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Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads

BAW-2192
Revision 0
Supplement 1NP-A
Revision 0

Topical Report

December 2018

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

December 7, 2018

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATIONS FOR BAW-2192, SUPPLEMENT 1, REVISION 0, "LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVEL A&B SERVICE LOADS" AND BAW-2178, SUPPLEMENT 1, REVISION 0, "LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVEL C&D SERVICE LOADS"

Dear Mr. Nowinowski:

By letter dated December 15, 2017, as supplemented by letters dated June 15, 2018 and June 29, 2018, the Pressurized Water Reactor Owner's Group (PWROG) submitted BAW-2192, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads," and BAW-2178, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C&D Service Loads," for review and approval by the U.S. Nuclear Regulatory Commission (NRC).

The NRC staff has completed its review of the subject topical reports (TRs). The staff concludes that BAW-2192, Supplement 1, Revision 0 and BAW-2178, Supplement 1, Revision 0 demonstrate, for the seven plants within the scope of the TRs, that there is adequate margin of safety against ductile fracture in the RPV welds for Service Level A and B loads, and for Service Level C and D loads through 80 calendar years of operation. The staff also concludes that both TRs may be referenced in subsequent license renewal applications for the plants within scope of the report, as a basis for demonstrating that the USE TLAA has been projected in accordance with Title 10 *Code of Federal Regulations* (CFR) 54.21(c)(1)(ii), for Linde 80 welds in those plants.

Individual licensees wishing to reference either TR as the basis for demonstrating compliance with the 10 CFR 50, Appendix G requirements related to USE must generate 80-year neutron fluence at RPV locations in accordance with NUREG-2192, "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants," to demonstrate that the fluence estimates provided in Table 3-1 are applicable to their plants. This is Applicant/licensee Action Item 1.

NOTICE: The enclosure transmitted herewith contains Proprietary Information. When separated from the enclosure, this transmittal document is decontrolled.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish approved versions of TRs BAW-2192, Supplement 1, Revision 0 and BAW-2178, Supplement 1, Revision 0, within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final safety evaluation (SE) after the title page. Also, they must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

As an alternative to including the request for RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of these TRs, PWROG will be expected to revise the TRs appropriately or justify their continued applicability for subsequent referencing. Licensees referencing these TRs would be expected to justify their continued applicability or evaluate their plant using the revised TRs.

If you have any questions, please contact Jason Drake at 301-415-8378.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 99902037

Enclosures:
1. Safety Evaluation of BAW-2192, Supplement 1, Revision 0
2. Safety Evaluation of BAW-2178, Supplement 1, Revision 0

SUBJECT: FINAL SAFETY EVALUATIONS FOR BAW-2192, SUPPLEMENT 1, REVISION 0, "LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVEL A&B SERVICE LOADS" AND BAW-2178, SUPPLEMENT 1, REVISION 0, "LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVEL C&D SERVICE LOADS" DATED DECEMBER 7, 2018

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO PRESSURIZED WATER REACTOR OWNERS' GROUP LICENSING TOPICAL REPORT BAW-2192, SUPPLEMENT 1, REVISION 0, "LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVELS A&B SERVICE LOADS"

PROJECT NO. 694

Enclosure 1

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO PRESSURIZED WATER REACTOR OWNERS' GROUP

LICENSING TOPICAL REPORT BAW-2192, SUPPLEMENT 1, REVISION 0,

"LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR

VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVELS A&B

SERVICE LOADS"

PROJECT NO. 694

1.0 INTRODUCTION

By letter dated December 15, 2017 (Ref. 1), as supplemented by letters dated June 15, 2018 (Ref. 2), and June 29, 2018 (Ref. 3), the Pressurized Water Reactor Owner's Group (PWROG) submitted BAW-2192, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Levels A&B Service Loads," (Ref. 4, the topical report (TR)) for review and approval by the U.S. Nuclear Regulatory Commission (NRC).

The TR is a supplement to BAW-2192PA, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads" (Ref. 5) dated April 30, 1994. The original report evaluated low upper shelf toughness for reactor pressure vessels (RPVs) of pressurized water reactors (PWRs) that have welds made with Linde 80 weld metal, a type of weld metal known to be susceptible to having projected irradiated upper shelf energy (USE) less than the Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G criterion of 50 foot-pounds (ft-lbs). The original report was applicable to 40 calendar years of operation, encompassing the original 40-year license for the plants. The new TR is intended to be applicable to 80 calendar years of operation, thus covers a subsequent period of extended operation (SPEO) for the plants. The original report was applicable to 16 units, while the revised report is applicable to seven units; all the units covered by the new TR are also covered by the original report. The plants within the scope of the TR are Oconee Nuclear Station, Units 1, 2, and 3 (ONS-1, -2, and -3), Surry, Units 1 and 2 (Surry 1 and 2), and Turkey Point Plant, Units 3 and 4 (TP3 and TP4). In addition to accounting for 80 years of operation, the TR also includes extended beltline materials (e.g., those that will accumulate at least $1x10^{17}$ n/cm^2 by the end of the SPEO, but were not included in the original USE evaluation).

The NRC approved BAW-2192 by a safety evaluation (SE) dated March 29, 1994 (Ref. 6) which is included in BAW-2192PA.

2.0 REGULATORY EVALUATION

The regulation in 10 CFR Part 50, Appendix G provides the staff's criteria for maintaining acceptable levels of USE for the RPV beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires RPV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the licensed period of operation of the facility, unless it can be demonstrated

through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Code. Such analyses are referred to as "equivalent margins analyses," or EMAs. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant RPV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H RPV material surveillance program.

For evaluations of low-upper shelf toughness, NUREG-0800 Section 5.3.2, "Pressure-Temperature Limits, Upper Shelf Energy, and Pressurized Thermal Shock," states that in addition to the ASME Code, Regulatory Guide (RG) 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," provides an acceptable methodology for the performance of analyses intended to meet the provisions for additional analysis specified in 10 CFR Part 50, Appendix G paragraph IV.A.1.a (which addresses low-upper shelf toughness).

USE is considered a time-limited aging analysis (TLAA) in accordance with 10 CFR 54.3. The "Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants Final Report", NUREG-2192, provides guidelines for review of the USE TLAA in Section 4.2.2.1.2.

The NRC RG 1.161 provides guidance for acceptable methods of evaluating low USE, and states that the analytical methods described in ASME Section XI, Appendix K (Appendix K), provide acceptable guidance for evaluating reactor pressure vessels when the Charpy USE falls below the 50 ft-lb limit of Appendix G to 10 CFR Part 50. However, Appendix K does not provide information on the selection of transients and gives very little detail on the selection of material properties. However, RG 1.161 provides guidance for selecting transients for Service Level A, B, C, and D conditions, and models for determining the J-R curves for various classes of RPV materials, including Linde 80 welds.

3.0 TECHNICAL EVALUATION

3.1 PWROG Evaluation

The TR covers the following major topics:

Section 1.0 of the TR discusses the scope and purpose of the TR. Section 1.0 also identifies the current licensing basis (CLB) of the in-scope plants, or EMA analysis of record. All the plants relied on BAW-2192PA for the EMA for 40 years. For 60 years, ONS-1, -2, and -3 relied on an analysis in BAW-2251A, while Surry 1 and 2 and TP3 and TP4 relied on a plant-specific analysis.

Section 2.0 of the TR addresses regulatory requirements and ASME Code requirements. Section 2.0 also notes that Selection of the limiting design transient (i.e., cooldown at 100 F/h) is consistent with BAW-2192PA, Section 5.3. TR Section 2.2 indicates that the analyses in the TR were performed in accordance with the 2007 Edition through 2008 Addenda of the ASME Code, Section XI, Appendix K. A reconciliation to the 2013 Edition of ASME Section XI, Appendix K (i.e., current edition required per 10 CFR 50.55a and Appendix G) is provided in Section 2.2 of the TR.

Section 3.0 provides a description of the RPVs for each plant, including Table 3-1 listing each weld, and diagrams showing the weld locations in each RPV. For each weld, the table provides the description of the location, the material identification number and heat number, the estimated 80-year neutron fluence, and the copper (Cu) and nickel (Ni) content of each weld.

Section 3.0 also states that all plants that reference [the TR] must generate 80-year neutron fluence at RPV locations in accordance with NUREG-2192, SLR-SRP, to demonstrate that the fluence estimates provided in Table 3-1 are applicable to their plants. Section 3.0 also states that copper and nickel content of the RPV shell welds is consistent with EMA analyses of record reported in Section 1.1; the copper and nickel content for transition welds and RPV nozzle-to-nozzle belt forging welds reported in Table 3-1 were obtained from either the EMA analysis of record or a search of ONS-1, -2, and -3; Surry 1 and 2; and TP3 and TP4 RPV fabrication reports.

Section 4.0 describes the J-integral resistance model used in the evaluation. The model is a vendor-specific model for Linde 80 welds. The TR uses the same model which was used in BAW-2192PA. Section 4.0 refers to Appendix A of the TR for a discussion of the extension of the applicability of the model to higher fluences (since the model was originally developed for 40 years of operation).

Section 4.0 also contains the mechanical properties of the RPV welds and base metal, both before and after irradiation. The TR states the irradiated mechanical properties used are consistent with those used in the EMA evaluations for 60 years.

Section 5.0 describes the fracture mechanics evaluation method, which is in accordance with Appendix K. Section 5.0 also provides the detailed results of the fracture mechanics evaluation. Two analyses are done for each weld: an evaluation of flaw extension and an evaluation of flaw stability. In the evaluation of flaw extension the ratio of the J-integral resistance at 0.10 inch crack extension ($J_{0.1}$) to the applied J-integral (J_1), with a safety factor of 1.15 applied on pressure, is calculated and must be greater than 1.0. In the evaluation of flaw stability, the J-R curve and the applied J-integral with a safety factor of 1.25 on pressure are plotted as a function of flaw extension. At the point of intersection of the two curves, the flaw is considered stable if the slope of the applied J-integral curve is less than that of the J-R curve.

The results are reported for each plant in TR Section 5.3 and 5.4, for the RPV shell welds, the RPV transition welds, and the RPV nozzle welds.

Section 6.0 contains the summary and conclusions. The conclusion for all seven units is that all welds met the acceptance criteria of Appendix K for Service Level A and B loadings. The conclusions for the limiting weld are detailed for each plant, including the ratio of $J_{0.1}$ to J_1.

Appendix A of the TR describes the development of the J-R model used in BAW-2192PA, which is also used in the TR.

Appendix A of the TR describes the development of the new model 6B, which incorporates additional higher fluence J-R test data. Appendix B also describes how the new model was compared to the original model, and how it was determined that the original model was adequate for use out to 80 years.

3.2 NRC Staff Evaluation

The NRC staff's review focused on:

- The demonstration of the applicability of the original J-R model for 80 calendar years of operation.
- Verification that the PWROG's analysis was done in accordance with the Appendix K, which is an NRC approved method.

3.2.1 Applicability of the Original Model to 80 Years

[

]

TR Figure A-3 plots original and new data and model fit normalized at standardized conditions versus Δa (change in crack size). It shows that [

] Therefore, in RAI 2, the NRC staff requested that the PWROG explain why it is valid to use the proposed J-R curve, [ ] In its June 15, 2018, response (Ref. 2), the PWROG indicated that BAW-2192PA, Figure B-13, used bounds of +/- 3 standard errors (Se), [

]

The NRC staff reviewed Figure 2-1 of the RAI response, and agrees that [

] RAI 2 is thus resolved.

Since the PWROG's J-R model bounds most of the original and new data, and includes data with fluences close to the highest predicted 1/4T fluence for the RPVs in scope of the TR, the staff finds the use of the model acceptable for the 80-year EMA.

3.2.2 Consistency with ASME Code, Section XI, Appendix K

Appendix K provides acceptance criteria and evaluation procedures for determining acceptability for operation of RPVs when the vessel metal temperature is in the upper shelf range. The methodology is based on the principles of elastic-plastic fracture mechanics. Flaws shall be postulated in RPV locations of predicted low USE, and the applied J-integral for these flaws shall be calculated and compared with the J-integral fracture resistance of the material to determine acceptability.

Appendix K does not contain a specific model for determining J_R. Instead, Paragraph K-3300 specifies that the J_R curve must be generated based on accepted test procedures, a database obtained from the same class of material, or an indirect method provided the method is justified for the material.

For Service Level A and B conditions, for evaluation of weld materials, Appendix K, K-2200 specifies postulation of an interior semi-elliptical flaw with a depth to thickness (a/t) ratio of 0.25, and an aspect ratio (length to depth) of 6, with the flaw's major axis oriented along the weld of concern, and the flaw plane oriented in the radial direction. The flaw must satisfy two acceptance criteria:

(1) The applied J-integral evaluated at a pressure 1.15 times the accumulation pressure as defined in the plant specific Overpressure Protection Report, with a structural factor of 1 on thermal loading for the plant specific heatup and cooldown conditions, shall be less than the J-integral of the material at a ductile flaw extension of 0.1 in. (2.5 mm).

(2) Flaw extensions at pressures up to 1.25 times the accumulation pressure of K-2200(a)(1) shall be ductile and stable, using a structural factor of 1 on thermal loading for the plant specific heatup and cooldown conditions. *(b)* The J-integral resistance versus flaw extension curve shall be a conservative representation for the vessel material under evaluation.

To evaluate the second acceptance criteria, Appendix K provides three different methods.

The J-R curve - crack driving force diagram procedure of K-4310 is the most commonly used of the three methods, and is the only one endorsed by RG 1.161. This method was used in the TR. In this method, flaw stability shall be evaluated by direct application of the flaw stability rules in K-3400. The applied J-integral shall be calculated for a series of flaw depths corresponding to increasing amounts of ductile flaw extension. The applied J-integral for Levels

A and B Service Loadings shall be calculated using the procedures provided in K-4210. The applied pressure p shall be equal to the accumulation pressure for Levels A and B Service Loadings, P_a; and the structural factor (SF) on pressure shall be 1.25. The applied J-integral shall be plotted against crack depth on the crack driving force diagram to produce the applied J-integral curve, as illustrated in Fig. K-4310-1. The J-R curve shall be plotted on the crack driving force diagram.

The NRC staff finds the methodology of determination of the Service Level A and B loadings for the RPV shell welds described in the TR to be consistent with the methodology described in Appendix K, and is therefore acceptable.

The methods of Appendix K are only applicable to RPV shell regions, not nozzles and transition areas. Section 5.2 of the TR states, "For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis." In RAI 4, the NRC staff requested the PWROG to identify the plant, for which the three-dimensional (3-D) finite element analysis (FEA) was performed, and explain why this plant-specific FEA is applicable to Surry 1 and 2 and TP3 and TP4 RVs. In its June 15, 2018, response (Ref. 2), the PWROG stated that the stress analysis consists of two separate 3-D FEA models: one for the TP3 and TP4 and the other for Surry 1 and 2. The PWROG further stated that each model used appropriate plant specific geometry, material property, nozzle loads, and plant operating conditions. Since the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from 3-D FEAs for TP3 and TP4 and Surry1 and 2 RVs, the NRC staff finds the response to RAI 4 acceptable.

With regard to the ONS-1, -2 and -3 nozzle-to-vessel welds, Section 5.3.2 of the TR states that the applied stress intensity factor (K_I) is conservatively calculated for the RPV nozzle-to shell weld using a flat plate solution by Newman and Raju. In RAI 5, the staff requested:

- the R/t ratio for this weld
- the percent difference in the applied K_I by using a flat plate solution instead of a cylindrical plate solution

In its June 29, 2018, response to RAI 5 (Ref. 3), the PWROG stated that Section 5.3.2 of the TR applies only to the ONS-1, -2, and -3 RPV nozzle-to-shell welds. The PWROG provided the R/t ratio for this weld. The PWROG also stated that 10 CFR Part 50 Appendix G requires the equivalent margins analysis use the latest edition and addenda of the ASME Code at the time the analysis is submitted (i.e., ASME Code Section XI, 2013 Edition). The NRC staff notes that the ASME Code, Section XI, 2013 Edition, Appendix A does not provide cylindrical shell solutions for inner diameter flaws; however, the ASME Code, Section XI, 2017 Edition provides solutions for inner diameter axial flaws in cylindrical shells. The PWROG provided a comparison of the K_I for a cylindrical shell and for a flat plate. The staff reviewed and verified the PWROG's calculated K_I based on the cylindrical shell solution in Section XI, Appendix A, 2017 edition. The review indicated that the percent difference in applied K_I calculated by the PWROG for both solutions is relatively small, with the flat plate solution being more conservative. Since the flat plate solution used by the PWROG is conservative, the NRC staff finds the PWROG's response to RAI 5 acceptable.

