Program Management Office
20 International Drive
Windsor, Connecticut 06095

Project No. 694

BAW-2308-NP, Rev 2

April 30, 2008

OG-08-155

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

Subject: PWR Owners Group
Transmittal of NRC Approved Topical Report Report BAW-2308-NP, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC NO. MD4241), PA-MSC-0229

Reference: 1. Letter, H. Nieh (NRC) to G. Bischoff (PWROG), "Final Safety Evaluation for Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) BAW-2308-NP, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC NO. MD4241), dated March 24, 2008.

The purpose of this letter is to transmit four (4) non-proprietary copies of BAW-2308-NP, Revision 2, for NRC files. BAW-2308-NP, Revision 2, contains the staff's Safety Evaluation. This transmittal completes action on topical report BAW-2308-NP, Revision 2; thus, the PWROG requests that TAC No. MD4241 be closed.

For technical questions regarding the enclosed report, please contact the program technical lead Brian Hall (AREVA NP) at (434) 832-2537 or Alan Thomas (AREVA NP) at (434) 832-2989. If you have any additional questions or comments on the enclosed report, feel free to contact Jim Molkenthin in the PWROG office at (860) 731-6727.

Sincerely,

Dennis E. Buschbaum, Chairman
PWR Owners Group

DEB:JPM:las

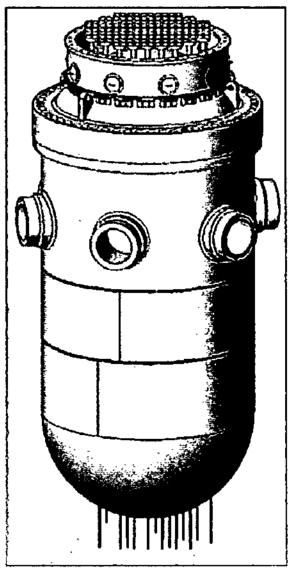
D048
NRR

Enclosure: (1) – PWROG Report BAW-2308-NP, Revision 2

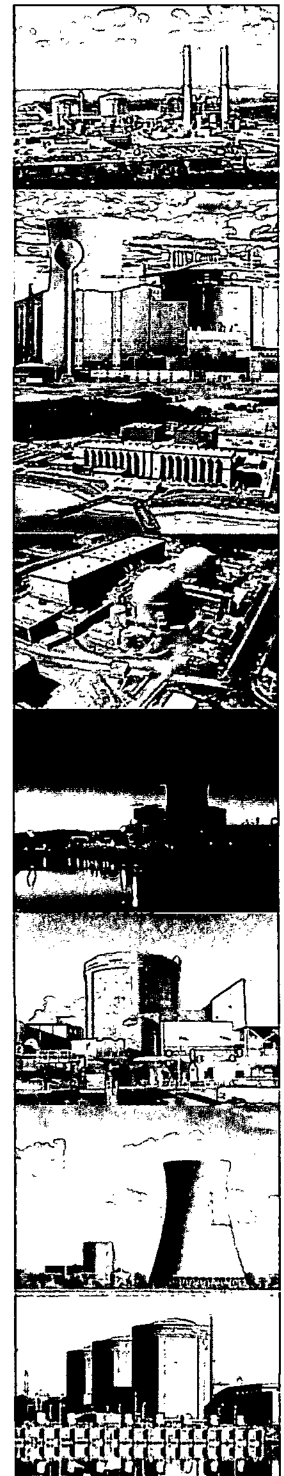
cc: PWROG Management Committee Participants in PA-MSC-0229
PWROG Materials Subcommittee Participants in PA-MSC-0229
PWROG PMO
E. Giavedoni, AREVA NP
A. Thomas, AREVA NP
B. Hall, AREVA NP
R. Schomaker, AREVA NP
C. Brinkman, Westinghouse
H. Cruz, USNRC (3 copies)
S. Peters, USNRC
S. Rosenberg, USNRC

BAW-2308 Revision 2-A
March 2008

INITIAL RT_{NDT} OF LINDE 80 WELD MATERIALS



PWROG
PWR Owners Group



Dominion Energy
Duke Energy Corporation
Entergy Operations, Inc.
Exelon Nuclear Corporation
FirstEnergy Nuclear Operating Company
Florida Power & Light Company
Nuclear Management Company
Progress Energy

AREVA NP Inc.

AREVA

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 24, 2008

RECEIVED
MAR 27 2008
PWROG
PROJECT OFFICE

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,

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
P.O. Box 355
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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 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

Program Management Office
20 International Drive
Windsor, Connecticut 06095

September 12, 2007

OG-07-407

BAW-2308-NP, Rev. 2
Project Number 694

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

Subject: Pressurized Water Reactor Owners Group
Responses to the NRC Request for Additional Information (RAI) on PWR Owners Group (PWROG) Report BAW-2308-NP, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC NO. MD4241) PA-MSC-0229

References:
1. Submittal of BAW-2308-NP, Revision 2 "Initial RTNDT of Linde 80 Weld Materials", PA-MSC-0229 (PWOG Letter OG-07-47), dated February 5, 2007.
2. Acceptance for Review of Pressurized Water Reactor Owners Group (PWROG) Topical Report BAW-2308-NP, Revision 2 "Initial RT_{NDT} of Linde 80 Weld Materials" (TAC NO. MD4241) PA-MSC-0229, (PWROG Letter OG-07-175), dated April 17, 2007.
3. NRC Letter from H. Cruz to G. Bischoff, Request For Additional Information Re: Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR) BAW-2308, Revision 2, "Initial RTNDT of Linde 80 Weld Materials" (TAC NO. MD4241), dated July 20, 2007.

In February 2007, the Pressurized Water Reactor Owners Group (PWROG), submitted Topical Report BAW-2308-NP, Revision 2 "Initial RTNDT of Linde 80 Weld Materials", for review and approval (Reference 1). On April 12, 2007, the NRC accepted the topical report (Reference 2) and provided a formal Request for Additional Information (RAIs) (Reference 3) on July 20, 2007.

