2, are acceptable for estimating the IRT_{T0} and σ_I values for various heats of the Linde 80 welds in future RPV integrity evaluation applications. In addition, TR BAW-2308, Revision 2, is acceptable for referencing in licensing applications for Arkansas Nuclear One, Unit 1; Crystal River Nuclear Plant, Unit 3; Davis-Besse Nuclear Power Station; Oconee Nuclear Station, Units 1, 2, and 3; Three Mile Island Nuclear Station, Unit 1; Point Beach Nuclear Plant, Units 1 and 2; Surry Power Station, Units 1 and 2; and Turkey Point Nuclear Plant, Units 3 and 4 as delineated in the TR, and to the extent specified under Section 4.0, Limitations and Conditions, of this SE. Due to the limited objective of this TR, this TR is considered a supplement to, not a replacement of, TR BAW-2308, Revision 1-A. Therefore, Items (1) to (4) listed under Section 5, “Conditions and Limitations,” of the NRC staff’s August 4, 2005, SE for TR BAW-2308, Revision 1-A, must still be addressed in all future plant-specific applications referencing TR BAW-2308, Revision 1-A, and this TR. This TR eliminates the need for future applicants to address Items (5) and (6) listed under Section 5 of the NRC staff’s August 4, 2005, SE.

Attachment: Resolution of Comments

Principle Contributor: Simon Sheng

Date: March 24, 2008

RESOLUTION OF PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

COMMENTS ON DRAFT SAFETY EVALUATION (SE) FOR TOPICAL REPORT (TR)

BAW-2308, REVISION 2, "INITIAL RT_{NDT} OF LINDE 80 WELD MATERIALS"

PROJECT NO. 694

By letter dated March 11, 2008, the PWROG provided one comment on the draft SE for TR BAW-2308, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials." The following is the NRC staff's resolution of this comment:

Page 1, Section 2.0, Line 32:

PWROG Comment:

> Remove "or" when referring to RPV [reactor pressure vessel] protection from brittle failure and ductile rupture and replace with "and."

NRC Response:

The NRC staff agrees with this change.

ATTACHMENT