3.2.3 Selection of Service Level A and B Transients

The PWROG selected a 100 °F/hr cooldown as the limiting transient for Service Level A and B transients. This is the same as the limiting transient used in BAW-2192PA, which was approved by the NRC staff in its SE of that report. Since there is no reason for the limiting transient to change for 80 years of operation versus 40 years of operation, the NRC staff finds the selection of the limiting transient to be acceptable.

3.2.4 Evaluation of Extended Beltline Materials

The TR evaluates all the weld materials for each in-scope plant that will receive $1x10^{17}$ n/cm^2 (E>1.0 MeV) by the end of the subsequent period of extended operation (80 years). As such, some of the welds evaluated in the TR were not included in the 40-year EMA and are outside of the cylindrical shell beltline region of the vessel. These include nozzle-to-vessel welds and the upper and lower transition welds. The equations for determining J_1 in Appendix K are not applicable to these welds.

For the ONS RPVs, the TR states that the effect of structural discontinuity at the transition welds (i.e., "upper weld" at Unit 1 connecting upper end of the intermediate shell to the lower nozzle belt forging & "lower weld" connecting the lower end of the lower shell to the Dutchman forging) is accounted for by applying a scaling factor of **[ ]** to the applied J-integral calculated by the above procedure for circumferential welds. The TR states that this approach is based on a previous finite element analysis of an applicable axisymmetric B&W-designed RPV shell model that included a detailed description of the transition regions. The TR states that the stresses due to pressure and thermal loads at the nozzle-to-shell welds are obtained from a previous axisymmetric analysis of the outlet nozzle of a B&W-designed plant that is also deemed applicable to the ONS RPVs. The TR states that for both the Surry 1 and 2 and TP3 and TP4 RPVs, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis. The TR also stated that based on the results of analysis performed for ONS it was deemed that the effects of structural discontinuities at the lower transition welds need not be explicitly addressed. However, TR Section 5.3.2 indicates that the lower head thickness, which is thinner than the RPV shell regions, was used to evaluate the lower transition welds for TP, which results in a higher stress intensity due to pressure. For Surry 1 and 2, the TR indicates that because the lower transition welds are approximately four feet below the bottom of the core, these welds will receive fluence less than $1x10^{17}$ n/cm^2, therefore were not evaluated.

The NRC staff finds the method used by the PWROG to determine the applied J_1 for the welds outside the cylindrical shell region of the RPVs, to be acceptable. For TP, not accounting explicitly for the structural discontinuity at the lower transition weld is acceptable, because the scaling factor for the loads for this weld should be similar to that for ONS, which was fairly small. Also, the results for these welds show large margins.

3.2.5 Inputs

3.2.5.1 <u>Material Chemistry</u>

The staff checked the PWROG's Cu and Ni values for each material against the Cu and Ni values in each plant's CLB. The staff found the Cu and Ni values to be consistent with the CLB

for each plant. The most recent Cu and Ni values for the beltline materials of ONS-1, -2, and -3 are reported in Reference 7, and approved in Reference 8. For Surry 1 and 2, the Cu content was verified to be consistent with the EMA analysis of record in Reference 9, approved in Reference 10. For TP3 and TP4, the staff verified the Cu and Ni are consistent with License Amendment Request (LAR) for Extended Power Uprate (EPU), Attachment 4, L-2010-113 (Ref. 11), which was approved in Reference 12, except the staff noted the Cu and Ni values of 0.23 % Cu and 0.59 % Ni for Heat No. 71249, used in the TP3 inlet nozzle to RPV weld according to TR Table 3-1, are not consistent with Reference 10, Table 2.1.2-1. In Reference 11, Table 2.1.2-1 (TP3) and Table 2.1.2-2 (TP4), the three heats of material associated with the inlet/outlet nozzle welds contain 0.34 % Cu, 0.68% Ni; 0.16 % Cu, 0.57% Ni; and 0.19% Cu, 0.57% Ni. The first pair is not consistent with the TR, while the last two are consistent. Therefore, in RAI 6, the NRC staff requested the PWROG to resolve this discrepancy. The PWROG's June 15, 2018, response to RAI 6 (Ref. 2), stated that several weld wire heats were used when fabricating the TP3 and TP4 inlet and outlet nozzle welds. In the EPU LAR, because the weld wire heats applicable to each nozzle weld were unknown at that time, all the weld wire heats were listed. The PWROG further indicated that the material containing 0.34% Cu and 0.68% Ni was only used in TP4. With respect to the heat with 0.23% Cu and 0.59% Ni (Heat # 71249), the PWROG stated the presence of this weld heat in the TP3 nozzle welds was discovered in a 2017 records search while developing the TR. Based on the PWROG's clarifications, the NRC staff finds that RAI 6 is resolved.

3.2.5.2 Neutron Fluence Values

The TR states that the fluence projections are reported for all RPV weld locations that are expected to exceed 1.0×10^{17} n/cm^2 at 80 years for the participating plants. The TR further states that all plants that reference this report must generate 80-year neutron fluence at RPV locations in accordance with NUREG-2192, SLR-SRP, to demonstrate that the fluence estimates provided in Table 3-1 are applicable to their plants. The NRC staff notes that guidance on determination of neutron fluence for time-limited aging analyses is contained in Section 4.2 of NUREG-2192.

3.2.5.3 Mechanical Properties

TR Section 4.2 includes the mechanical properties of the RPV weld metal used in the analysis (other than the J-R curves). These include the elastic modulus, coefficient of thermal expansion, and yield strength. The TR states the irradiated material properties used are consistent with those used for the plants 60-year license renewal low upper shelf toughness analysis submittals. The only irradiated property used is the yield strength. Use of the 60-year irradiated yield strength is conservative since the yield strength would increase with an additional 20 years of irradiation. However, a lower yield strength results in a larger J_I, therefore it is conservative.

3.2.6 Confirmatory Calculations

The NRC staff performed confirmatory calculations using the methodology of Appendix K, and the PWROG model for fracture toughness, on several different materials from several different plants. The staff also used the copper-fluence fracture toughness model for Linde 80 welds from NUREG/CR-5729 for comparison to B&WOG Model 4B. This independent evaluation using the NUREG/CR-5729 copper-fluence model is restricted to the plants within the scope of

the TR. The staff calculated the margin with respect to flaw extension in terms of $J_{0.1}/J_1$ with a structural factor of 1.15 on pressure, and found that the margin is acceptable (> 1.0) for both the PWROG model and the NUREG/CR-5729 model.

Also, the NRC staff checked the flaw stability using the crack driving force diagram in accordance with K-4310, with a structural factor of 1.25 on pressure. The results showed the slope of the J_1 curve to be less than the slope of the J-R curve at the point of intersection, therefore meeting the acceptance criteria of Appendix K.

Figure 1 shows an example of the crack driving force diagram method from the staff's confirmatory calculations for ONS-1, for the limiting shell weld (lower shell longitudinal weld SA-1073). In the figure, the horizontal axis shows crack extension, and the vertical axis shows the J-integral in units of inch-pounds per square inch. The blue (top) curve is the J_R curve from the TR J-integral resistance model, and the lower orange curve is the J_R curve from the NUREG/CR-5729 copper-fluence model for Linde 80 welds. The straight gray line is J_1 as a function of crack extension. Since the slope of the J_1 line is less than the slope of the J_R curve at the point of intersection, the second criteria of K-2200 is met.
[

]

4.0 CONCLUSIONS

The staff concludes that BAW-2192, Supplement 1, Revision 0 demonstrates, for the seven plants within the scope of the TR, that there is adequate margin of safety against ductile fracture in the RPV welds for Service Level A and B loads, through 80 calendar years of operation. The NRC staff also concludes that the TR may be referenced in SLRAs for the plants within scope of the report, as a basis for demonstrating that the USE TLAA has been projected in accordance with 10 CFR 54.21(c)(1)(ii), for Linde 80 welds in those plants.

The NRC staff identifies one condition for applicants or licensees wishing to reference the TR. Individual applicants or licensees wishing to reference this TR as the basis for demonstrating compliance with the 10 CFR Part 50, Appendix G requirements related to USE must generate 80-year neutron fluence at RPV locations in accordance with NUREG-2192, SLR-SRP, to demonstrate that the fluence estimates provided in Table 3-1 are applicable to their plants.

5.0 REFERENCES

1. PWR Owners Group - Submittal of BAW-2192-P, Revision 0, Supplement 1 and BAW-2178-P, Revision 0, Supplement 1 (PA-MSC-1481), December 15, 2017 (ADAMS Accession No. ML17354A011).

2. Transmittal of the Response to Request for Additional Information, - RAIs 2, 4, and 6 Associated with BA W-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," - RAIs 7, 8, and 10 Associated with BAW-2178-P Supplement 1, Revision O, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" PA-MSC-1481, June 15, 2018 (ADAMS Accession No. ML18170A107).

3. Transmittal of the Response to Request for Additional Information: RAI 5 Associated with BAW-2192-P, Supplement 1, Revision 0, "Low Upper- Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," - RAI 9 Associated with BAW-2178-P Supplement 1, Revision 0, "Low Upper - Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads"(PA-MSC-1481), June 29, 2018 (ADAMS Accession No. ML18184A078).

4. BAW-2192P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," December 31, 2017 (ADAMS Accession No. ML17354A013.

5. "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads," April 30, 1994 (ADAMS Legacy Accession No. 9406240263).

6. Letter from Brian W. Sheron to George Lehman, Re: Acceptance for Reference of "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads," March 29, 1994 (ADAMS Legacy Accession No. 9404220112).

7. Oconee Nuclear Station Units 1, 2 & 3 - ANP-3127, Revision 2, "Pressure-Temperature Limits at 54 EFPY," Enclosure 1. September 30, 2013 (ADAMS Accession No. ML13305A121).

8. Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Revised Pressure-Temperature Limits (TAC Nos. MF0763, MF0764, MF0765). February 27, 2014 (ADAMS Accession No. ML14041A093).

9. Surry Power Station, Units 1 & 2, Update to NRC Reactor Vessel Integrity Database and Exemption Request for Alternate Material Properties Basis Per 10 CFR 50.60(b), June 13, 2006 (ADAMS Accession No. ML061650080).

10. Surry Power Station, Units 1 & 2 - Issuance of Amendments Regarding Reactor Vessel Heatup and Cooldown Curves for 48 Effective Full-Power Years (TAC Nos. ME3920 and ME3921), May 31, 2011 (ADAMS Accession No. ML11110A111).

11. Turkey Point, Units 3 and 4 - License Amendment Request for Extended Power Uprate, Attachment 4; Licensing Report, December 14, 2010 (ADAMS Accession No. ML103560177).

12. Turkey Point Units 3 And 4 - Issuance of Amendments Regarding Extended Power Uprate (TAC Nos. ME4907 AND ME4908 (ADAMS Accession No. ML11293A365).

Principal Contributor: Jeffrey Poehler, NRR/DMLR/MVIB

Date: November 30, 2018

COMMENT RESOLUTION TABLE

BAW-2192, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads" & BAW-2178, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads"

[The comment and resolution responses regarding BAW-2178 have been removed from the following table since they are not applicable to this TR.]

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
1.0	Introduction						
		1, 12	MR	E	Title for BAW-2192, Supplement 1, Revision 0, should be ..."Levels A & B" vs "Level A&B" The title is correct for BAW-2192PA, Revision 00	NA	Comment incorporated.
		1, 31	MR	E	n/cm2 should be n/cm^2	NA	Comment incorporated.
2.0	Regulatory Evaluation						
		1, 39	AN	E	"Regulation at" should be Regulation in"		Comment incorporated.
		2, 19	MR	E	NUREG-2192 (SRP-SLR), Section 4.2.2.1.2 provides guidance for USE vs NUREG-2191, which references Section 4.2.2.1.2 of the SRP-SLR	NA	Comment incorporated.
3.1	PWROG Evaluation						

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
		2, 44	MR	T	The TR was performed in accordance with ASME Section XI, 2007 Edition, with 2008 Addenda; however, a reconciliation to the 2013 Edition of ASME Section XI is discussed in Section 2.2 of the TR. This reconciliation should be acknowledged in the SE since 10 CFR 50, Appendix G requires that the current 10 CFR 50.55a Edition of ASME Section XI shall be used, which is the 2013 Edition of ASME Section XI.	NA	Comment incorporated.
		2, 46	AN	E	E: "essentially identical to" instead of "are identical in"		Comment incorporated.
		3,35	MR	E	Suggest TR Sections 5.3 and 5.4 versus TR Section 5.4. Section 5.3 presents flaw extension results and 5.4 flaw stability results.	NA	Comment incorporated.
		3,45/46	MR	E	Appendix B should be revised to Appendix A and model 6 should be Model 6B	NA	Comment incorporated.
3.2.1	Applicability of the Original Model to 80 Years						
		4, 15	AN	E	"0.# inch" not "0.#" without any units	<-- proprietary value replaced	Comment incorporated.

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
						with #, see mark-up of DSE for values	
		4,19	AB	E	#X10^{19} should be changed to #.##X10^{19} or state #.##X10^{19} , which is approximately #X10^{19}	This section is appropriately marked as proprietary <-- proprietary value replaced with #, see mark-up of DSE for values	Comment incorporated.
		4,21	AB,MR	E	Suggest replacing "However, the 1/4t fluences…" with "However, one quarter thickness 80-year fluence estimates are approximately #.# E+19 n/cm^2 for ONS, #.#0E+19 n/cm^2 for Surry, and #.##E+19 n/cm^2 for Turkey Point." Text is from TR Appendix A, Page A-4. Based on review of 1/4t ARTs for TP and Surry (source references for the TRs), only TP3 and TP4 each have one weld with 1/4T fluence > #.# E19; all remaining welds in the TR are less than #.#E19 at 1/4t.	This section is appropriately marked as proprietary <-- proprietary value replaced with #, see mark-up of	Comment incorporated.

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
						DSE for values	
		4,25	AB,MR	T	Revise the sentence as follows. Plots of the normalized J versus crack extension (Δa) for models 4B and 6B show that the model curves are very similar, with model 6B being slightly more conservative (i.e., lower values of Jd) at smaller crack extensions (<0.1 inch) and slightly more less conservative (i.e., higher values of Jd) at high crack extensions (> 0.# inch).	This section is appropriately marked as proprietary <-- proprietary value replaced with #, see mark-up of DSE for values	Comment incorporated.
		4,35-37	MR	E	In general bounds are either ± #Se or ± #SE in this paragraph.	This section is appropriately marked	Comment incorporated.

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
						as proprietary <-- proprietary value replaced with #, see mark-up of DSE for values	
3.2.2	Consistency with ASME Code, Section XI, Appendix K						
		5,44	MR,AN	E	Revise sentence--This method was used in the TR.	NA	Comment incorporated.
3.2.3	Selection of Service Level A and B Transients						
		6, 47	MR,AN	E	Symbol "F" for fahrenheight missing		Comment incorporated.
3.2.4	Evaluation of Extended Beltline Materials						

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
		7,34	MR,AN	E	Revise text as follows "For ~~Surry and~~ TP, not..." The lower transition weld was not evaluated for Surry	NA	Comment incorporated.
3.2.5.1	Material Chemistry						
		NA	RH	NA	No comments	No proprietary data	
3.2.5.3	Mechanical Properties						
		8, 36	MR,AN	E	"it is conservative" instead of "is conservative"		Comment incorporated.
3.2.6	Confirmatory Calculations						
		8, 43	MR	T	Please restrict the comparison/independent evaluation to the NUREG/CR-5729 cooper-fluence model to the plants within the scope of the TR. Suggest adding "; this independent evaluation using the NUREG/CR-5729 copper-fluence model is restricted to the plants within scope of the TR."	NA	
		9, 4	AN	E		The "blue (top) curve appears to be gray in color in the figure	Comment incorporated.

BAW-2192							
Section Number	**Section Title**	**Page/Line Number**	**Reviewer**	**Technical (T) or Editorial (E)**	**Comment**	**Proprietary Marking Comment**	**NRC Resolution**
		9, Figure 1	MR	E	Figure 1 is missing from the figure heading	The J-R lower bound curve is proprietary so this Figure should be marked as proprietary. Reference BAW-2192P, Supplement 1, Revision 0, Figure 5-1	Comment incorporated.