Enclosure 1 to this letter provides RAI responses to the 3 questions received in Reference 3.

If you have any questions, please do not hesitate to contact me at (630) 657-3897, or if you require further information, please contact Mr. Jim Molkenthin of the PWR Owners Group Project Management Office at (860) 731-6727.

Regards,

J. Molkenthin approving for T. Schiffley

Frederick P. "Ted" Schiffley, II, Chairman
PWR Owners Group

FPS:JPM:las

Enclosures (1) – RAI Responses to BAW-2308-NP, Revision 2

cc: M. Mitchell, USNRC
S. Peters, USNRC
S. Rosenberg, USNRC
C. Brinkman, W
B. Hall, AREVA NP
R. Schomaker, AREVA NP
PWROG MSC Participants in PA-MSC-0229
PWROG Management Participants in PA-MSC-0229
PWROG PMO
D. Napior, AREVA NP
B. Gray, AREVA NP

REQUEST FOR ADDITIONAL INFORMATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
TOPICAL REPORT (TR) BAW-2308-NP, REVISION 2
"INITIAL RTNDT OF LINDE 80 WELD MATERIALS"
PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)
PROJECT NO. 694

1. Loading rate effect on the reference temperature, T_0, is discussed in Section 3 of TR BAW-2308, Revision 2. It was stated on Page 10, "Using the above equations [the proposed ASTM E1921 equations] to adjust the loading rates of the five data sets that were tested faster than 2 Mpa√m/s to the limit of 2 Mpa√m/s, results in a reduction of 0.9°F in T_0. This model predicts a loading rate effect of 13.3°F on T_0 from the slowest to the fastest loading rate extremes shown in Table 5, while the AREVA model predicts an effect of 22.9°F."

- Provide information regarding the calculation of the reduction of 0.9°F in T_0 as a result of adjusting the loading rate to 2 Mpa√m/s for the five data sets mentioned in the quote. Explain the use of this reference loading rate of 2 Mpa√m/s here for the five data sets, while both the BAW-2308 and the proposed E1921 use a reference loading rate of 1 Mpa√m/s to derive their adjusted T_0 values. Further, "loading rate extremes shown in Table 5," which is part of the above quote, should be revised to "loading rate extremes shown in Table 4."

<u>Response</u>:

The statement regarding the effect on T_0 due to the loading rate being above 2 Mpa√m/s for the five data sets was simply to demonstrate that the effect on T_0 being above the loading rate allowed in E1921-05 is minor. Adjustment to 2 Mpa√m/s was not used in the subsequent calculations. All data was adjusted to the reference loading rate of 1 Mpa√m/s. Reference to Table 5 will be corrected to Table 4 on page 10 in the approved version of this topical report.

- Provide information regarding the calculation of the loading rate effect of 13.3°F on T_0 using the proposed ASTM E1921 equations and the effect of 22.9°F on T_0 using the BAW-2308 model. Table 4-3 of BAW-2308, Revision 1, does not support the stated effect of 22.9°F on T_0. Please clarify that this calculation is based on T_0 values for all data sets shown in Table 4, not just the five data sets.

<u>Response</u>:

The comparison of the AREVA model, which shows a loading rate effect of 22.9F between 0.22 Mpa√m/s and 2.35 Mpa√m/s, and the proposed ASTM E1921 model (13.3F) was to simply show the difference between the two models. For the AREVA model: T_0 adjustment = 5.33 ln (R2/R1) = 5.33 ln (2.35/0.22) = 12.6C (or 23F). Selecting the slowest test data set (71249; SA-1094 at 0.22 Mpa√m/s) and the fastest (299L44; SA-1526 at 2.35 Mpa√m/s) from Table 4 and finding these in Table 4-3 of BAW-2308, Revision 1, it can be seen that the PCS adj. T_0 for SA-1094 is -97F and the PCS+rate adjusted is -83F, a rate effect of 14F. For SA-1526, the PCS adj. T_0 is -96F and the PCS+rate adjusted is -105F, a rate effect of -9F. These were both adjusted to a loading rate of 1 Mpa√m/s using the AREVA model, so the difference between these values is 14F – (-9F) = 23F, the same as that reported on page 10 of BAW-2308, Revision 2.

The loading rate effect of the proposed ASTM E1921 model is less (13.3F) between 0.22 Mpa√m/s and 2.35 Mpa√m/s. All the data in BAW-2308, Revision 2 was adjusted to 1 Mpa√m/s using the proposed ASTM E1921 model:

$$T_{o,X}^{est} = \frac{(T_o + 273.15)\cdot\Gamma}{\Gamma - \ln(X)} - 273.15$$

Or, rearranging to solve for T_0 (at 1 MPa√m/s)

$$T_0 = \frac{(T_{0,X}^{est} + 273.15)\cdot(\Gamma - \ln(X))}{\Gamma} - 273.15$$

with X in MPa□m/s and temperatures in °C. The function □ is given by:

$$\Gamma = 9.9\cdot\exp\left[\left[\frac{(T_o + 273.15)}{190}\right]^{1.66} + \left(\frac{\sigma_{ys,To}}{722}\right)^{1.09}\right]$$

and where:

T_0 refers to the quasi-static loading rate of dK/dt = X = $\dot{K}_I$ = 1 MPa√m/s, and

□$_{ys,To}$ = quasi-static yield strength measured or estimated at T_o in MPa.

For the slowest and fastest data sets:

Weld	$T_{0,X}$	X = $\dot{K}_I$ (MPa√m/s)	□$_{ys,To}$ (MPa)	□	T_0	ΔT_0 (F)
SA-1094	-97.2F (-72C)	0.22	559	64	-88.5F (-67C)	8.7F
SA-1526	-96.4F (-71C)	2.35	567	65	-101.2 (-74C)	-4.8F

The total change in T_0 from the loading rate extremes using in BAW-2308, Revision 2 in adjusting the slowest test and the fastest data sets to 1 MPa√m/s results in 8.7F - (-4.8F) = 13.5F.