AREVA

Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads

BAW-2192NP
Supplement 1
Revision 0

Topical Report

December 2017

AREVA Inc.

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
B&W	Babcock and Wilcox
B&WOG	Babcock and Wilcox Owners Group
CvUSE	Charpy Upper Shelf Energy
EFPY	Effective Full Power Years
EMA	Equivalent Margins Analysis
INF	Inlet Nozzle Forging
Jd	J deformation
J-R	J-integral Resistance
LAR	License Amendment Request
ONF	Outlet Nozzle Forging
ONS	Oconee Nuclear Station
PTN	Turkey Point Plant
RV	Reactor Vessel
RVWG	Reactor Vessel Working Group
SLR	Subsequent License Renewal
SRP	Standard Review Plan
Sy	Yield Strength
TSs	Technical Specifications
USE	Upper Shelf Energy

ABSTRACT

This Supplement 1 to BAW-2192PA reports an equivalent margins analysis (EMA) considering Levels A and B service loads for high copper Linde 80 weld metals and applicable non-Linde 80 welds using fluence values expected at 80-years (subsequent license renewal--SLR). This supplement to BAW-2192 applies to the following B&W-designed and Westinghouse-designed reactor vessels fabricated by B&W/Rotterdam: Oconee Nuclear Station (ONS) Units 1, 2, and 3 (Oconee Units 1, 2, and 3), Surry Units 1 and 2, and Turkey Point Plant (PTN) Units 3 and 4.

The analytical procedure used in this supplement is in accordance with ASME Section XI, Appendix K, Subarticle K-1200, with selection of design transients based on the guidance in Regulatory Guide 1.161, Section 4.0. EMA results are reported for all reactor vessel weld locations with 80-year fluence projections that exceed 1.0 E+17 n/cm^2 (E> 1.0 MeV). The ASME Section XI, acceptance criteria for Levels A & B Service Loads for all reactor vessel shell welds are satisfied. The acceptance criteria for Levels A & B Service Loads for RV transition welds and RV nozzle welds are also satisfied. Consistent with BAW-2192PA, Revision 00, the B&WOG J-R Model 4B is used for Linde 80 welds and Rotterdam welds. Plants that reference this report must calculate 80-year neutron fluence at reactor vessel weld locations in accordance with the requirements of NUREG-2192 (SLR-SRP), to demonstrate that the fluence estimates provided in Section 3.0 are applicable to their plants.

The EMA utilizes the B&WOG J-integral resistance (J-R) Model 4B reported in BAW-2192PA, Appendix B. Model 4B was developed based on fracture toughness test data obtained through approximately 1990, with specimen fluence that ranges from 0.0 to 8.45E18 n/cm^2. Eighty-year fluence estimates for the participating plants exceeds 8.45E18 n/cm^2 (e.g., maximum 80-year 1/4 T fluence is estimated at 6.5E19 n/cm^2) and use of Model 4B to estimate J-integral resistance values, including the associated model uncertainty, for 80-years is made by extrapolation of the model. To assess the model extrapolation uncertainty, Model 4B is compared to new fracture toughness test data (1990 to 2017) irradiated to fluence ranging from 8.0E18 n/cm^2 to 5.8E19 n/cm^2. The majority of test data fell above the Model 4B mean and all of the test data fell above the Model 4B mean minus 2 standard error band. Therefore, use of Model 4B and associated uncertainty to extrapolate J-integral resistance for 80-year fluence applications was determined to be appropriate. This assessment is reported in Appendix A.

To further substantiate the use of Model 4B, all of the original fracture toughness data used to develop Model 4B was combined with new fracture toughness data, using the same model form, and a new Model 6B was generated. Model 6B was found to be essentially equivalent to Model 4B with respect to model mean and 2 standard errors. The EMA results reported herein using Model 4B were reconciled to Model 6B, with little or no change to the EMA results. Model 6B development and the EMA reconciliation to Model 4B are reported in Appendix A.

1.0 INTRODUCTION

The purpose of Supplement 1 to BAW-2192PA is to report an equivalent margins analysis (EMA) considering Levels A and B service loads for high copper Linde 80 weld metals and applicable non-Linde 80 welds using fluence values expected at 80-years (subsequent license renewal--SLR). This supplement to BAW-2192 applies to the following B&W-designed and Westinghouse-designed reactor vessels fabricated by B&W/Rotterdam: Oconee Nuclear Station (ONS) Units 1, 2, and 3 (Oconee Units 1, 2, and 3), Surry Units 1 and 2, and Turkey Point Plant (PTN) Units 3 and 4, also referred to as Turkey Point Units 3 and 4.

Equivalent margins analyses for the plants within the scope of this report are reported for all reactor vessel weld locations with 80-year fluence projections that exceed 1.0 E+17 n/cm^2 (E> 1.0 MeV) [2]. Plants that reference this report must calculate 80-year neutron fluence at reactor vessel weld locations in accordance with the requirements of NUREG-2192 [3], Standard Review Plan for Review of Subsequent License Renewal Applications, to demonstrate that the fluence estimates provided in Section 3.0 are applicable to their plants. Upper shelf energy evaluations at reactor vessel base metal locations with 80-year fluence projections greater than 1.0 E+17 n/cm^2, if needed, will be addressed separately by each license renewal applicant and are not within the scope of this report.

The EMA utilizes the B&WOG J-integral resistance (J-R) model reported in BAW-2192PA, Appendix B, [1]. The following groups are used for the welds within the scope of this report:

- Reactor Vessel Shell Welds—circumferential and longitudinal welds (if applicable) within the upper and lower shell assemblies for Oconee Units 1, 2, and 3 (also referred to as Oconee reactor vessels), and within the intermediate and lower shell assemblies for Surry Units 1 and 2 (also referred to as Surry reactor vessels) and Turkey Point Units 3 and 4 (also referred to as Turkey Point reactor vessels). There are no geometric discontinuities at these weld locations and all reactor vessel shell welds surround the effective height of the active core. These locations have historically been considered "beltline" or "beltline region" as defined by 10 CFR 50, Appendix G. All reactor vessel shell welds are Linde 80 welds with the exception of Surry Unit 2 weld R-3008 (Figure 3—6), which is a Rotterdam weld.

- Transition Welds and RV Nozzle Welds—welds that are located above and below the reactor vessel shell welds that may experience 80-year fluence greater than 1.0 E+17 n/cm^2 [2] and, must consider the effects of neutron irradiation embrittlement. In addition, the transition welds are located at geometric discontinuities (e.g., lower shell to lower head and upper shell to nozzle belt forging). These locations may or may not have been included as part of the 10 CFR 50 Appendix G [4] "beltline" definition for 60-years for the participating plants. All transition welds and RV nozzle welds (also referred to as RV nozzle-to-shell welds) are Linde 80 welds with the exception of the following: Surry Unit 1 transition weld J726 (Figure 3—5), Surry Unit 2 transition weld L737 (Figure 3—6), and Surry Unit 2 RV outlet nozzle-to-nozzle belt forging welds, which are Rotterdam welds.

The EMA evaluations in this report are for all weld locations expected to receive fluence > 1.0E17 n/cm^2 [2] at 80 years. The use of the terms "beltline" and/or "extended beltline" are not used in this report. It is the responsibility of each license renewal applicant to address the 80-year 10 CFR 50 Appendix G beltline definition for their plant(s) in their license renewal application submittal.

The 60-year EMA summary reports for Oconee Units 1, 2, and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4 are reported in Section 1.1. Section 2.0 provides the current NRC regulatory requirements for the EMA. Section 3.0 provides a description of all reactor vessels within the scope of this report, with illustrations of reactor vessel welds that are evaluated for equivalent margins in Figures 3-1 through 3-8. Section 4.0 provides the material properties that are required for the EMA, and Section 5.0 presents the results of the EMA. Section 6.0 provides the summary and conclusions, Section 7.0 lists all references and Appendix A provides the technical basis for the use of B&WOG J-R Model 4B for the EMA reported herein.

1.1 *Equivalent Margins Analysis—Analysis of Record*

BAW-2192PA, Revision 00 [1] provided the EMA analysis of record for Levels A and B service loads for Oconee Units 1, 2, and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4 for 40 years. For 60 years, Oconee Units 1, 2, and 3 relied on the EMA reported in BAW-2251A [5], while Surry Units 1 and 2 and Turkey Point Units 3 and 4 reported plant-specific evaluations. The summary reports for EMA analyses of record are as follows.

Oconee Units 1, 2, and 3

In February 2014, Duke Energy submitted a license amendment request to the NRC to revise the Oconee Units 1, 2, and 3 Technical Specifications (TSs) by replacing the reactor pressure vessel pressure-temperature (P-T) limits in TS 3.4.3 with new P-T limits applicable to 54 effective full power years [6]. The Duke P-T limit submittal referenced AREVA topical report ANP-3127, Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY [7]. This report provided updated 54 EFPY fluence projections for traditional beltline locations.

The NRC staff noted that the 54 EFPY fluence projections reported in ANP-3127 exceeded the 48 EFPY fluence reported in the Oconee Nuclear Station (ONS) License Renewal Application [8], which referenced AREVA report BAW-2251-A [5], Appendix B, for the ONS 48 EFPY equivalent margins analysis. The NRC performed equivalent margins analysis reconciliation for the Oconee reactor vessel Linde 80 welds, including the lower shell to dutchman weld, which was not addressed in BAW-2251-A, Appendix B. The NRC performed CvUSE drop calculations for the plates (Unit 1) and forgings (Units 2 and 3) to show that the upper shelf energy for these items remained above 50 ft-lbs at 54 EFPY [6]. The NRC did not revise the EMA for the RV nozzles reported in Section 3.1, Page 3-6, of BAW-2251A.

Surry Units 1 and 2

The Surry Units 1 and 2 current licensing basis equivalent margins analysis at 48 EFPY is summarized in Section 3.2.3 of NRC document "SURRY POWER STATION, UNIT NOS. 1 AND 2 -ISSUANCE OF AMENDMENTS REGARDING REACTOR VESSEL HEATUP AND COOLDOWN CURVES FOR 48 EFFECTIVE FULL-POWER YEARS," Adams Accession number ML11110A111 [9]. The NRC SER of the 48 EFPY P-T limits references the Dominion submittal entitled, VIRGINIA ELECTRIC AND POWER COMPANY SURRY POWER STATION UNITS 1 AND 2 UPDATE TO NRC REACTOR VESSEL INTEGRITY DATABASE AND EXEMPTION REQUEST FOR ALTERNATE MATERIAL PROPERTIES BASIS PER 10 CFR 50.60(b) [10]. Specifically, Attachment 4 to Reference [10] includes AREVA document BAW-2494, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of Surry Units 1 and 2 for Extended Life through 48 Effective Full Power Years.

Turkey Point Units 3 and 4

The Turkey Point Units 3 and 4 current licensing basis equivalent margins analysis at 48 EFPY is summarized in Section 2.1.2 of NRC document "TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING EXTENDED POWER UPRATE (TAC NOS. ME4907 AND ME4908)," Adams Accession number ML11293A365 [11]. NRC acceptance of the Turkey Point EMA at 48 EFPY for EPU is based on the following documentation:

- LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE, ATTACHMENT 4, L-2010-113, Attachment 4 ADAMS --ML103560177 [12]
- Supplemental Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Equivalent Margin Analysis (EMA), L-2010-303, ADAMS-ML103610321 [13]. Reference [13] contains AREVA document 77-2312-03 (P), LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF TURKEY POINT UNITS 3 AND 4 FOR EXTENDED LIFE THROUGH 48 EFFECTIVE FULL POWER YEARS
- Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Materials Issues – Round 1, L-2011-029, ADAMS--ML110700068 [14]. Reference [14] provides a response to RAI CVIB-1.2 regarding the code year used to perform the equivalent margins analysis (i.e., 1998 Edition vs 2004 Edition). The NRC SER for the Turkey Point Extended Power Uprate, Section 2.1.2, contains the evaluation of upper shelf energy--ADAMS-ML 11293A365

2.0 REGULATORY REQUIREMENTS

2.1 *Regulatory Requirements*

In accordance with 10 CFR 50 Appendix G [4], IV, A, 1., Reactor Vessel Upper Shelf Energy Requirements are as follows:

a. Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into 10 CFR 50.55a (b)(2) at the time the analysis is submitted.

b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests for use in the analysis specified in section IV.A.1.a.

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in § 50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation or Director, Office of New Reactors, as appropriate.

When the reactor vessels within the scope of this report were fabricated, charpy V-notch testing of the reactor vessel welds were in accordance with the original construction code, which did not specifically require charpy V-notch tests on the upper shelf.

Applicable construction codes are as follows:

- Oconee--ASME Section III, 1965 Edition with Addenda through Summer 1967 (BAW-2251-A)
- Surry--ASME B&PV Code, Section III, 1968 Edition through Winter 1968 Addenda (UFSAR Table 4.1-9)
- Turkey Point--ASME B&PV Code, Section III, 1965 Edition through Summer 1966 Addenda

In accordance with NRC Regulatory Guide 1.161 [15], the NRC has determined that the analytical methods described in ASME Section XI, Appendix K, provide acceptable guidance for evaluating reactor pressure vessels when the Charpy upper-shelf energy falls below the 50 ft-lb limit of Appendix G to 10 CFR Part 50. However, the staff noted that Appendix K does not provide information on the selection of transients and gives very little detail on the selection of material properties. Selection of the limiting design transient (i.e., cooldown at 100 F/h) is consistent with BAW-2192PA [1], Section 5.3. Section 4.1 of this report includes a summary of the B&WOG J-integral resistance model, and Section 4.2 provides mechanical properties of weld metals.

The Linde 80 and Rotterdam weld locations that are included within the scope of this report (i.e., weld locations with 80-year projected fluence > 1.0E+17 n/cm^2) are all assumed to have upper shelf energy values below 50 ft-bs and thus require an equivalent margins analysis.

2.2 *Compliance with 10 CFR 50 Appendix G and Acceptance Criteria*

The analyses reported herein are performed in accordance with the 2007 Edition with 2008 Addenda [16] of Section XI of the ASME Code, Appendix K. The current edition of ASME Section XI listed in 10 CFR 50.55a is the 2013 Edition [17]. With regard to Appendix K, there are no differences between the 2007 Edition with 2008 Addenda and the 2013 Edition of ASME Section XI, and hence these ASME Section XI, Appendix K analyses are equally applicable to the 2013 Edition of the ASME Code.

The material properties used in this analysis are based on ASME Section II, Part D, 2007 Edition with 2008 Addenda. The only change in the material properties listed in the 2013 Edition of ASME Section II for the applicable properties is the coefficient of thermal expansion for stainless steel at 600°F; this value was changed from 9.8E-6 in/in/°F to 9.9E-6 in/in/°F. This does not impact the Levels A and B evaluation reported herein.

2.2.1 Acceptance Criteria Levels A and B

ASME Section XI [17], Subarticle K-2200, provides the following acceptance criteria for Levels A and B Service Conditions:

a) When evaluating adequacy of the upper shelf toughness for the weld material for Levels A and B Service Loadings, an interior semi-elliptical surface flaw with a depth one-quarter of the wall thickness and a length six times the depth shall be postulated, with the flaw's major axis oriented along the weld of concern, and the flaw plane oriented in the radial direction. When evaluating adequacy of the upper shelf toughness for the base material, both interior axial and circumferential flaws with depths one quarter of the wall thickness and lengths six times the depth shall be postulated, and toughness properties for the corresponding orientation shall be used. Smaller flaw sizes may be used when justified. Two criteria shall be satisfied:

1. The applied J-integral evaluated at a pressure 1.15 times the accumulation pressure as defined in the plant specific Overpressure Protection Report, with a structural factor of 1 on thermal loading for the plant specific heatup and cooldown conditions, shall be less than the J-integral of the material at a ductile flaw extension of 0.1 in. (2.5 mm).

2. Flaw extensions at pressures up to 1.25 times the accumulation pressure of K-2200(a)(1) shall be ductile and stable, using a structural factor of 1 on thermal loading for the plant specific heatup and cooldown conditions.

b) The J-integral resistance versus flaw extension curve shall be a conservative representation for the vessel material under evaluation.