2. There are inconsistent information on Pages 5, 6, 7, and 13 regarding the number of additional data that was used in the proposed analysis.

- It was stated on Page 5 that, "Two 0.394 TC(T), four 0.500 TC(T), and two 0.936 TDC(T) specimens also from SA-1135 with an average fluence of 1.368 x 10^{19} n/cm^2 (E > 1 MeV) were also tested per ASTM E1921-02." Table 2 (Page 6) and Table 3 (Page 7) show, however, that there are three irradiated 0.936 DC(T) specimens from SA-1135. Please clarify the number of the 0.936 DC(T) specimens.

Response:

There is an error in the text on page 5 as indicated. Tables 2 and 3 are correct. This error will be corrected in the approved version of this topical report. The "two 0.936TDC(T) specimens" will be changed to "three 0.936TDC(T) specimens."

- Table 7 (Page 13) uses the following wording to describe the revised sample size of the Linde 80 weld for Heat 299L44: "299L44 with 8 added tests from TMI2-LG2." This suggests that results from eight additional specimens are included in the proposed sample size uncertainty analysis. However, the Table 7 data show that the uncensored specimen number for 299L44 for the corresponding BAW-2308, Revision 1 analysis was 22 and the uncensored specimen number for 299L44 for the proposed analysis is 29, suggesting that seven additional uncensored specimens are included in the proposed analysis. Please confirm that one of the 8 added specimens is censored, and it is not used in any data analyses described in BAW-2308, Revision 2.

Response:

One of the eight added tests for heat 299L44 taken from RVSP capsule report TMI2-LG2 (BAW-2439 Table 5-2) is censored data. The censored test was on a precracked Charpy size specimen tested at -100F with a resulting J_c of 536 lb/in (K_{Jc} = 126.7 ksi$\sqrt{\text{in}}$). It is treated as a censored data point in the multi-temperature T_0 calculation method as described in ASTM E1921-05 section 10.2.2. It is not used in the sample size uncertainty calculation as indicated in Table 7, where it lists the number of uncensored data used. The Monte Carlo analysis treats this data point as censored per the ASTM E1921-05 section 10.2.2 also.

3. It was stated on Page 11 that the Monte Carlo analyses use the same procedure in BAW-2308, Revision 1, suggesting that the results shown in Table 6 (Page 12) are based on all the specimens." Please confirm that the entire specimen set used in the analysis consists of 314 existing specimens, 8 added specimens for Heat 299L44, and 7 added specimens for Heat 61782. The sample size uncertainty shown in Table 7 (Page 13) indicates that the sample uncertainty results are based on uncensored specimens, i.e., 249 existing specimens and 14 added specimens. Provide the basis for this inconsistent approach of using different data sets for the Monte Carlo and sample size uncertainty analyses.

Response:

All the data (including the censored data) is included in the Monte Carlo data set. The procedure selects eight specimens at random from the data set. These eight specimens must have enough valid (uncensored) data to meet the criteria in ASTM E1921 section 10.4.1, if not then this simulation is discarded and another eight specimens is selected at random. This calculation of T_0 for eight randomly selected specimens is repeated until 1000 or 5000 T_0 calculations meet the ASTM E1921 section 10.4.1 validity criteria. The standard deviation is then calculated for these T_0 values and is shown for each heat in Table 6.

The σ calculation (Table 7) comes from ASTM E1921-05 section X4.2:

$$\sigma = \beta / \sqrt{r}$$

where r is defined as the total number of valid specimens used to establish the value of T_0.

Therefore, only the uncensored data is considered in this calculation.

The total population consisted of 329 data points with 263 that were not censored.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 20, 2007

RECEIVED
JUL 25 2007
OG PROJECT OFFICE

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: 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 (Agencywide Documents Access and Management System Accession No. ML070430445), the PWROG submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR BAW-2308, Revision 2, "Initial RT_{NDT} of Linde 80 Weld Materials." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On July 16, 2007, Ms. Christine DiMuzio, Project Manager, Licensing Subcommittee, and I agreed that the NRC staff will receive your response to the enclosed Request for Additional Information (RAI) questions by August 31, 2007. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1053.

Sincerely,

Holly D. Cruz

Holly D. Cruz, Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: RAI questions

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) BAW-2308-NP, REVISION 2

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

PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

PROJECT NO. 694

1. Loading rate effect on the reference temperature, T_0, is discussed in Section 3 of TR BAW-2308, Revision 2. It states on Page 10, "Using the above equations [(the proposed American Society for Testing and Materials (ASTM) E1921 equations)] to adjust the loading rates of the five data sets that were tested faster than 2 Mpa√m/s to the limit of 2 Mpa√m/s, results in a reduction of 0.9 °F in T_0. This model predicts a loading rate effect of 13.3 °F on T_0 from the slowest to the fastest loading rate extremes shown in Table 5, while the AREVA model predicts an effect of 22.9 °F."

   - Provide information regarding the calculation of the reduction of 0.9 °F in T_0 as a result of adjusting the loading rate to 2 Mpa√m/s for the five data sets mentioned in the quote. Explain the use of this reference loading rate of 2 Mpa√m/s here for the five data sets, while both the TR BAW-2308 and the proposed ASTM E1921 use a reference loading rate of 1 Mpa√m/s to derive their adjusted T_0 values. Further, "loading rate extremes shown in Table 5," which is part of the above quote, should be revised to "loading rate extremes shown in Table 4."

   - Provide information regarding the calculation of the loading rate effect of 13.3 °F on T_0 using the proposed ASTM E1921 equations and the effect of 22.9 °F on T_0 using the TR BAW-2308 model. Table 4-3 of TR BAW-2308, Revision 1, does not support the stated effect of 22.9 °F on T_0. Please clarify that this calculation is based on T_0 values for all data sets shown in Table 4, not just the five data sets.

2. There is inconsistent information on Pages 5, 6, 7, and 13 regarding the number of additional data that was used in the proposed analysis.

   - It state on Page 5 that, "Two 0.394 TC(T), four 0.500 TC(T), and two 0.936 TDC(T) specimens also from SA-1135 with an average fluence of 1.368 x 10^{19} n/cm^2 (E > 1 MeV) were also tested per ASTM E1921-02." Table 2 (Page 6) and Table 3 (Page 7) show, however, that there are three irradiated 0.936 DC(T) specimens from SA-1135. Please clarify the number of the 0.936 DC(T) specimens.

   - Table 7 (Page 13) uses the following wording to describe the revised sample size of the Linde 80 weld for Heat 299L44: "299L44 with 8 added tests from TMI2-LG2." This suggests that results from eight additional specimens are included in the proposed sample size uncertainty analysis. However, the Table 7 data show that the uncensored specimen number for 299L44 for the corresponding TR BAW-2308, Revision 1, analysis was 22 and the uncensored specimen number for 299L44 for the proposed analysis is 29, suggesting that seven additional uncensored specimens are included in the proposed analysis. Please confirm that one of the 8 added specimens is censored, and it is not used in any data analyses described in TR BAW-2308, Revision 2.

3. It states on Page 11 that the "Monte Carlo analyses were performed using the same procedure described in BAW-2308," suggesting that the results shown in Table 6 (Page 12) are based on all the specimens. Please confirm that the entire specimen set used in the analysis consists of 314 existing specimens, 8 added specimens for Heat 299L44, and 7 added specimens for Heat 61782. The sample size uncertainty shown in Table 7 (Page 13) indicates that the sample uncertainty results are based on *uncensored specimens, i.e., 249 existing specimens and 14 added specimens.* Provide the basis for this inconsistent approach of using different data sets for the Monte Carlo and sample size uncertainty analyses.

PWROG
PWR Owners Group

Program Management Office
20 International Drive
Windsor, Connecticut 06095

February 5, 2007

OG-07-47

BAW-2308-NP, Rev.2
Project Number 694

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

Subject: Pressurized Water Reactor Owners Group
<u>Submittal of BAW-2308-NP, Revision 2 "Initial RT_{NDT} of Linde 80 Weld Materials" PA-MSC-0229</u>

Reference 1: Letter, H. N. Berkow (NRC) to J. S. Holm (AREVA), "Final Safety Evaluation for Topical Report BAW-2308, Revision 1, 'Initial RT_{NDT} of Linde 80 Weld Materials' (TAC No. MB6636)," August 4, 2005 (ML052070408).

Reference 2: Letter, Howard Crawford (B&WOG Steering Committee) to NRC Document Control Desk, "Publication of BAW-2308(NP), Revision 1, "Initial RT_{NDT} of Linde 80 Weld Materials," November 17, 2005.

The Pressurized Water Reactor Owners Group (PWROG) is requesting formal review of BAW-2308 Revision 2 in accordance with the Nuclear Regulatory Commission (NRC) licensing topical report program for review and acceptance for referencing in licensing actions. BAW-2308 Revision 2 is applicable to the ANO 1, Crystal River 3, Davis Besse, Oconee 1-3, TMI 1, Point Beach 1-2, Surry 1-2, and Turkey Point 3-4 plants. Four paper copies of the report are being submitted with this letter.

BAW-2308 Revision 2 presents data for Linde 80 weld wire heat 61782 which was requested in the NRC safety evaluation (Reference 1) to BAW-2308, Revision 1-A (Reference 2). In addition, this report also updates all the IRT_{To} values and associated initial margin terms to account for an industry consensus approved loading rate correction which is in the process of being adopted into ASTM E1921.

Consistent with the Office of Nuclear Reactor Regulation, Office Instruction LIC-500, "Processing Request for Reviews of Topical Reports," the PWROG requests that the NRC provide target dates for any Request(s) for Additional Information and for issuance of the Safety Evaluation for BAW-2308, Revision 2.

D048

Correspondence related to this transmittal and invoices associated with the review of BAW-2308, Revision 2 should be addressed to:

Mr. Gordon Bischoff
Program Manager, PMO Office
Westinghouse Electric Company
Energy Center
P. O. Box 355
Pittsburgh, PA 15230-0355

If you have any questions, please do not hesitate to contact me at (630) 657-3897, or if you require further information, please contact Mr. Jim Molkenthin of the PWR Owners Group Project Management Office at (860) 731-6727.

Regards,

Frederick P. "Ted" Schiffley, II, Chairman
PWR Owners Group

FPS:JPM:las

Enclosures (4)

cc: M. Mitchell, USNRC
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BAW-2308 Revision 2

January 2007

INITIAL RT_{NDT} OF LINDE 80 WELD MATERIALS

Prepared for

PWR Owners Group

Participating Members:
Dominion Energy
Duke Energy Corporation
Entergy Operations, Inc.
Exelon Nuclear Corporation
FirstEnergy Nuclear Operating Company
Florida Power Corporation
Florida Power & Light Company
Nuclear Management Company

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AREVA

Executive Summary

This is a supplement to BAW-2308, Revision 1-A which was prepared for the PWR Owners Group (PWROG) to update alternative initial reference temperatures (IRT_{To}) for the Linde 80 beltline welds in the B&W fabricated reactor vessels. The alternative IRT_{To} values were determined based on brittle-to-ductile transition range fracture toughness test data of these weld metals obtained in accordance with ASTM Standard E1921 and using ASME Boiler and Pressure Vessel Code Case N-629. This report was prepared to provide data for Linde 80 weld wire heat 61782 which was requested in the NRC safety evaluation to BAW-2308, Revision 1-A. In addition, this report also updates the IRT_{To} values and associated initial margin terms to account for a consensus approved loading rate correction which is in the process of being adopted into ASTM E1921. This additional assessment revealed that the IRT_{To} values in the safety evaluation are conservative except for two IRT_{To} values for weld wire heat 72105 and heat 299L44. A licensee who wants to utilize the methodology of BAW-2308, Revision 1-A must request an exemption, per 10CFR50.12, from the requirements of 10CFR50.61 or 10CFR50 Appendix G. A license exemption has only been requested for Surry Units 1 and 2 using the lowered IRT_{To} values in BAW-2308, Revision 1-A, therefore none of the other applicable units are currently affected. In Surry Unit 1 heat 299L44 is the limiting material in terms of margin to the PTS (10CFR50.61) screening limit, however even with this increase in IRT_{To}, there is ample margin.