3.0 DESCRIPTION OF OCONEE, SURRY AND TURKEY POINT REACTOR VESSELS

The Oconee, Surry and Turkey Point reactor vessels and applicable weld locations are shown in Figures 3-1 through 3-8. All weld locations evaluated for equivalent margins in this report are identified by an asterisk (*) in each Figure. Plant-specific weld copper and nickel content and 80-year fluence projections data needed for the equivalent margins analysis are provided in Table 3—1. The fluence projections are reported for all reactor vessel weld locations that are expected to exceed 1.0E+17 n/cm^2 at 80 years for the participating plants. All plants that reference this report must generate 80-year neutron fluence at reactor vessel locations in accordance with NUREG-2192, SLR-SRP, to demonstrate that the fluence estimates provided in Table 3-1 are applicable to their plants. Note that all fluence values are inside wetted surface with the exception of selected locations for Surry Units 1 and 2 and Turkey Point 3 and 4 marked by an * (clad/base metal interface). Copper and nickel content of the reactor vessel shell welds is consistent with EMA analyses of record reported in Section 1.1; the copper and nickel content for transition welds and RV nozzle-to-nozzle belt forging welds reported in Table 3-1 were obtained from either the EMA analysis of record or a search of Oconee, Surry, and Turkey Point reactor vessel fabrication reports.

Table 3—1
Reactor Vessel Weld Locations--Copper Content and 80-Year Fluence Projections

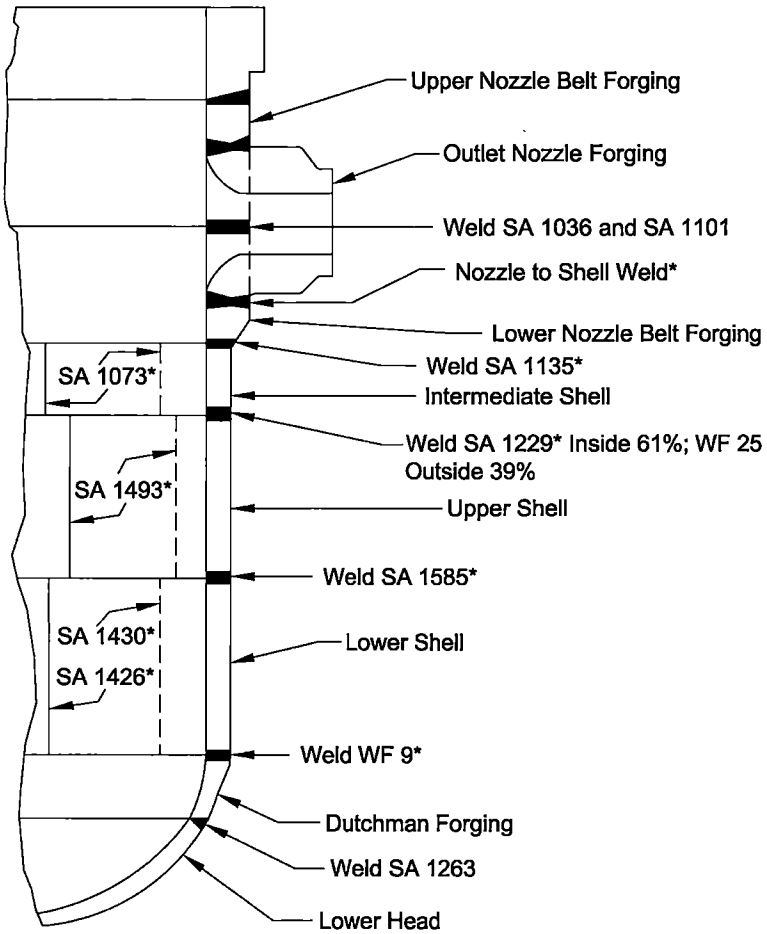
Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
Oconee Unit 1, 80-YearFluence (E > 1.0 MeV)				
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 299L44	0.34	0.68	(IS) 1.50E+18
	Wire Ht. 8T1554B	0.16	0.57	(IS) 1.50E+18
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 299L44	0.34	0.68	(IS) 1.50E+18
	Wire Ht. 8T1554B	0.16	0.57	(IS) 1.50E+18
LNB to Intermediate Shell (IS) Circ. Weld	SA-1135 (Wire Ht. 61782)	0.23	0.52	(IS) 1.90E+18
IS Long. Welds (Both)	SA-1073 (Wire Ht. 1P0962)	0.21	0.64	(IS) 1.58E+19
IS to Upper Shell (US) Circ. Weld (ID 61%)	SA-1229 (Wire Ht. 71249)	0.23	0.59	(IS) 2.02E+19
US Long. Welds (Both)	SA-1493 (Wire Ht. 8T1762)	0.19	0.57	(IS) 2.05E+19
US to Lower Shell (LS) Circ. Weld	SA-1585 (Wire Ht. 72445)	0.22	0.54	(IS) 2.14E+19
LS Long. Weld (1)	SA-1426 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.82E+19
LS Long. Weld (2)	SA-1430 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.82E+19
LS to Dutchman Circ. Weld	WF-9 (Wire Ht. 72445)	0.22	0.54	(IS) 4.88E+17
Oconee Unit 2, 80-Year Fluence (E > 1.0 MeV)				

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 72445	0.22	0.54	(IS) 1.50E+18
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 8T1762	0.19	0.57	(IS) 1.50E+18
	Wire Ht. 72445	0.22	0.54	(IS) 1.50E+18
LNB to Upper Shell (US) Circ. Weld	WF-154 (Wire Ht. 406L44)	0.27	0.59	(IS) 1.99E+19
US to Lower Shell (LS) Circ. Weld	WF-25 (Wire Ht. 299L44)	0.34	0.68	(IS) 2.18E+19
LS to Dutchman Circ. Weld	WF-112 (Wire Ht. 406L44)	0.27	0.59	(IS) 5.20E+17
Oconee Unit 3, 80-Year Fluence (E > 1.0 MeV)				
Lower Nozzle Belt (LNB) to Outlet Nozzle Forging (ONF) Welds	Wire Ht. 72105	0.32	0.58	(IS) 1.50E+18
	Wire Ht. 406L44	0.27	0.59	(IS) 1.50E+18
LNB to Inlet Nozzle Forging (INF) Welds	Wire Ht. 72105	0.32	0.58	(IS) 1.50E+18
	Wire Ht. 72102	0.21	0.58	(IS) 1.50E+18
	Wire Ht. 82102	0.35	1.00	(IS) 1.50E+18
LNB to Upper Shell (US) Circ. Weld	WF-200 (Wire Ht. 821T44)	0.24	0.63	(IS) 1.92E+19
US to Lower Shell (LS) Circ. Weld (ID 75%)	WF-67 (Wire Ht. 72442)	0.26	0.60	(IS) 2.04E+19
LS to Dutchman Circ. Weld	WF-169-1 (Wire Ht. 8T1554)	0.16	0.57	(IS) 4.78E+17
Surry Unit 1, 80-Year Fluence (E > 1.0 MeV)				

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
Nozzle Shell (NS) to Outlet Nozzle Forging Welds	SA-1493(Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
	SA-1494(Wire Ht. 8T1554B)	0.16	0.57	(IS) 1.50E+18
NS to Inlet Nozzle Forging Welds	SA-1526 (Wire Ht. 299L44)	0.34	0.68	(IS) 1.50E+18
	SA-1580 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
NS to Intermediate Shell (IS) Circ. Weld	J726 (Wire Ht. 25017)	0.33	0.10	(*) 7.98E+18
IS Long. Welds (Both)	SA-1494(Wire Ht. 8T1554)	0.16	0.57	(*)1.33E+19
IS to Lower Shell (LS) Circ. Weld (ID 40% 0	SA-1585 (Wire Ht. 72445)	0.22	0.54	(*)6.67E+19
IS to LS Circ. Weld (OD 60%)	SA-1650 (Wire Ht. 72445)	0.22	0.54	NA
LS Long. Weld (1)	SA-1494 (Wire Ht. 8T1554)	0.16	0.57	(*)1.34E+19
LS Long. Weld (2)	SA-1526 (Wire Ht. 299L44)	0.34	0.68	(*)1.34E+19
Surry Unit 2, 80-Year Fluence (E > 1.0 MeV)				
Nozzle Shell (NS) to Outlet Nozzle Forging Welds	Rotterdam	0.35	1.0	(IS) 1.50E+18
NS to Inlet Nozzle Forging Welds	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
	WF-8 (Wire Ht. 8T1762)	0.19	0.57	(IS) 1.50E+18
NS to Intermediate Shell (IS) Circ. Weld	L737 (Wire Ht. 4275)	0.35	0.10	(*) 9.21E+18
IS Long. Weld (1), and (2) (100% and OD 50%)	SA-1585 (Wire Ht. 72445)	0.22	0.54	(*) 1.36E+19

Reactor Vessel Material	Material ID and/or Heat Number	Cu, wt%	Ni, wt%	(IS) Inside Wetted Surface Fluence or (*) clad/base metal n/cm^2 E> 1.0 MeV)
IS Long. Weld (2) (ID 50%)	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(*) 1.36E+19
IS to Lower Shell (LS) Circ. Weld	R3008 (Wire Ht. 0227)	0.187	0.545	(*) 7.67E+19
LS Long. Weld (Both)	WF-4 (Wire Ht. 8T1762)	0.19	0.57	(*) 1.37E+19
Turkey Point Unit 3, 80-Year Fluence (E > 1.0 MeV)				
US Forging to INF Welds	Heat 8T1762	0.19	0.57	(IS) 1.50E+18
	Heat 8T1554B	0.16	0.57	(IS) 1.50E+18
	Heat 71249	0.23	0.59	(IS) 1.50E+18
US Forging to ONF Welds	Heat 8T1762	0.19	0.57	(IS) 1.50E+18
US to IS Circ. Weld	SA-1484 Heat 72442	0.26	0.60	(*) 1.19E+19
IS to LS Circ. Weld	SA-1101 Heat 71249	0.23	0.59	(*) 1.04E+20
LS to Dutchman Circ. Weld	SA-1135 Heat 61782	0.23	0.52	(IS) 1.5E+18
Turkey Point Unit 4, 80-Year Fluence (E > 1.0 MeV)				
US to INF Welds	Heat 8T1762	0.19	0.57	(IS) 1.50E+18
	Heat 8T1554B	0.16	0.57	(IS) 1.50E+18
	Heat 299L44	0.34	0.68	(IS) 1.50E+18
US to ONF Welds	Heat 8T1554B	0.16	0.57	(IS) 1.50E+18
	Heat 299L44	0.34	0.68	(IS) 1.50E+18
US to IS Circ. Weld (ID 67%)	WF-67 Heat 72442	0.26	0.60	(*) 1.21E+19
IS to LS Circ. Weld	SA-1101 Heat 71249	0.23	0.59	(*) 1.03E+20
LS to Dutchman Circ. Weld	SA-1135 Heat 61782	0.23	0.52	(IS) 1.5E+18

Figure 3—1
Reactor Vessel—Oconee Unit 1



* Equivalent Margins Analysis performed for these Linde 80 Welds.

Figure 3—2
Reactor Vessel Shell Longitudinal Welds—Oconee Unit 1

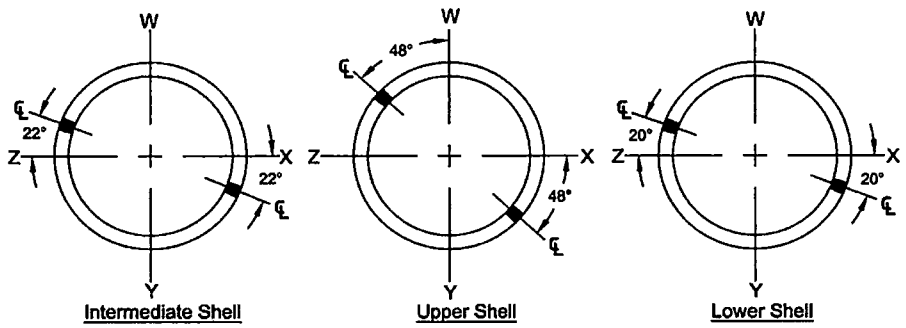
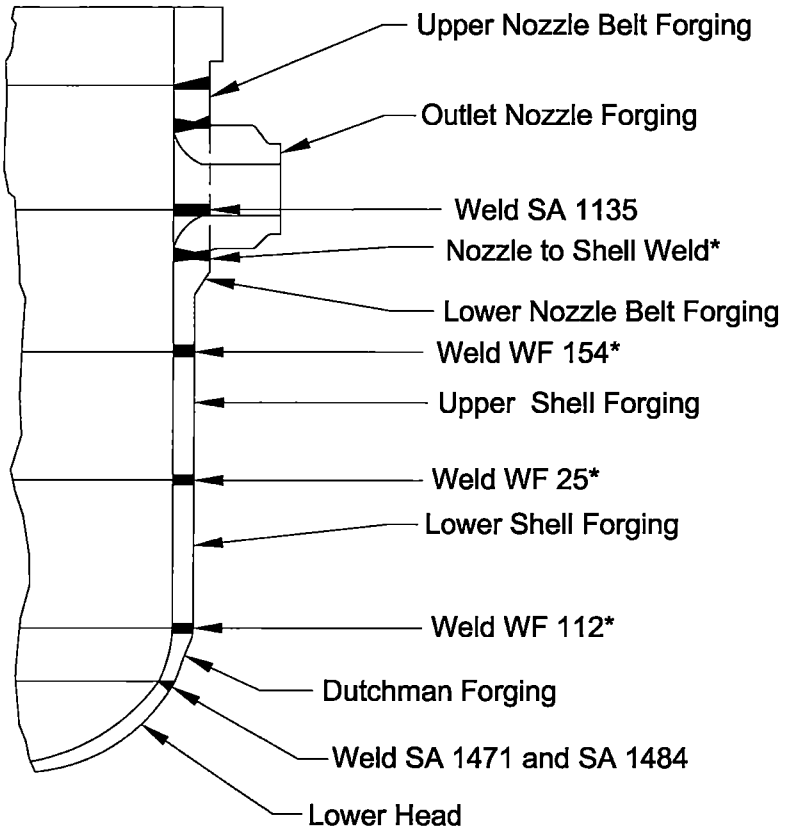
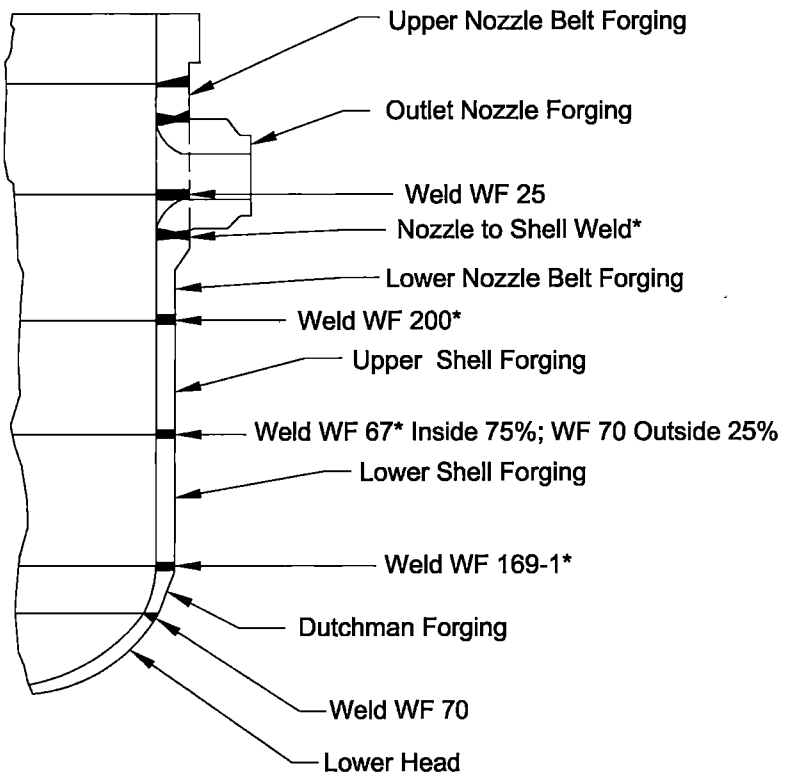


Figure 3—3
Reactor Vessel—Oconee Unit 2



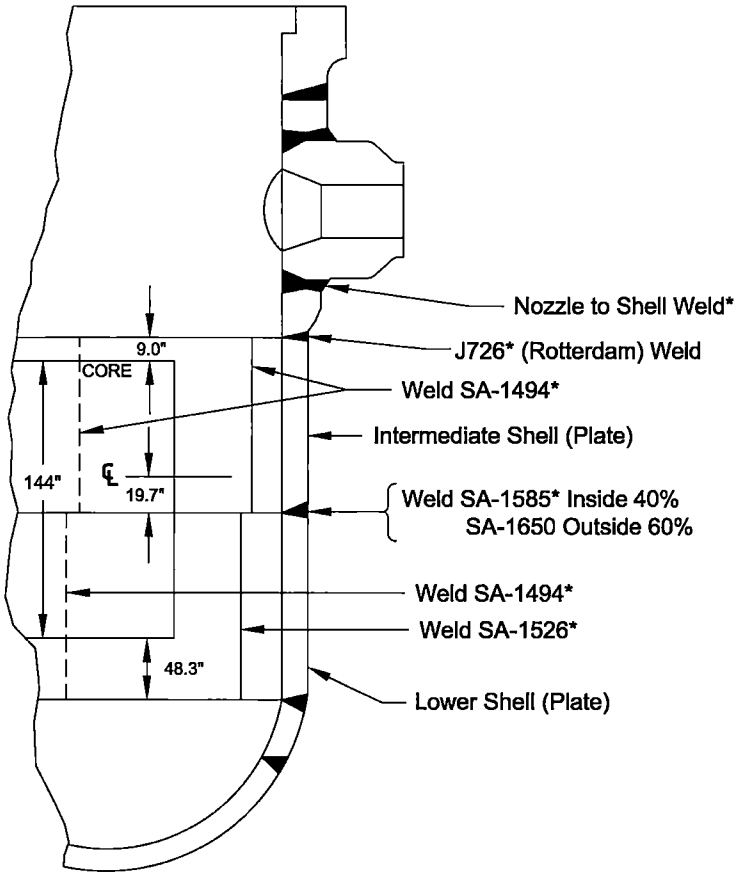
* Equivalent Margins Analysis performed for these Linde 80 Welds.