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1. Background

BAW-2308 was prepared for the B&W Owners Group (B&WOG) Reactor Vessel Working Group (RVWG) to justify alternative initial reference temperatures (IRT_{NDT}) for the Linde 80 beltline welds in the B&W fabricated reactor vessels.[a] The alternative IRT_{NDT} was determined based on brittle-to-ductile transition range fracture toughness test data of these weld metals obtained in accordance with ASTM Standard E1921 and using ASME Boiler and Pressure Vessel Code Case N-629. This report was submitted to the NRC for review and acceptance as a B&WOG topical for application to the pressurized thermal shock (PTS) rule (10CFR50.61) and 10CFR50, Appendix G, pressure-temperature limits.

Topical report BAW-2308 was submitted to the NRC in July 2002.[1] A request for additional information (RAI) was received in April 2003. A response was sent to the NRC and BAW-2308 Revision 1 was issued in August 2003 addressing the RAI.[2] In January 2004 an additional RAI was received. This RAI reflected a concern which was raised at an ASME code meeting regarding Code Case N-629. This additional RAI was addressed in June 2004. Through various communications with the NRC, additional information was provided in early 2005. The final safety evaluation (SE) was issued in August 2005.[3] The conclusions from the SE included two actions that must be addressed:

1. The B&WOG stated in their August 19, 2003 RAI response that fracture toughness data from 1 more heat of Linde 80 weld material (weld wire heat 61782) is to be obtained. The NRC staff expects B&WOG to evaluate this data to determine whether or not the conclusions of topical report BAW-2308 Rev. 1 and this SE are non-conservative and to communicate B&WOG's conclusion to the NRC staff. Non-conservatism in the BAW-2308 Rev. 1 report would be evident if 1) the IRT_{To} value from the to be tested Linde 80 weld wire heat turns out to be higher than the generic IRT_{To} value approved in this SE or 2) if the data from the to be tested Linde 80 weld wire heat results in an increase in the Linde 80 generic σ_I value.
2. Although the staff concludes that there is reasonable assurance that the use of IRT_{To} values for Linde 80 weld materials which were determined using the loading rate correction addressed in BAW-2308 rev 1 is acceptable for the purpose of reactor pressure vessel material property determination, the staff expects that action will be pursued within the appropriate consensus codes and standards organizations to address

[a] B&WOG merged with the Westinghouse Owners Group at the end of 2005 forming the PWROG. This report was prepared under funding from the PWROG.

loading rate effects on a more generic basis (or determine that they do not need to be addressed) in the appropriate ASME Code Cases and/or ASTM Standard Test Methods. The staff requests that the B&WOG revise the recommended values in BAW-2308, Revision 1 in accordance with Table 3. When consensus codes and standards organizations address loading rate effects on a more generic basis, the staff also expects that the B&WOG will re-evaluate BAW-2308, Revision 1 to determine whether or not revision of the topical report is warranted.[3]

An approved version of the topical report[4] was issued in 2005.[5] Virginia Electric and Power Company has requested an exemption from the requirements of 10CFR50.61 and 10CFR50 Appendix G to revise Surry 1 and 2 initial RT_{NDT} values using BAW-2308, Revision 1-A.[6] Nuclear Management Company, Exelon, FirstEnergy Nuclear Operating Company, and Florida Power and Light intend in the future to submit license exemption requests from 10CFR50.61 or 10CFR50 Appendix G for Point Beach Units 1 and 2, Three Mile Island Unit 1, Davis-Besse, and Turkey Point Units 3 and 4, respectively.

2. Test Results from Linde 80 Weld Wire Heat 61782

B&WOG Master Integrated Reactor Vessel Surveillance Program (MIRVP) capsule DB1-LG2 was irradiated in Davis-Besse from cycle 2 through cycle 11. This capsule contained specimens fabricated from weld wire heat 61782. Baseline and irradiated specimen testing was conducted and reported in the DB1-LG2 capsule report.[7] The capsule report was provided to the NRC for information in 2005.[8] This data was not available when BAW-2308 was written and was not included in the BAW-2308 analysis at the time.

The SA-1135 (wire heat 61782 and flux lot 8457) baseline specimens used to measure the transition temperature fracture toughness consisted of three 0.500 TC(T) and four 0.936 DC(T) specimens tested in compliance with the requirements of ASTM Standard E 1921-02. The test results are shown in Table 1. Two 0.394 TC(T), four 0.500 TC(T), and ~~two~~ three 0.936 TDC(T) specimens also from SA-1135 with an average fluence of 1.368×10^{19} n/cm^2 (E > 1 MeV) were also tested per ASTM E1921-02. See Table 2 for the irradiated specimen test results.

AREVA

Table 1. Baseline Fracture Toughness Data for Weld Metal SA-1135 (Oconee-2 nozzle drop-out; Wire Heat 61782)

Specimen Identification	Specimen Geometry	Test Temperature (°F)	J_c (in-lb/in^2)	K_{Jc} (ksi$\sqrt{in}$)	Violations
PW048	0.936 DC(T) [(a)]	-80	444	115.2	
PW049	0.936 DC(T)	-80	373	105.6	
PW050	0.936 DC(T)	-80	429	113.2	
PW051	0.936 DC(T)	-80	488	120.8	
PW021	0.5 TC(T) [(b)]	-110	397	109.2	
PW024	0.5 TC(T)	-110	459	117.4	
PW030	0.5 TC(T)	-110	190	75.5	

(a) Disk shaped compact fracture specimen.
(b) Compact fracture specimen.

Table 2. Fracture Toughness Data for Weld Metal SA-1135 (Wire Heat 61782) Irradiated to an Average Fluence of 1.368 x 10^{19} n/cm^2 (E > 1 MeV) in the DB1-LG2 Capsule

Specimen Identification	Specimen Geometry	Test Temperature (°F)	J_c (in-lb/in^2)	K_{Jc} (ksi$\sqrt{in}$)	Violations
PW008	0.394TC(T)	45	455	115.4	
PW006	0.394TC(T)	45	499	120.8	
PW028	0.5TC(T)	70	221	80.2	
PW027	0.5TC(T)	70	468	116.8	
PW020	0.5TC(T)	70	524	123.5	
PW026	0.5TC(T)	70	644	137.0	
PW045	0.936DC(T)	32	90.4	51.5	
PW040	0.936DC(T)	70	155	67.2	
PW041	0.936DC(T)	100	360	102.1	

The multi-temperature method (ASTM Standard E1921-05) of calculating the reference temperate was used to calculate the reference temperatures. Both data sets yielded valid T_0's as defined by ASTM Standard E 1921-05 as shown in Table 3. All validity criteria were met as specified in ASTM Standard E1921-05. No crack growth was observed on any of the fracture toughness test specimens. The optical crack length measurements were consistent with those determined from the compliance method. The rate adjusted T_0 values are adjusted to a loading rate of 1 MPa√m/sec as described in BAW-2308:

$$T_0\big|_{R2} = T_0\big|_{R1} + 5.33\,\ell n(R2/R1)$$

where: R1 and R2 are loading rates in units of and T_0 is in °C.

Table 3. Master Curve Reference Temperature (T_0) Data for Linde 80 Weld Wire Heat 61782

Weld	Average Fluence x 10^{19} n/cm^2 (E > 1 MeV)	Specimen Type	Number of Uncensored Specimens/ Minimum Required	T_0 (°F)	dK/dt (MPa √m/s)	Rate Adjusted T_0 (°F)	E1921-05 Validity
SA-1135 (Oconee-2 nozzle drop-out)	0.000	0.5TC(T) and 0.936DC(T)	7/6	-99.0	0.38	-89.8	Valid
SA-1135 (Oconee-2 nozzle drop-out) Irradiated in the DB1-LG2 Capsule	1.368	0.394TC(T), 0.5TC(T) and 0.936DC(T)	9/6	65.5	0.38	74.6	Valid

The IRT_{T_0} value for 61782 and the margin term are discussed below after considering the ASTM E1921 proposed loading rate effect.

3. Loading Rate Effect

BAW-2308 Revision 0 used rate adjusted T_0 values and margins to account for the various test loading rates (dK/dt) per an AREVA developed correlation.[9] BAW-2308 Revision 1 included T_0 and margin values that did not include any loading rate adjustment, since a rate adjustment had not been included in the consensus ASTM E1921 standard at the time. However, the unadjusted values turned out to be less conservative (lower T_0), in most cases, than the rate adjusted values reported in BAW-2308, Revision 0. Therefore the NRC accepted the use of the rate adjusted T_0 and margin values in the final safety evaluation. The NRC requested that the loading rate issue be reexamined when the loading rate effect was addressed in ASTM E1921.

The allowed quasi-static loading rate range in the E1921-97 and -02 versions was defined as the time to reach P_M which was restricted to 0.1-10 minutes. P_M is defined as 40% of the limit load, where the limit load is a function of the specimen size, geometry and the material yield strength. All the test data used in BAW-2308 is within the allowed loading rate range of E1921-97 and E1921-02. The ASTM E08 committee recognized that there is an effect of loading rate within the loading rate range defined in E1921-02, so they responded by reducing and redefining the allowable loading rate in the E1921-05 version. In E1921-05 the loading rate is limited to a dK/dt of 0.1 to 2 MPa$\sqrt{m}$/s during the elastic portion of specimen loading. In BAW-2308 the loading rate was reported as dK/dt, the same as defined in E1921-05. These dK/dt values are reported for each group of tests as an average value. The dK/dt of the various unirradiated Linde 80 data sets used to establish the RT_{T_0} values in BAW-2308, Revision 0 are listed in Table 4. Five groups of data exceed the E1921-05 loading rate restriction by a relatively small amount.

Table 4 Unirradiated Linde 80 Data Set Loading Rates

Weld-Wire	Weld Id	Specimen Type	dK/dt (MPa√m/s)
71249	SA-1094	PCS	0.22
72105	WF-70(B)	RPCS	0.22
72442	WF-67	Various CT	0.22
72442	WF-67	PCS	0.22
72445	SA-1585	PCS	0.22
299L44	WF-25	PCS	0.22
406L44	WF-112	PCS	0.22
406L44	WF-193	0.5TCT	0.22
406L44	WF-193	RPCS/PCS	0.22
821T44	WF-182-1	PCS	0.22
821T44	WF-182-1	0.5TCT	0.22
61782	SA-1135	Various CT	0.38
72442	SA-1484	PCS	1.21
299L44	WF-25 64W	PCCS	1.87
72105	WF-70(B)	1TCT	1.40
72105	WF-70(B)	1TCT	1.50
72105	WF-70(B) ORNL	Various	1.87
72105	WF-70(N) ORNL	Various	1.87
72445	SA-1585 65W	PCCS	1.87
299L44	WF-25 63W	PCCS	1.87
299L44	WF-25	0.5TCT	2.20
406L44	WF-193	0.5TCT	2.26
821T44	WF-182-1	0.5TCT	2.27
406L44	WF-112	0.5TCT	2.29
299L44	SA-1526	0.5TCT	2.35