Figure 3—4
Reactor Vessel—Oconee Unit 3



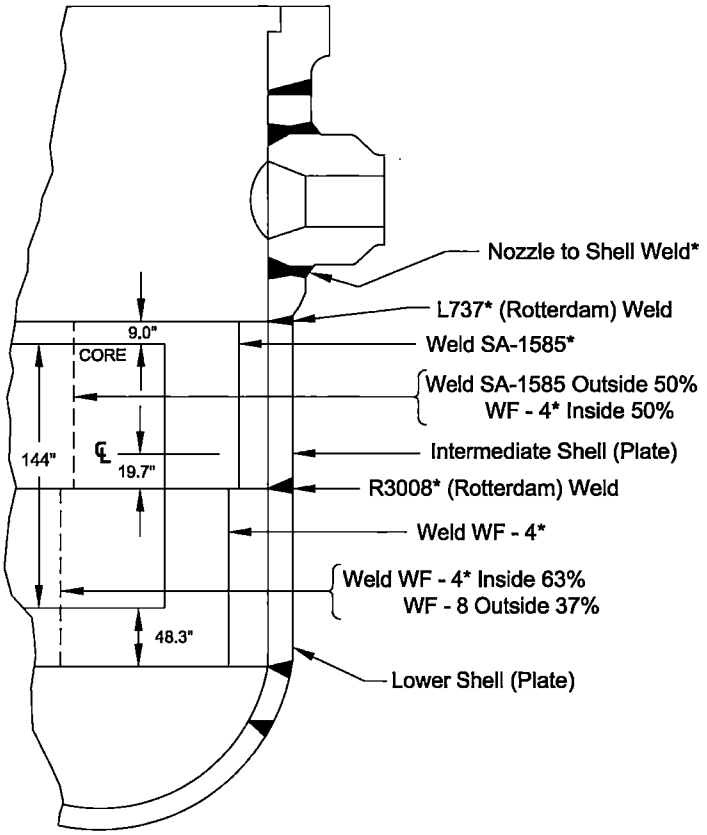
* Equivalent Margins Analysis performed for these Linde 80 Welds.

Figure 3—5
Reactor Vessel—Surry Unit 1



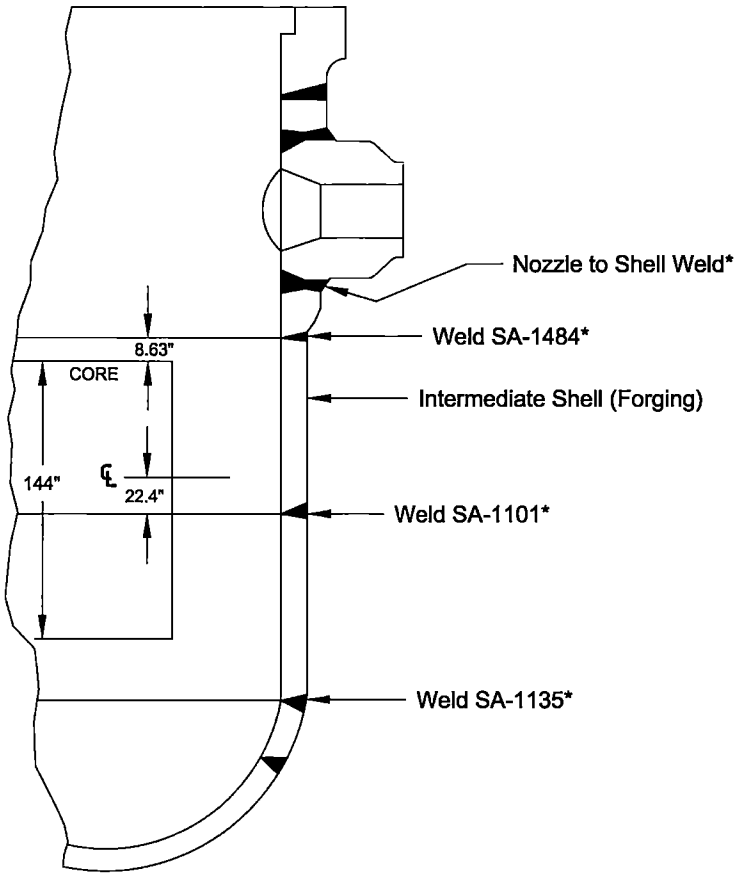
* Equivalent Margins Analysis performed for these Linde 80 and Rotterdam Welds.

Figure 3—6
Reactor Vessel—Surry Unit 2



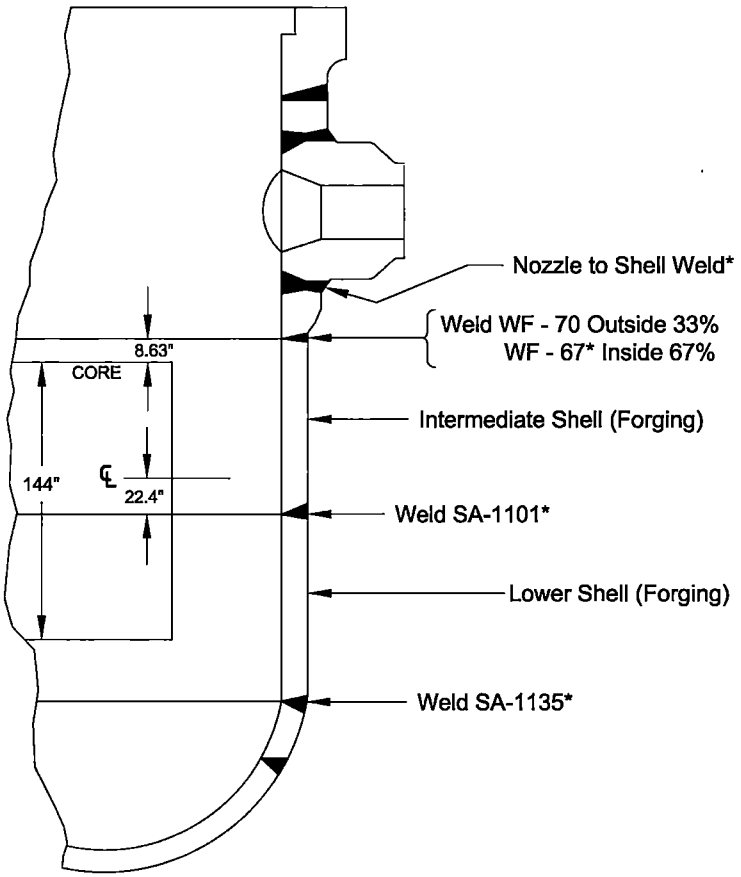
* Equivalent Margins Analysis performed for these Linde 80 and Rotterdam Welds.

Figure 3—7
Reactor Vessel—Turkey Point Unit 3



* Equivalent Margins Analysis performed for these Linde 80 Welds.

Figure 3—8
Reactor Vessel—Turkey Point Unit 4



* Equivalent Margins Analysis performed for these Linde 80 Welds.

4.0 MATERIAL PROPERTIES

4.1 *J-Integral Resistance Model*

The J-integral resistance model for Mn-Mo-Ni/Linde 80 welds in the reactor vessels of the B&WOG RVWG plants were developed using a large J-resistance data base. A detailed description of this model is provided in Appendix B of BAW-2192PA [1], Revision 00. This model was developed using specimens irradiated to 8.45E+18 n/cm^2, and the range of applicability of the model was extended (qualitatively) to approximately 1.90 E+19 n/cm^2 in Appendix B, Figure 3-1, to BAW-2251A [5]. See Appendix A of this report for a discussion of the extension of the range of applicability of the B&WOG J-R model to fluence values expected at 80 years for Oconee, Surry, and Turkey Point. Consistent with BAW-2192PA, Revision 00, this J-R model is used for Linde 80 welds and Rotterdam welds.

The coefficients a, d, and C4 are provided in Table 4-1. As required by ASME Section XI, ASME K-3300, when evaluating the vessel for Levels A, B, and C Service Loadings, the J-integral resistance versus crack-extension curve (J-R curve) shall be a conservative representation of the toughness of the controlling beltline material at upper shelf temperatures in the operating range. As such, the Jd correlation minus 2 standard errors is used for evaluation of Levels A & B service loadings (i.e., equation (1) multiplied by [ ]).

As discussed in Appendix B to BAW-2192PA, the J-R curve was generated from a J-integral database obtained from the same class of material with the same orientation as the applicable reactor vessel materials using correlations for the effects of temperature, chemical composition, and fluence level. Crack extension was by ductile tearing with no cleavage. This complies with the ASME Code, Section XI, K-3300.

Table 4—1
Parameters in Jd Model 4B

[ ]

4.2 *Mechanical Properties of Weld Metals*

The following subsections provide representative properties for the Oconee, Surry and Turkey Point reactor vessels. The temperature dependent mechanical properties are developed from the 2007 Edition with 2008 Addenda of the ASME Code (Section II, Part D) for the reactor base metal and cladding (the ASME Code does not provide separate mechanical properties for base and weld metal). Both ASME Code minimum and representative irradiated yield strengths are also provided. The mechanical properties such as weld metal yield strengths typically used were the irradiated properties but in some cases the ASME Code minimum properties were conservatively considered. The irradiated material properties used herein are consistent with those used for the plants 60-year license renewal low upper shelf toughness analysis submittals (See Section 1.1 above).

4.2.1 Mechanical Properties for the Oconee Reactor Vessels

The Oconee reactor vessels are fabricated using either A-508 Grade 2 Class 1 (3/4Ni-1/2Mo-1/3Cr-V) forging or A-533 Grade B Class 1 (Mn-1/2Mo-1/2Ni) plate Low Alloy Steel (LAS) materials. The stainless steel cladding material is considered to be fabricated from 18Cr-8Ni. Table 4-2 provides the Young's modulus (E), the mean coefficient of thermal expansion (α), and the yield strength (Sy) for the RV shell regions.

Table 4—2 Mechanical Properties of Oconee RV Shell Materials

	RV Base Metal			Weld Metal
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	Sy (ksi)
70	27800	6.40E-06	50.0	[ ]
200	27100	6.70E-06	47.0	[ ]
300	26700	6.90E-06	45.5	[ ]
400	26200	7.10E-06	44.2	[ ]
500	25700	7.30E-06	43.2	[ ]
600	25100	7.40E-06	42.1	[ ]

For the RV shell and transition regions, the downcomer or steady state cold leg operating condition temperature is [ ] (Table 5-1). For the nozzle-to-shell welds the normal operating steady state condition temperature is [ ] .

For the Linde 80 weld materials in the RV shell, transition and the nozzle welds of the Oconee reactor vessels, the measured yield strength value of weld metal SA-1585 is used. The yield strength value for this material is [ ] . This weld metal was used in the Oconee Unit 1 RV shell and represents the lowest yield strength value among the Linde 80 welds tested for three Oconee reactor vessels.

4.2.2 Mechanical Properties for the Surry Reactor Vessels

The Surry reactor vessels are fabricated using A-533 Grade B Class 1 (Mn-1/2Mo-1/2Ni) Low Alloy Steel (LAS) and stainless steel (18Cr-8Ni) cladding materials. Table 4-3 provides the Young's modulus (E), the mean coefficient of thermal expansion (α), and the yield strength (Sy) for the RV shell.

For the Surry reactor vessel shell welds, transition welds, and RV nozzle welds, the normal operating steady state condition cold leg temperature value is [ ] (Table 5-1). The yield strength value for SA-1526 at [ ] corresponding to a value of [ ] was used in the analysis.

Table 4—3 Mechanical Properties of Surry RV Shell Materials

	RV Base Metal			Weld Metal
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	SA-1526 (ksi)
70	29000	7.00E-06	50.0	[ ]
200	28500	7.30E-06	47.0	[ ]
300	28000	7.40E-06	45.5	[ ]
400	27600	7.60E-06	44.2	[ ]
500	27000	7.70E-06	43.2	[ ]
600	26300	7.80E-06	42.1	[ ]

4.2.3 Mechanical Properties for the Turkey Point Reactor Vessels

The Turkey Point reactor vessels are fabricated using A-508 Grade 2 Class 1 (3/4Ni-1/2Mo-1/3Cr-V) Low Alloy Steel (LAS) and stainless steel (18Cr-8Ni) cladding materials. Table 4-4 provides the Young's modulus (E), the mean coefficient of thermal expansion (α), and the yield strength (Sy) for the RV shell regions.

Table 4—4 Mechanical Properties of Turkey Point RV Shell Materials

	RV Base Metal			Weld Metals	
Temp. (°F)	E (ksi)	α (in/in/°F)	Sy (ksi)	SA-1101 (ksi)	SA-1135 (ksi)
70	27800	6.40E-06	50.0	[ ]	[ ]
200	27100	6.70E-06	47.0	[ ]	[ ]
300	26700	6.90E-06	45.5	[ ]	[ ]
400	26200	7.10E-06	44.2	[ ]	[ ]
500	25700	7.30E-06	43.2	[ ]	[ ]
600	25100	7.40E-06	42.1	[ ]	[ ]

For the Turkey Point reactor vessel shell and RV nozzle regions, the normal operating steady state condition cold leg temperature value is [ ] (Table 5-1). Both the Turkey Point Units have SA-1101 and SA-1135 Linde 80 welds in the RV shell regions. The yield strength values for these weld metals at [ ], respectively.

5.0 FRACTURE MECHANICS ANALYSIS

5.1 *Methodology*

In accordance with ASME Section XI, Appendix K [16], Subarticle K-1200, the following analytical procedure was used for Levels A & B Service Loads.

a. The postulated flaws in the reactor vessel shell welds, the transition welds as well as RV nozzle-to-shell welds were postulated in accordance with the acceptance criteria of Subarticle K-2200.

b. Loading conditions at the locations of the postulated flaws were determined for Levels A and B Service Loadings. For Levels A and B Service loadings the equations to calculate the stress intensity factor (SIF) due to pressure and thermal gradients for a given pressure and cooldown rate are given in Article K-4210. Consistent with Section 5 of BAW-2192PA [1], the accumulation pressure is taken as ten percent above the design pressure and the maximum cooldown rate is 100°F/hr. In the area of the nozzle-to-shell weld, applied loadings consist of pressure, thermal, and attached piping reactions.

c. Material properties, including E, α, σ_y, and the J-integral resistance curve (J-R curve), were determined at the locations of the postulated flaws. Young's modulus, mean coefficient of thermal expansion and yield strength are addressed in Section 4.2. The J-R curve is discussed in Section 4.1.

d. The postulated flaws were evaluated in accordance with the acceptance criteria of Article K-2000. Requirements for evaluating the applied J-integral are provided in Subarticle K-3200, and for determining flaw stability in Subarticle K-3400. Subarticle K-3500(a) invokes the procedure provided in Subarticle K-4200 (K-4220) for evaluating the applied J-integral for a specified amount of ductile flaw extension. Three permissible evaluation methods to address flaw stability are described in Subarticle K-3500(b). The evaluation method selected herein is the J-R curve crack driving force diagram procedure described in Subarticle K-4310.