The ASTM E08.08 committee is in the process balloting a change to E1921 which includes a method to adjust T_0 due to the loading rate effect. The loading rate adjustment is based on a paper written by K. Wallin:[10]

$$T_0 = \frac{T_{01} \cdot \Gamma}{\Gamma - \ln(\dot{K}_I)}$$

or for the loading rate induced temperature shift:

$$\Delta T_0 = \frac{T_{01} \cdot \ln(\dot{K}_I)}{\Gamma - \ln(\dot{K}_I)}$$

where:

$$\Gamma = 9.9 \cdot \exp\left\{\left(\frac{T_{01}}{190}\right)^{1.66} + \left(\frac{\sigma_{YS}}{722}\right)^{1.09}\right\}$$

and where:

T_{01} refers to the quasi-static loading rate of dK/dt = $\dot{K}_I$ = 1 MPa√m/s,

T_0 and T_{01} are in degrees Kelvin and σ_{YS} is in MPa.

Using the above equations to adjust the loading rates of the five data sets that were tested faster than 2 MPa√m/s to the limit of 2 MPa√m/s, results in a reduction of 0.9°F in T_0. This model predicts a loading rate effect of 13.3°F on T_0 from the slowest to the fastest loading rate extremes shown in Table 5, while the AREVA model predicts an effect of 22.9°F. The proposed E1921 loading rate model predicts a lower loading rate effect on T_0 than the AREVA model for the Linde 80 welds.

T_0 was calculated for each heat using the new loading rate equation being considered for E1921 (The precracked Charpy size (PCCS) bias used in BAW-2308 was also included). The results are tabulated in Table 5 along with the values reported in BAW-2308, Revision 1 for comparison.

Table 5 Multi-Temperature T_0 Calculation Results
(Italicized values are from BAW-2308, Revision 1 Table 4-4)

Heat	Adjusted T_0[a] (°F)		
	Without Rate Adjustment	BAW-2308 Rate Adjustment[c]	Proposed E1921 Rate Adjustment
406L44	*-138.6*	*-129.9*	-133.0
71249	*-97.0*	*-82.4*	-88.5
72105	*-64.1*	*-67.7*	-66.1
821T44	*-125.8*	*-115.2*	-119.2
299L44	*-113.8*	*-116.8*	---
299L44 with 8 added tests from TMI2-LG2	-107.7	---	-109.3
72442	*-72.8*	*-65.0*	-68.2
72445	*-108.2*	*-107.5*	-107.5
61782	-99.0	-89.8	-93.5
All 7 Linde 80 Heats[b]	***-110.5***	***-102.6***	---
All 8 Linde 80 Heats with 61782 and new 299L44 data[b]	**-109.4**	---	**-103.6**

a Adjustment includes PCCS bias correction as appropriate.

b All the data were combined for single multi-temperature calculation.

c Basis for IRT_{T_0} values approved in the SE (except 61782).

Eight additional baseline transition temperature tests were conducted on heat 299L44 as part of the TMI2-LG2 capsule testing in 2003 and are reported in the corresponding surveillance capsule report.[11] This data has been added to the 299L44 multi-temperature T_0 calculation with the results shown in Table 5. The resulting T_0 value with the new data and the new loading rate adjustment is 7.5°F higher than the 299L44 dataset considered in BAW-2308, Revision 0 and Revision 1.

4. Uncertainty Evaluation

Monte Carlo analyses were performed using the same procedure described in BAW-2308 using PCCS bias and the proposed E1921 loading rate adjusted data. The new data for the 299L44 heat was included. Analysis was also performed for the entire Linde 80 dataset combined including the new 299L44 data and the 61782 data. The results are reported in Table 6 along with the results from BAW-2308 Revision 1 for comparison.

There were 7 tests conducted to determine the initial T_0 for heat 61782. There is insufficient data to properly perform a Monte Carlo analysis since a minimum of 6 tests are required to calculate T_0. In addition, all the material was from one source and a Monte Carlo analysis would not properly reflect the material variability. Therefore, an estimate of σ_I is determined by using the average standard deviation from the other individual Linde 80 heats (See Table 6).

Table 6 Summary of Monte Carlo Analysis Results
(Italicized values are from BAW-2308, Revision 1 Table 4-5)

Heat	Adjusted Standard Deviation[a] (°F)		
	Without Rate Adjustment	BAW-2308 Rate Adjustment[b]	Proposed E1921 Rate Adjustment
406L44	*10.9*	*9.7*	10.4
71249	*7.3*	*7.2*	7.1
72105	*13.0*	*12.9*	13.4
821T44	*7.5*	*6.5*	7.0
299L44	*9.8*	*9.3*	---
299L44 with 8 added tests from TMI2-LG2	---	---	11.3
72442	*9.3*	*9.3*	9.7
72445	*6.7*	*7.5*	6.9
61782	---	---	9.4[c]
All 7 Linde 80 Heats	*20.0*	*17.1*	---
All 8 Linde 80 Heats with 61782 and new 299L44 data	---	---	17.9

a Adjustment includes PCCS bias correction as appropriate.

b Basis for IRT_{T_0} values approved in the SE.

c Average of the 7 other heat standard deviations with the proposed E1921 rate adjustment.

The sample size uncertainty is calculated using the same procedure as in BAW-2308, Revision 1. The results are reported in Table 7 with the data from BAW-2308, Revision 1 included.