5.2 *Procedure for Evaluating Levels A and B Service Loadings*

For RV shell regions remote from structural discontinuities, the applied *J*-integral is calculated in accordance with Appendix K, Subsubarticle K-4210, using an effective flaw depth to account for small scale yielding at the crack tip, and evaluated per K-4220 for upper-shelf toughness and per K-4310 for flaw stability, as outlined below.

(1) For an axial flaw of depth *a*, the stress intensity factor due to internal pressure is calculated with a structural factor (*SF*) on pressure using the following:

$$K_{Ip} = (SF)p\left(1+\frac{R_i}{t}\right)(\pi a)^{0.5}F_1$$

where

$$F_1 = 0.982 + 1.006\left(\frac{a}{t}\right)^2, \qquad 0.20 \le \left(\frac{a}{t}\right) \le 0.50$$

(2) For a circumferential flaw of depth *a*, the stress intensity factor due to internal pressure is calculated with a structural factor (*SF*) on pressure using the following:

$$K_{Ip} = (SF)p\left(1+\frac{R_i}{2t}\right)(\pi a)^{0.5}F_2$$

where

$$F_2 = 0.885 + 0.233\left(\frac{a}{t}\right) + 0.345\left(\frac{a}{t}\right)^2, \qquad 0.20 \le \left(\frac{a}{t}\right) \le 0.50$$

(3) For an axial or circumferential flaw of depth *a*, the stress intensity factor due to radial thermal gradients are calculated using the following:

$$K_{It} = C_m(CR)t^{2.5}F_3, \qquad 0 \le (CR) \le 100^\circ\text{F/hr}$$

where for SA-508, Class 2 or SA-533, Grade B, Class1 steels the material coefficient C_m is defined as:

$$C_m = \frac{E\alpha}{(1-\nu)d} = 0.0051,$$

CR = cooldown rate (°F/hr), and

$$F_3 = 0.1181 + 0.5353\left(\frac{a}{t}\right) - 1.273\left(\frac{a}{t}\right)^2 + 0.6046\left(\frac{a}{t}\right)^3, \quad 0.20 \le \left(\frac{a}{t}\right) \le 0.50$$

(4) The effective flaw depth for small scale yielding, a_e, is calculated using the following:

$$a_e = a + \left(\frac{1}{6\pi}\right)\left[\frac{K_{Ip} + K_{It}}{\sigma_y}\right]^2$$

(5) For an axial flaw of depth a_e, the stress intensity factor due to internal pressure is:

$$K'_{Ip} = (SF)p\left(1+\frac{R_i}{t}\right)(\pi a_e)^{0.5} F'_1$$

where

$$F'_1 = 0.982 + 1.006\left(\frac{a_e}{t}\right)^2, \qquad 0.20 \le \left(\frac{a_e}{t}\right) \le 0.50$$

(6) For a circumferential flaw of depth a_e, the stress intensity factor due to internal pressure is:

$$K'_{Ip} = (SF)p\left(1+\frac{R_i}{2t}\right)(\pi a_e)^{0.5} F'_2$$

where

$$F'_2 = 0.885 + 0.233\left(\frac{a_e}{t}\right) + 0.345\left(\frac{a_e}{t}\right)^2, \qquad 0.20 \le \left(\frac{a_e}{t}\right) \le 0.50$$

(7) For an axial or circumferential flaw of depth a_e, the stress intensity factor due to radial thermal gradients is:

$$K'_{It} = C_m(CR)t^{2.5} F'_3, \qquad 0 \le (CR) \le 100^{\circ}\text{F/hr}$$

where

$$F'_3 = 0.1181 + 0.5353\left(\frac{a_e}{t}\right) - 1.273\left(\frac{a_e}{t}\right)^2 + 0.6046\left(\frac{a_e}{t}\right)^3, \quad 0.20 \le \left(\frac{a_e}{t}\right) \le 0.50$$

(8) The *J*-integral due to applied loads for small scale yielding is calculated using the following:

$$J_1 = 1000 \frac{\left(K'_{Ip} + K'_{It}\right)^2}{E'}$$

where

$$E' = \frac{E}{1-\nu^2}$$

(9) Evaluation of upper-shelf toughness at a flaw extension of 0.10 in. is performed for a flaw depth,

$$a = 0.25t + 0.10\text{in.},$$

using

$$SF = 1.15$$

$$p = P_a$$

where P_a is the accumulation pressure for Levels A and B Service Loadings, such that

$$J_1 < J_{0.1}$$

where

J_1 = the applied J-integral for a safety factor of 1.15 on pressure, and a safety factor of 1.0 on thermal loading

$J_{0.1}$ = the lower bound J-integral resistance at a ductile flaw extension of 0.10 in.

(10) Evaluation of flaw stability is performed through use of a crack driving force diagram procedure, by comparing the slopes of the applied J-integral

curve and the lower bound *J-R* curve. The applied *J*-integral is calculated for a series of flaw depths corresponding to increasing amounts of ductile flaw extension. The applied pressure is the accumulation pressure for Levels A and B Service Loadings, P_a, and the safety factor (*SF*) on pressure is 1.25. Flaw stability at a given applied load is verified when the slope of the applied *J*-integral curve is less than the slope of the *J-R* curve at the point on the *J-R* curve where the two curves intersect.

For the Oconee reactor vessels, the effect of structural discontinuity at the transition welds (i.e., "upper weld" at Unit 1 connecting upper end of the intermediate shell to the lower nozzle belt forging & "lower weld" connecting the lower end of the lower shell to the Dutchman forging) is accounted for by applying a scaling factor of [ ] to the applied J-integral calculated by the above procedure for circumferential welds. This approach is based on a previous finite element analysis of an applicable axisymmetric B&W-designed reactor vessel shell model that included a detailed description of the transition regions. The stresses due to pressure and thermal loads at the nozzle-to-shell welds are obtained from a previous axisymmetric analysis of the outlet nozzle of a B&W-designed plant that is also deemed applicable to the Oconee reactor vessels.

For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis. Path line stresses were used to determine applied J-integrals that included a plastic zone correction to account for small scale yielding. Based on the results of analysis performed for Oconee it was deemed that the effects of structural discontinuities at the lower transition welds need not be explicitly addressed.

5.3 *Evaluation for Flaw Extension*

The applied J-integrals for the RV shell welds, the RV transition welds, and the RV nozzle welds are calculated as discussed in Section 5.2.

5.3.1 Reactor Vessel Shell Welds

The basic reactor vessel shell geometry and design pressure along with operating condition temperature information for each of the Oconee, Surry and Turkey Point reactor vessels (also referred to as three groups of reactor vessels) is provided in Table 5-1. Initial flaw depths equal to ¼ of the vessel wall thickness are analyzed for Levels A and B service loadings following the procedure outlined in Section 5.2 and evaluated for acceptance based on values for the J-integral resistance of the material from the Linde 80 J-R model discussed in Section 4.1. For each of the three groups of reactor vessels, calculations are initially carried out to identify the controlling weld such that subsequent detailed low upper shelf toughness flaw evaluations can be performed using the controlling weld.

The results of the plant specific flaw evaluations for each of the RV shell welds of the three groups of reactor vessels are presented in Table 5-2. From the results of the evaluation in Table 5-2, the controlling RV shell welds for each of the three groups of reactor vessels can be observed. The controlling welds are determined by noting the minimum ratio of the material J-resistance ($J_{0.1}$) to the applied J-integral (J_1) (also referred to as "margin") for each of the three groups of reactor vessels.

For the Oconee reactor vessels, the controlling RV shell weld is the SA-1073 longitudinal weld of Oconee Unit 1. This weld is located in the intermediate shell. The minimum ratio of material J-resistance to applied J-integral ($J_{0.1}/J_1$) or margin is [ ], which is higher than the minimum acceptable value of 1.0.

For the Surry reactor vessels, the controlling RV shell weld is the SA-1526 longitudinal weld of Surry Unit 1. This weld is located in the lower shell. The minimum margin ($J_{0.1}/J_1$) is [ ], which is higher than the minimum acceptable value of 1.0.

For the Turkey Point reactor vessels, the controlling RV shell weld is the circumferential weld SA-1101 which is located between the intermediate and lower shell courses of both Units 3 and 4 reactor vessels. The minimum ratio of material J-resistance to applied J-integral ($J_{0.1}/J_1$) is [ ], which is significantly higher than the minimum acceptable value of 1.0.

5.3.2 Reactor Vessel Transition Welds and RV Nozzle Welds

The reactor vessel nozzle welds are located in the substantially thicker cylindrical shell section (reinforced to account for the inlet/outlet RV nozzle openings and typically referred to as the nozzle belt), located above the reactor vessel shell welds. The reactor vessel nozzle belt dimensions are reported in Table 5-3.

For the Oconee reactor vessels there is only one upper transition weld: SA-1135 for Oconee Unit 1 (see Figures 3-1 through 3-4). This circumferential weld is addressed in Table 5-2. The calculated ratio of $J_{0.1}/J_1$ is [ ]. The flaw evaluations for the Oconee lower transition welds are summarized in Table 5-4. The minimum ratio of material J-resistance to applied J-integral ($J_{0.1}/J_1$) is [ ], which is significantly higher than the minimum acceptable value of 1.0. The flaw evaluation of the RV nozzle-to-shell weld are summarized in Table 5-5. The applied stress intensity factor is conservatively calculated for this location in terms of tension and bending loads using a flat plate solution by Newman and Raju. The calculated $J_{0.1}/J_1$ ratios are [ ] and [ ], respectively for flaws oriented in the axial and circumferential direction with respect to the RV, which are higher than the minimum acceptable value of 1.0.

For the Surry reactor vessels upper RV transition and RV nozzle weld regions, the J-applied results with the safety factor of 1.15 on applied pressure are compared against the lower bound J-integral resistance at a ductile flaw extension of 0.1 inches ($J_{0.1}$) in Table 5-6. The most limiting ratio or margin ($J_{0.1}/J_1$) is [ ] due to postulating an axial flaw in the longitudinal weld SA-1585 located near the base of the taper transition section. Weld SA-1585 has the highest copper content of any longitudinal weld in the intermediate shells at Surry. The fluence value corresponding to the base of the taper transition section was utilized. The lower transition welds are located approximately four feet below the bottom of the core and are predicted to receive less than the threshold fluence value of 1.0E17 n/cm^2 at 80-years As such, these lower transition welds were not evaluated in this report.

For the Turkey Point reactor vessels upper RV transition and RV nozzle weld regions, the J-applied results with the safety factor of 1.15 on applied pressure is compared against the lower bound J-integral resistance at a ductile flaw extension of 0.1 inches ($J_{0.1}$) in Table 5-7. The limiting item in this case is the inlet nozzle with a margin of [ ]. For the lower transition weld SA-1135, it is noted that this circumferential weld is located at the thickness transition between the cylindrical shell and the thinner lower head. This location was additionally evaluated using the dimensions applicable to the lower head [ ] since this will result in higher pressure stresses but lower thermal stresses when compared against the thicker RV cylindrical shell. For evaluation of the SA-1135 weld, the margin reduces from [ ] (considering the thicker RV shell as shown in Table 5-2) to [ ] (as given in Table 5-8) when considering the thinner lower head. Since the reduced margin is still significantly larger than 1.0, this simplified analytical approach is deemed to be an acceptable means of addressing the structural discontinuity at the lower transition weld.

5.4 *Evaluation for Flaw Stability*

The flaw stability analysis is performed by calculating the applied J-integrals for various amounts of flaw extension with a safety factor (on pressure) of 1.25. The resulting applied J-integral curve can then be compared against the lower bound J-R curve for the weld metal. It is noted that applied J-integrals are also calculated with a safety factor on pressure of 1.15 for illustration of the $J_{0.1}/J_1$ margin with respect to the lower bound J-R curve at a flaw extension of 0.1 inch.

5.4.1 Reactor Vessel Shell Welds

For the Oconee reactor vessels, the applied J-integral values of the controlling weld (SA-1073) is calculated for various amounts of flaw extension with safety factors (on pressure) of 1.15 and 1.25 in Table 5-9. The corresponding mean and lower bound *J-R* curve values are calculated and shown in Table 5-10. The resulting J-applied curves are compared against the lower bound *J-R* curve for the SA-1073 material in Figure 5-1. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-2 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-1 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

The applied J-integral values for the controlling weld of the Surry reactor vessels (SA-1526) is similarly calculated and shown in Table 5-11 with the corresponding mean and lower bound *J-R* curve values shown in Table 5-12. The resulting J-applied curves are then compared against the lower bound J-R curve for this material in Figure 5-2. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-2 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-2 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

Similarly, for the controlling SA-1101 weld material of Turkey Point Units 3 and 4, the applied J-integral values are calculated and shown in Table 5-13 with the corresponding mean and lower bound *J-R* curve values shown in Table 5-14. The resulting J-applied curves are then compared against the lower bound J-R curve for this material in Figure 5-3. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-2 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-3 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

5.4.2 Reactor Vessel Transition Welds and RV Nozzle Welds

As discussed in Section 5.3.2, the controlling weld for the Oconee RV transition welds and the RV nozzle welds is the RV nozzle-to-shell weld. The results of the flaw evaluation of this weld is provided in Table 5-5 with a margin of [ ] and [ ], respectively for stresses in the circumferential (applicable for axial flaw) and longitudinal directions (applicable for circumferential flaw). The margins and flaw stability considering both flaw orientations are shown in Figure 5-4. Both the mean and the lower bounding *J-R* curves are illustrated in Figure 5-4. An evaluation line at a flaw extension of 0.1 inch is included to confirm the results of Table 5-5 by showing the margin between the applied J-integral with the safety factor of 1.15 and the lower bound J-integral resistance of the material. The requirement for ductile and stable crack growth is demonstrated by Figure 5-4 since the slope of the applied J-integral curve for a safety factor of 1.25 is less than slope of the lower bound *J-R* curve at the point where the two curves intersect.

For the Surry reactor vessels, the controlling weld and flaw orientation is the axial flaw in the longitudinal weld SA-1585 of Surry Unit 2 as discussed previously in Section 5.3.2. The applied J-integral for this axial flaw with a safety factor of 1.25 on pressure at various flaw extensions is plotted with the lower bound J-resistance curve (mean *J-R* curve provided for information only) in Figure 5-5. The slope of the applied J-integral is less than the slope of the lower bound J-resistance curve at the point of intersection, which demonstrates that the flaw is stable as required by ASME Section XI, Appendix K.

As discussed in Section 5.3.2 and shown in Table 5-7, the limiting location for the Turkey Point reactor vessels is the inlet nozzle with a margin of [ ]. The applied J-integral for the inlet nozzle with a safety factor of 1.25 on pressure at various flaw extensions is plotted with the lower bound J-resistance curve in Figure 5-6. The slope of the applied J-integral is less than the slope of the lower bound J-resistance curve at the point of intersection, which demonstrates that the flaw is stable as required by ASME Section XI, Appendix K.