Table 7 Sample Size Uncertainty
(Italicized values are from BAW-2308, Revision 1 Table 4-6)

Heat	**Number of Uncensored Specimens**	**1T $K_{Jc(med)}$ ($MPa\sqrt{m}$)**	**β**	**σ (°F)**
406L44	*39*	*103*	*18.0*	*5.2*
71249	*10*	*71*	*18.8*	*10.7*
72105	*121*	*105*	*18.0*	*2.9*
821T44	*24*	*102*	*18.0*	*6.6*
299L44	*22*	*107*	*18.0*	*6.9*
299L44 with 8 added tests from TMI2-LG2	29	103	18.0	6.0
72442	*21*	*82*	*18.8*	*7.4*
72445	*12*	*77*	*18.8*	*9.8*
61782	7	112	18.0	12.2
All 7 Linde 80 Heats	*249*	*95*	*18.0*	*2.1*
All 8 Linde 80 Heats with 61782 and new 299L44 data	263	96	18.0	2.0

The Monte Carlo standard deviation is combined with the sample size uncertainty using the square root of the sum of the squares in the same manner that was done in BAW-2308. The σ_I values are shown in Table 8 as well as the values from BAW-2308 for comparison.

Table 8 Summary of σ_I Values
(Italicized values are from BAW-2308, Revision 0 Table 4-7 or BAW-2308, Revision 1 Table 4-7)

Heat	σ_I Values (°F)		
	PCCS Adjusted	PCCS + BAW-2308 Rate Adj.[a]	PCCS + proposed E1921 Rate Adj.
406L44	*12.1*	*11.0*	11.6
71249	*13.0*	*12.9*	12.8
72105	*13.3*	13.2[b]	13.7
821T44	*10.0*	*9.3*	9.6
299L44	*12.0*	*11.6*	---
299L44 with 8 added tests from TMI2-LG2	---	---	12.8
72442	*11.9*	*11.9*	12.2
72445	*11.9*	*12.3*	12.0
61782	---	---	15.4
All 7 Linde 80 Heats[a]	*20.1*	*17.2*	---
All 8 Linde 80 Heats with 61782 and new 299L44 data[a]	---	---	18.0

[a] Basis for IRT_{To} values approved in the SE.

[b] 11.8F was used in the SE, which was based on an incomplete dataset from BAW-2308, Revision 0.

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5. Conclusions

The RT_{To} for each heat and all heats combined using the new loading rate equation being considered for ASTM E1921 and the new data described herein are tabulated in Table 9 along with the values approved in the SE for comparison. When combining the σ_I values in Table 9 and $\sigma_\Delta = 28°F$ with IRT_{To}, the values from the SE are conservative relative to the IRT_{To} and σ_I values presented in this revision, with the exception of heats 72105 and 299L44. Heat 72105 is non-conservative relative to the SE by 3.2°F and heat 299L44 is non-conservative relative to the SE by 8.5°F. In the Surry 1 license exemption request, heat 299L44 is the limiting material in terms of margin to the PTS (10CFR50.61) screening limit, however even with this increase in IRT_{To}, there is ample margin.[6]

Table 9. Heat Specific and Generic Initial RT_{To} with Associated Initial Margin

Linde 80 Heat	Values Approved in BAW-2308, Rev. 1 SE		With Proposed ASTM E1921 Loading Rate Adjustment	
	IRT_{To} (°F)	Initial Margin σ_I (°F)	IRT_{To} (°F)	Initial Margin σ_I (°F)
406L44	-94.9	11.0	-98.0	11.6
71249	-47.4	12.9	-53.5	12.8
72105	-32.7	11.8	-31.1	13.7
821T44	-80.2	9.3	-84.2	9.6
299L44	-81.8	11.6	-74.3[a]	12.8[a]
72442	-30.0	11.9	-33.2	12.2
72445	-72.5	12.3	-72.5	12.0
61782[b]	---	---	-58.5	15.4
Other heats	-47.6	17.2	-48.6[a]	18.0[a]

a New data included
b New heat

6. Certification

This report is an accurate and true description of the fracture toughness characterization of Linde 80 weld materials and the results are accurately reported. The conclusions described are based on the data analysis presented.

J. B. Hall J B Hall 1-5-07 Date
Materials and Structural Analysis

This report was reviewed and was found to be an accurate description of the work reported.

K. K. Yoon K. K. Yoon 1-5-07 Date
Structural and Fracture Mechanics

Verification of independent review.

B. R. Grambau 1/5/07 Date
Manager,
Materials and Structural Analysis

This report has been approved for release.

W. R. Gray 1/5/07 Date
Program Manager,
PWR Owners Group - Materials Subcommittee

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7. References

1. K. K. Yoon and J. B. Hall, "Initial RT_{NDT} of Linde 80 Weld Materials," AREVA Document BAW-2308, Revision 0, July 2002 (ML022200555).

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3. Letter, H. N. Berkow (NRC) to J. S. Holm (AREVA), "Final Safety Evaluation for Topical Report BAW-2308, Revision 1, 'Initial RT_{NDT} of Linde 80 Weld Materials' (TAC No. MB6636)," August 4, 2005 (ML052070408).

4. K. K. Yoon and J. B. Hall, "Initial RT_{NDT} of Linde 80 Weld Materials," AREVA Document BAW-2308, Revision 1-A, August 2005.

5. Letter, Howard Crawford (B&WOG Steering Committee) to NRC Document Control Desk, "Publication of BAW-2308(NP), Revision 1, "Initial RT_{NDT} of Linde 80 Weld Materials," November 17, 2005.

6. Letter, E. S. Grecheck (Virginia Electric and Power Company) to NRC Document Control Desk, "Virginia Electric and Power Company Surry Power Station Units 1 an 2 Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternative Materials Properties Basis per 10CFR 50.60(b)," June 13, 2006.

7. J. B. Hall and J. N. Byard, "Analysis of the B&W Owners Group Capsule DB1-LG2," AREVA document BAW-2486, November 2005.

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9. Hall, J. B. and Yoon, K. K., "Quasi-Static Loading Rate Effect on the Master Curve Reference Temperature of Ferritic Steels and Implications," ASME PVP-Vol. 461 Fatigue Fracture and Damage Analysis, 2003.

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Summary of Change to BAW-2308 Revision 2

Page 5, last paragraph, was corrected per RAI response #2.

No other changes were made.

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