Table 5—1
Reactor Vessel Shell Dimensions and Operating Conditions

Table 5—2 Plant Specific Flaw Evaluation Summary for RV Shell Regions

Table 5—3 Reactor Vessel Nozzle Belt Dimensions

Table 5—4 Flaw Evaluation Summary of Lower Transition Welds of Oconee Units

Table 5—5 Flaw Evaluation Summary of Oconee RV Nozzle-to-Shell Weld

Table 5—6 Flaw Evaluation Summary of Surry Upper Transition and RV Nozzle-to-Shell Welds

Table 5—7 Flaw Evaluation Summary of Turkey Point Upper Transition and RV Nozzle-to-Shell Welds

Table 5—8 Flaw Evaluation Summary of Turkey Point Lower Transition Welds

Table 5—9 Applied J-Integral versus Flaw Extensions of Oconee Controlling RV Shell Weld (SA-1073)

Table 5—10 Mean & Lower Bound J-R Curve Values for Oconee Controlling RV Shell Weld (SA-1073)

Table 5—11 Applied J-Integral versus Flaw Extensions of Surry Controlling RV Shell Weld (SA-1526)

Table 5—12 Mean & Lower Bound J-R Curve Values for Surry Controlling RV Shell Weld (SA-1526)

Table 5—13 Applied J-Integral versus Flaw Extensions of Turkey Point Controlling RV Shell Weld (SA-1101)

Table 5—14 Mean & Lower Bound J-R Curve Values for Turkey Point Controlling RV Shell Weld (SA-1101)

Figure 5—1
J-Integral versus Flaw Extension for ONS -1, 2, 3 Controlling Reactor Vessel Shell Weld (SA-1073)

Figure 5—2
J-Integral versus Flaw Extension for Surry 1 and 2 Controlling Reactor Vessel Shell Weld (SA-1526)

Figure 5—3
J-Integral Flaw Extension for Turkey Point Controlling Reactor Vessel Shell weld (SA-1101)

Figure 5—4 J-Integral versus Flaw Extension for Oconee RV Nozzle-to-Shell Weld

Figure 5—5 J-Integral versus Flaw Extension for Surry Axial Weld (SA-1585) Near Transition

Figure 5—6 J-Integral versus Flaw Extension for Turkey Point Inlet Nozzle

6.0 SUMMARY AND CONCLUSIONS

6.1 *Reactor Vessel Shell Welds*

The ASME Section XI, acceptance criteria for Levels A & B Service Loads for all reactor vessel shell welds are satisfied. The results of the limiting welds for Oconee Units 1, 2, and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4 are reported below.

Oconee Units 1, 2, and 3

- The limiting RV shell weld is Oconee Unit 1 axial weld SA-1073. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-1). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect (Figure 5-1).

Surry Units 1 and 2

- The limiting RV shell weld is Surry Unit 1 axial weld SA-1526. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-2). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied *J*-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect (Figure 5-2).

Turkey Point Units 3 and 4

- The limiting RV shell weld is TP 3 and 4 circumferential weld SA-1101. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-3). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect (Figure 5-3).

6.2 *Reactor Vessel Transition Welds and RV Nozzle Welds*

The acceptance criteria for Levels A & B Service Loads for RV transition welds and RV nozzle welds are satisfied. The results of the limiting weld considering transition welds and RV nozzle welds (inlet and outlet) for Oconee, Surry, and Turkey Point are reported below.

Oconee Units 1, 2, and 3

- The limiting weld for the Oconee Units is the RV outlet nozzle-to-shell weld. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-4). The ratio J0.1/J1 = [ ] and 1.61, respectively, for axial and circumferential flaws.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect (Figure 5-4).

Surry Units 1 and 2

- The limiting weld for the Surry Units 1 and 2 is the longitudinal weld SA-1585 near the base of the transition section. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-5). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0. This is due to postulating an axial flaw in the longitudinal weld of Surry Unit 2 while using the highest copper content of any intersecting longitudinal weld in the intermediate shell.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect (Figure 5-5).

Turkey Point Units 3 and 4

- The limiting weld for the Turkey Point Units 3 and 4 RV transition welds (upper and lower) and the RV nozzle welds is the RV inlet nozzle-to-shell weld. With factors of safety of 1.15 on pressure and 1.0 on thermal loading, the applied J-integral (J1) is less than the J-integral of the material at a ductile flaw extension of 0.10 in. (J0.1)—(Figure 5-6). The ratio J0.1/J1 = [ ] is greater than the required value of 1.0.
- With a factor of safety of 1.25 on pressure and 1.0 on thermal loading, flaw extensions are ductile and stable since the slope of the applied J-integral curve is less than the slope of the lower bound J-R curve at the point where the two curves intersect (Figure 5-6).

7.0 REFERENCES

1. BAW-2192PA, Revision 00, "Low Upper Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Levels A and B Conditions," April 1994, ADAMS Accession (Legacy) 9406240261 (P), 9312220294 (NP)
2. NRC Regulatory Issue Summary 2014-11, Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components
3. NUREG-2192, Volume 2, Standard Review Plan for Review of Subsequent License Renewal Applications for Nuclear Power Plants, ADAMS Accession Number ML17188A158
4. Code of Federal Regulations, Title 10, Part 50 – Domestic Licensing of Production and Utilization Facilities, Appendix G – Fracture Toughness Requirements, Federal Register Vol. 60. No. 243, December 19, 1995.
5. AREVA Document BAW-2251A, "Demonstration of the Management of Aging Effects for the Reactor Vessel, The B&W Owners Group Generic License Renewal Program," August 1999, ADAMS Accession Number 9909300150
6. OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING REVISE PRESSURE-TEMPERATURE LIMITS (TAC NOS. MF0763, MF0764, AND MF0765)-ADAMS ACCESSION Number ML14041A093
7. AREVA Document ANP-3127, Revision 01, "Oconee Nuclear Station Units 1, 2 & 3 Pressure-Temperature Limits at 54 EFPY", ADAMS Accession Number ML13058A060, and Revision 02, ADAMS Accession Number ML13305A121

8. Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3, U. S. Nuclear Regulatory Commission, ACN: 9807200136, Fiche: A4344:001-A4347:255, July 6, 1998

9. SURRY POWER STATION, UNIT NOS. 1 AND 2 -ISSUANCE OF AMENDMENTS REGARDING REACTOR VESSEL HEATUP AND COOLDOWN CURVES FOR 48 EFFECTIVE FULL-POWER YEARS," Adams Accession number ML11110A111

10. VIRGINIA ELECTRIC AND POWER COMPANY SURRY POWER STATION UNITS 1 AND 2 UPDATE TO NRC REACTOR VESSEL INTEGRITY DATABASE AND EXEMPTION REQUEST FOR ALTERNATE MATERIAL PROPERTIES BASIS PER 10 CFR 50.60(b), Adams accession number ML061650080

11. TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING EXTENDED POWER UPRATE (TAC NOS. ME4907 AND ME4908)," Adams Accession number ML11293A365

12. LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE, ATTACHMENT 4, L-2010-113, Attachment 4 ADAMS --ML103560177

13. Supplemental Response to NRC Request for Additional Information (RAI) Regarding Extended Power Uprate (EPU) License Amendment Request (LAR) No. 205 and Equivalent Margin Analysis (EMA), L-2010-303, ADAMS-ML103610321

14. Response to NRC Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205 and Reactor Materials Issues – Round 1, L-2011-029, ADAMS--ML110700068

15. NRC Regulatory Guide 1.161, Evaluation of Reactor Pressure Vessels With Charpy Upper Shelf Energy Less Than 50 ft-lb

16. 2007 Edition (with 2008 Addenda) ASME & Boiler Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear power Plant Components, Appendix K

17. 2013 Edition ASME & Boiler Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear power Plant Components, Appendix K

18. BAW-1975, Applicability of the HSST Program Second and Third Irradiation Series Data to the Integrity of Nuclear Reactor Vessels, A.L. Lowe, June 1987

19. NUREG-1766, Safety Evaluation Report Related to the License Renewal of North Anna Power Station, Units 1 and 2 and Surry Power Station, Units 1 and 2, ADAM Accession Number ML030160825

20. NUREG/CR-5729, Multivariable Modeling of Pressure Vessel and Piping J-R Data, May 1991, E. D. Eason, J. E. Wright, E. E. Nelson

21. BAW-1543, Revision 4, Supplement 6-A, Supplement to the Master Integrated Reactor Vessel Surveillance Program, June 2007

22. BAW-2178P, Supplement 1, Revision 00, December 2017

8.0 CERTIFICATION

This report is an accurate description of the low upper-shelf toughness fracture analysis of ONS Units 1, 2 and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4vessels.

Ashok Nana/Mark Rinckel
Component Analysis and Fracture
Mechanics Unit/Nuclear Analysis Unit

This report has been reviewed and is an accurate description of the low upper-shelf toughness fracture analysis of reactor vessels of ONS Units 1, 2 and 3, Surry Units 1 and 2, and Turkey Point Units 3 and 4.

D.E. Killian
Component Analysis and Fracture
Mechanics Unit

Verification of independent review.

David Cofflin
Component Analysis and Fracture
Mechanics Unit

This report is approved for release.

Danielle ~~Paige~~ Blair → Page MAL 12/15/17
NSSS Project Management

APPENDIX A
B&WOG J-R MODEL-DATA ANALYSIS AND EMPIRICAL MODEL DEVELOPMENT

A.1 Background

[ ]

Table A—1 Model 4B, Range of Test Data

Figure A—1 BAW-2251A, Appendix B, Figure 3-1

A.2 New B&WOG J–Δa Data and Comparison to B&WOG J-R Model

Figure A—2 Jd (0.1) vs Fluence B&WOG J-R Model 4B and New Test Data (Normalized to Standard Conditions)

[ ]

A.3 New B&WOG J-R Model

[ ]

Table A—3 Jd Model Coefficients (Models 4B, 5B, and 6B)

Figure A—3 Original and New Data and Model Fit Normalized at Standardized Conditions vs Δa

Figure A—4 Original and New Data and Model Fit Normalized at Standard Conditions vs Fluence

Figure A—5 Model 6B Residuals vs Fitted Values

Figure A—6 Model 6B Standardized Residuals vs Fitted Values

Figure A—7 Normal Q-Q Plot of Standardized Residuals

A.4 Model 6B Reconciliation to EMA Results Presented in Section 6.0

[ ]

Figure A—8 Comparison of Models 4B, 5B, and 6B at Standard Conditions

[ ]

Table A—4 EMA Reconciliation for Limiting RV Shell Welds—Models 4B and 6B

A.5 Summary and Conclusions

Appendix Table of Contents

1. Letter, OG-17-329 Revision 1, Submittal of BAW-2192-P, Revision 0, Supplement 1 and BAW-2178-P, Revision 0, Supplement 1 (PA-MSC-1481), December 15, 2017.

2. NRC Letter of Acceptance for Review of the Pressurized Water Reactor Owners Group Topical Report BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," March 6, 2018 (ML18053A154).

3. AMRDEC Safe Access File Exchange from the NRC (Holonich for Benney), to the PWROG (Holderbaum), "Request for Additional Information from the Office of Nuclear Reactor Regulation BAW-2192, Supplement 1, Revision 1, 'Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads', Topical Report, BAW-2178, Supplement 1, Revision 1, 'Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads Topical Report'," April 25, 2018.

4. Email from NRC (Benney), to the PWROG (Holderbaum) and Framatome (Page Blair), "BAW-2192 and BAW-2178 Revisions to RAIs 1 5 9.docx", May 16, 2018.

5. PWR Owner's Group letter OG-18-148, "Transmittal of the Response to Request for Additional Information, - RAIs 2, 4 and 6 Associated with BAW-2192-P, Supplement 1, Revision 0, 'Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads', - RAIs 7, 8, and 10 Associated with BAW-2178-P Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" (PA-MSC-1481), June 15, 2018.

6. PWR Owner's Group letter OG-18-155, Transmittal of the Response to Request for Additional Information: RAI 5 Associated with BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads", - RAI 9 Associated with BAW-2178-P Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" (PA-MSC-1481), June 29, 2018.

Program Management Office
20 International Drive
Windsor, Connecticut 06095

December 15, 2017

OG-17-329 Rev. 1

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Submittal of BAW-2192-P, Revision 0, Supplement 1 and BAW-2178-P, Revision 0, Supplement 1 (PA-MSC-1481)

References: Transmittal of the PWROG Meeting Materials from the November 8, 2017 NRC/PWROG Pre-Submittal Meeting for BAW-2192-P Supplement 1 and BAW-2178-P Supplement 1 (PA-MSC-1481), OG-17-332, dated December 11, 2017

The purpose of this letter is to submit the following Topical Report (TR) Supplements developed by the Pressurized Water Reactor Owners Group (PWROG) in program PA-MSC-1481 for Linde 80 Weld Low Upper-Shelf Toughness Fracture Mechanics Analysis:

1. BAW-2192-P, Supplement 1, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads
2. BAW-2178-P, Supplement 1, Revision 0, Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads

As discussed at the pre-submittal meeting with the NRC on November 8, 2017, the purpose of BAW-2192, Supplement 1, and BAW-2178, Supplement 1 is to provide the NRC with 80-year equivalent margins analyses (EMA) for Linde 80 and applicable Rotterdam welds associated with reactor vessels fabricated by the Babcock & Wilcox Company for the applicable units. The EMA is reported for all reactor vessel welds that are expected to receive neutron fluence > 1.0E17 n/cm2 (E > 1.0 MeV) at 80 years. The EMA is in direct support of future subsequent license renewal applications.

The EMA utilizes the B&WOG J-integral resistance (J-R) Model 4B reported in BAW-2192PA, Appendix B. Model 4B was developed based on fracture toughness test data obtained through approximately 1990 to support 40-years of operation. Eighty-year fluence estimates for the participating plants exceed 40-year fluence and use of Model 4B to estimate J-integral resistance values, including the associated model uncertainty, for 80-years is made by extrapolation of the model. To assess the model extrapolation uncertainty, Model 4B is compared to new fracture toughness test data (1990 to 2017) irradiated to fluence expected at 80-years. The majority of test data fell above the Model 4B mean and all of the test data fell above the Model 4B mean minus 2 standard error band. Therefore, use of Model 4B and associated uncertainty to extrapolate J-integral resistance for 80-year fluence applications was determined to be appropriate. This assessment is reported in BAW-2192, Supplement 1, Appendix A.

To further substantiate the use of Model 4B, all of the original fracture toughness data used to develop Model 4B was combined with new fracture toughness data, assuming the same model form, and a new Model 6B was generated. Model 6B was found to be essentially equivalent to Model 4B with respect to model mean and 2 standard errors. The EMA results reported herein using Model 4B were reconciled to Model 6B, with little or no change to the EMA results. Model 6B development and the EMA reconciliation to Model 4B are reported in BAW-2192, Supplement 1, Appendix A.

The PWROG requests the following from the NRC relative to review of these topical reports.

- For the Linde 80 and Rotterdam welds in scope of the reports--finding that the equivalent margins analyses have been projected to the end of the subsequent period of extended operation in accordance with 10 CFR 54.21 (c)(1)(ii)
- NRC approval of EMA--Japplied and use of B&WOG J-integral resistance Model 4B for fluence levels expected at 80-years for the participating plants.
- NRC to identify license renewal application applicant action items for use of the TRs by the participating utilities (see NRC SER of BAW-2251A)

The enclosed TR Supplements (Enclosures 1 and 3) contain information proprietary to AREVA Inc; which is supported by an affidavit signed by AREVA, owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations. The affidavit is included as Enclosure 5.

Accordingly, it is respectfully requested that this information which is proprietary to AREVA be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the information or supporting AREVA affidavit should reference this letter and should be addressed to Mr. Philip Opsal, Manager, Product Licensing, AREVA Inc., 3315 Old Forest Road, Lynchburg, Virginia 24506-0935.

TR Classification: As discussed above, these TR Supplements address low upper-shelf toughness fracture mechanics analyses of B&W fabricated reactor vessels, that will be used for evaluating the upper shelf energy (equivalent margins analyses), on a plant-specific basis.

Specialized Resource Availability: These TR Supplements are being submitted to the NRC for review and approval so that the NRC approved version can be utilized for performing plant-specific evaluations of the equivalent margins analyses for the applicable units. NRC approval of the generic TR Supplements will provide a common approach that will be utilized to address the low upper-shelf toughness fracture mechanics analyses of B&W fabricated reactor vessels.

This letter transmits BAW-2192-P, Supplement 1, Revision 0 (Enclosure 1), BAW-2192-NP, Supplement 1, Revision 0 (Enclosure 2), BAW-2178-P, Supplement 1, Revision 0 (Enclosure 3), and BAW-2178-NP, Supplement 1, Revision 0, (Enclosure 4). A notarized Affidavit is provided as Enclosure 5.

Applicability: These TR Supplements are applicable to B&W fabricated Nuclear Steam Supply System (NSSS) plants that are participating in the PWROG program, PA-MSC-1481.

Request for Review Fee Waiver

The PWROG will be requesting that a fee waiver be considered for BAW-2192-P Supplement 1 and BAW-2178-P Supplement.

- This is the first submittal in industry that supports operation to 80 years. Review of these topical reports will permit the NRC to review an industry first evaluation of time-limited aging analyses (TLAA) in accordance with 54.21 C (1) (ii) for subsequent license renewal.
- The B&WOG J-R model has historically been compared to the Linde 80 J-R model reported in NUREG/CR-5729. The NUREG/CR-5729 model is based on data that does not cover the subsequent period of operation. If interested, the NRC may request the additional test data for use to update the NUREG/CR-5729 model.

NRC Review Schedule

The PWROG requests that the NRC complete their review of the TR by December 31, 2018.

Correspondence related to the non-proprietary transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (434) 832-2382 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

D. Page Blair Approving for J. Stringfellow

Jack Stringfellow
Chief Operating Officer & Chairman
Pressurized Water Reactor Owners Group

NJS:DRPB:am

Enclosure 1: BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads" (Proprietary)
Enclosure 2: BAW-2192-NP, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads"
Enclosure 3: BAW-2178-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" (Proprietary)
Enclosure 4: BAW-2178-NP, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads"
Enclosure 5: Affidavit for Withholding Proprietary Information

cc with enclosures:
B. Benney, US NRC

cc with non-proprietary enclosures:
PWROG Materials Committee Representatives in MSC-1481

cc without enclosures:
PWROG Steering and Management Committee
PWROG Licensing Committee
PWROG PMO
E. Dorman, AREVA
J. Fleck, AREVA
B. Haibach, AREVA
P. Thallapragada, AREVA
M. Rinckel, AREVA
P. Opsal, AREVA
B. Watson, AREVA
G. Elliott, AREVA
D. Page Blair, AREVA

March 6, 2018

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: ACCEPTANCE FOR REVIEW OF THE PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT BAW-2192-P, SUPPLEMENT 1, REVISION 0, LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVELS A & B SERVICE LOADS

Dear Mr. Nowinowski:

By letter dated December 15, 2017 (Agencywide Documents Access and Management System Accession No. ML17354A011), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has found that the material presented is sufficient to begin our review. The NRC staff expects to issue its request for additional information by June 1, 2018, and issue its draft safety evaluation (SE) by August, 2018. This schedule information takes in consideration the NRC's current review priorities and available technical resources and may be subject to change. If modifications to these dates are deemed necessary, we will provide appropriate updates to this information. Section 170.21 of Title 10 of the *Code of Federal Regulations* requires that TRs are subject to fees based on the full cost of the review.

As with all TRs, the SE will be reviewed by the NRC's Office of the General Counsel (OGC) to determine whether it falls within the scope of the Congressional Review Act (CRA). During the course of this review, OGC considers whether any endorsement or acceptance of a TR by the NRC amounts to a rule as defined in the CRA. If this initial review concludes that the SE, with its accompanying TR, may be a rule, the NRC will forward the package to the Office of Management and Budget (OMB) for further review and consideration. Any review by OMB would impact the schedule for the issuance of the final SE.

If you have questions regarding this matter, please contact Brian Benney at (301) 415-2767.

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket No. 99902037

SUBJECT: ACCEPTANCE FOR REVIEW OF THE PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT BAW-2192-P, SUPPLEMENT 1, REVISION 0, LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR LEVELS A & B SERVICE LOADS
DATED: MARCH 6, 2018

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RINCKEL Mark (FRA-IB)

To: RINCKEL Mark (FRA-IB)
Subject: FW: RAIs for BAW-2178 and BAW-2192

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REQUEST FOR ADDITIONAL INFORMATION FROM THE

OFFICE OF NUCLEAR REACTOR REGULATION

BAW-2192, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR

LEVEL A&B SERVICE LOADS TOPICAL REPORT

BAW-2178, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP FOR

LEVEL C&D SERVICE LOADS TOPICAL REPORT

Note, proprietary information in this enclosure is denoted by bold brackets. **[ ]**

Regulatory Basis

10 CFR Part 50, Appendix G provides the NRC staff's criteria for maintaining acceptable levels of upper shelf energy (USE) for the reactor pressure vessel (RPV) beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires RPV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the licensed period of operation of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Such analyses are referred to as "equivalent margins analyses," or EMAs. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant RPV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H RV material surveillance program.

The NRC staff is reviewing the Pressurized Water Reactor Owner's Group (PWROG) topical reports (TRs) BAW-2192, Rev. 0 Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A&B Service Loads," (Ref. 1) and BAW-2178, Rev. 0, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads," (Ref. 2) which provides the EMAs applicable to 80 years of operation for several plants. In order to verify the technical adequacy of the PWROG's EMAs, the staff requires additional information as detailed below.

RAIs Related to BAW-2192P, Supplement 1 Rev. 0.

RAI 1

Section 4.1 of the BAW-2192, Rev. 0, Supplement 1 (the TR) states, "Consistent with BAW-2192PA, Revision 0 (Ref. 3), this J-R model is used for Linde 80 welds and Rotterdam welds." BAW-2192PA did not mention Rotterdam welds. Provide justification that the J-integral resistance (J-R) model based on Linde 80 welds is applicable to Rotterdam welds. This justification should consider other J-R models which has been used for RPV welds.

RAI 2

Figure A-3 plots original and new data and model fit normalized at standardized conditions versus Δa (change in crack size). It shows that **[**

] Explain why it is valid to use the proposed J-R curve, **[**

]

RAI 4

Section 5.2 of the TR states, "For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis." Identify the plant, for which the three-dimensional finite element model was developed, and explain why this plant-specific finite element analysis is applicable to Surry and Turkey Point reactor vessels.

RAI 5

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the maximum error by using a flat plate solution instead of a cylindrical plate solution?

RAI 6

For Turkey Point, Unit 3 (TP3), and Turkey Point, Unit 4 (TP4), the staff verified the copper (Cu) and nickel (Ni) content is consistent with License Amendment Request for Extended Power Uprate, Attachment 4, L-2010-113, Attachment 4 (Ref. 4), except the staff noted the Cu and Ni values of 0.23 % Cu and 0.59 % Ni for Heat No. 71249, used in the TP3 inlet nozzle to RPV weld according to TR Table 3-1, are not consistent with Ref. 10, Table 2.1.2-1. In Reference 10, Table 2.1.2-1 the three heats of material associated with the inlet/outlet nozzle welds contain 0.34 % Cu, 0.68 % Ni , 0.16 % Cu, 0.57 % Ni, and 0.19 % Cu, 0.57 % Ni. The first one is not consistent with the TR, while the last two are consistent. Therefore, the staff requests the PWROG to resolve this discrepancy. If the TR is incorrect, please provide an update to the TR to correct these Cu and Ni values and the associated calculations.

RAIs related to BAW-2178, Rev. 0 Supplement 1

RAI 7

RG 1.161 requires identification of limiting Service Level C and D design transients in accordance with Standard Review Plan 3.9.3. The NRC noted that (1) RG 1.161 was issued after publication of BAW-2178PA, Rev. 0, and (2) the selected limiting transients for Service Level C and D design transients in the TR are not identical to those in BAW-2178PA, Rev. 0. The staff is concerned that the most bounding transient could be missed during the qualitative selection process.

Table 1 below compares the transients evaluated in the TR versus those originally evaluated in BAW-2178PA, Rev. 0. (Ref. 5) There were differences in the nomenclature used for some of the transients evaluated in the two reports, as well as some additional transients considered in the TR.

Table 1 - Comparison of Transients in TR vs. BAW-2178, Rev. 0

Plant	Level	BAW-2178, Rev. 0 Suppl. 1	BAW-2178PA, Rev. 0
Oconee Nuclear Station (ONS) 1,2,3	C	Stuck Open Turbine Bypass Valve (SOTBV)	[ ] [ ]
	D	Design Basis Steam Line Break (DB-SLB) Steam Line Break (ALT-SLB) Core Flood Line Break (CFLB) Hot Leg Large Break Loss of Coolant Accident (HL-LOCA)	[ ] [ ]
Surry 1, 2	C D	Steam Line Break (SM-0979) Steam Line Break (SSDC 1.3 SLB)	[ ] [ ]
TP 3, 4	C D	Steam Line Break (SLB Without Offsite Power) Steam Line Break (SSDC 1.3 SLB)	[ ] [ ]

The staff therefore requests the following information:

a) Reconcile the differences between the transients evaluated in BAW-2178PA, Rev. 0, and the TR.

b) Provide the source/reference for the transients evaluated, such as an Updated Final Safety Analysis Report (UFSAR) section, or other design basis document.

c) Demonstrate compliance with Standard Review Plan 3.9.3 for the selection of the Level C and D transients in this supplement to eliminate the NRC staff's concern that the truly bounding transient could be missed during the qualitative selection.

d) In TR Figures 4-5 ad 4-6, [

] Considering that the two plants have very similar RPV geometry, explain why the applied loadings are so different for the same transient.

RAI 8

The TR states that the maximum one-tenth wall thickness (1/10T) adjusted reference temperature ART for the ONS 1, 2, and 3 nozzle-to-shell welds is [ ], and the TR uses a 1/10T ART of [ ] for the Surry and Turkey Point limiting nozzle-to-shell welds. The staff could not independently confirm these ART values. The staff therefore requests that the PWROG:

a) Identify which material heats correspond to the two ART values above.

b) Provide the unirradiated reference temperature ($RT_{NDT(u)}$), σ_Δ , σ_I, fluence, chemistry factor, copper and nickel used to calculate the two ART values above, and the source/reference for these values.

RAI 9

In cladding stress calculation related to Section 5.2.3 of the Supplement, [ ] Address the sensitivity of this assumption.

RAI 10

Section 5.3.1.1 mentioned that, "The stress intensity factor K_I calculated by the PCRIT code is the sum of thermal, residual, and pressure terms." Since RG 1.161 and the ASME Code, Section XI, Appendix K do not consider residual stresses, please confirm that your assumed residual stress distribution is conservative such that it would resulted in a positive contribution to total applied K_I.

References

1. BAW-2192P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," December 31, 2017 (ADAMS Accession No. ML17354A013
2. BAW-2178P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads," December 31, 2017 (ADAMS Accession No. ML18029A200
3. "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads," April 30, 1994, ADAMS Legacy Accession No. 9406240263
4. Turkey Point, Units 3 and 4 - License Amendment Request for Extended Power Uprate, Attachment 4; Licensing Report, December 14, 2010 (ADAMS Accession No. ML103560177)
5. BAW-2178PA, Revision 00, "Low Upper Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level C and D Conditions," April 1994, ADAMS Accession (Legacy) 9406290288 (P)

From:	Benney, Brian
To:	PAGE BLAIR Danielle (FRA-IB)
Cc:	Holderbaum, Chad M.
Subject:	BAW-2192 and BAW-2178 Revisions to RAIs 15 9.docx
Date:	Wednesday, May 16, 2018 9:46:54 AM
Attachments:	BAW-2192 and BAW-2178 Revisions to RAIs 15 9.docx

---

Danielle,
Here are the revised RAIs based on the clarification call on Monday. RAI #1 has been deleted.
Please let me know if a different response date (other than June 15) is needed.
Thank you,
Brian

RAI-1

Section 4.1 of the BAW-2192, Rev. 0, Supplement 1 (the TR) states, "Consistent with BAW-2192PA, Revision 0 (Ref. 3), this J-R model is used for Linde 80 welds and Rotterdam welds." BAW-2192PA did not mention Rotterdam welds. Provide justification that the J-integral resistance (J-R) model based on Linde 80 welds is applicable to Rotterdam welds. This justification should consider other J-R models which has been used for RPV welds.

Revision:

Deletion.

RAI-5

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the maximum error by using a flat plate solution instead of a cylindrical plate solution?

Revision:

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the percent difference in the stress intensity factor by using a flat plate solution instead of a cylindrical plate solution?

RAI-9

To gain confidence in the cladding stress evaluation methodology presented in Section 5.2.3 of the Supplement, please provide for the axial flaw in a clad RPV, the actual stress distribution and the cladding stress distribution and their associated stress intensity factors. For the axial flaw in the same RPV without cladding, please provide the stress distribution and its corresponding stress intensity factor.

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

BAW-2178-P, Supplement 1, Revision 0
BAW-2192-P, Supplement 1, Revision 0
Project Number 99902037

June 15, 2018

OG-18-148

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Transmittal of the Response to Request for Additional Information, - RAIs 2, 4 and 6 Associated with BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads", - RAIs 7, 8, and 10 Associated with BAW-2178-P Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" PA-MSC-1481

References:

1. Letter OG-17-329 Rev 1, Submittal of BAW-2192-P, Revision 0, Supplement 1 and BAW-2178-P, Revision 0, Supplement 1 (PA-MSC-1481), dated December 15, 2017

2. NRC Letter of Acceptance for Review of the Pressurized Water Reactor Owners Group Topical Report BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," dated March 6, 2018 (ML18053A154)

3. NRC Letter of Acceptance for Review of the Pressurized Water Reactor Owners Group Topical Report BAW-2178-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads," dated March 6, 2018 (ML18053A149)

4. AMRDEC Safe Access File Exchange from the NRC (Holonich for Benney) to the PWROG (Holderbaum), "Request for Additional Information from the Office of Nuclear Reactor Regulation BAW-2192, Supplement 1, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel

Working Group for Levels A & B Service Loads Topical Report, BAW-2178, Supplement 1, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads Topical Report," dated April 25, 2018

5. Email from the NRC (Benney) to the PWROG (Holderbaum) and Framatome (Page Blair), "BAW-2192 and BAW-2178 Revisions to RAIs 1 5 9.docx", dated May 16, 2018

On December 15, 2017, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of BAW-2192-P, Supplement 1, Revision 0 and BAW-2178-P, Supplement 1, Revision 0 for referencing in regulatory actions (Reference 1). The NRC Staff has accepted the Topical Reports for review on March 6, 2018 (References 2 and 3).

The PWROG was notified on April 25, 2018 of the NRC staff's determination that additional information is needed to complete the review (Reference 4). The PWROG, supported by Framatome and participating utility members, held a call with the NRC staff to discuss clarification of the RAIs on May 14, 2018. Subsequently on May 16, 2018, the NRC staff provided an email communication to remove RAI-1 and revise RAI-5 and RAI-9 (Reference 5). NRC also acknowledged that RAI-3 placeholder was not used and thus RAI-3 did not require a response.

Enclosures 1 and 2 to this letter provide responses to RAIs 2, 4 and 6 (Reference 4) Associated with BAW-2192-P, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads".

Enclosures 3 and 4 to this letter provide responses to RAIs 7, 8, and 10 (Reference 4) Associated with BAW-2178-P, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads".

Responses to NRC RAIs 5 and 9 are being developed with a target submittal date of June 29, 2018.

The enclosed RAI responses (Enclosures 1 and 3) contain information proprietary to Framatome Inc (previously known as AREVA Inc); which is supported by an affidavit signed by Framatome, owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations. The affidavit is included as Enclosure 5.

Accordingly, it is respectfully requested that this information which is proprietary to Framatome be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the information or supporting Framatome affidavit should reference this letter and should be addressed to Mr. Philip Opsal, Manager, Product Licensing, AREVA Inc., 3315 Old Forest Road, Lynchburg, Virginia 24506-0935.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken Schrader, COO & Chairman
PWR Owners Group

JKS:am

Enclosure 1: BAW-2192-P, Supplement 1, Revision 0, Q1P, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 1 (RAIs 2, 4 and 6)" (Proprietary)

Enclosure 2: BAW-2192-NP, Supplement 1, Revision 0, Q1NP, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 1 (RAIs 2, 4 and 6)"

Enclosure 3: BAW-2178-P, Supplement 1, Revision 0, Q1P, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 1 (RAIs 7, 8 and 10)" (Proprietary)

Enclosure 4: BAW-2178-NP, Supplement 1, Revision 0, Q1NP, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 1 (RAIs 7, 8 and 10)"

Enclosure 5: Affidavit for Withholding Proprietary Information

cc with enclosures:
B. Benney, US NRC

cc with enclosures:
PWROG Materials Committee Representatives in MSC-1481 Revisions 0-2

cc without enclosures:
PWROG Steering and Management Committee
PWROG Licensing Committee
PWROG PMO
E. Dorman, Framatome
J. Fleck, Framatome
B. Haibach, Framatome
P. Thallapragada, Framatome
M. Rinckel, Framatome
P. Opsal, Framatome
G. Elliott, Framatome
T. Zalewski, Westinghouse
J. Andrachek. Westinghouse
A. Udyawar, Westinghouse

Electronically Approved Records Are Authenticated in the Electronic Document Management System

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Philip A. Opsal. I am Manager, Product Licensing, for Framatome Inc., (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome Inc., to determine whether certain Framatome Inc. information is proprietary. I am familiar with the policies established by Framatome Inc. to ensure the proper application of these criteria.

3. I am familiar with the Framatome Inc. information contained in the following documents, referred to herein as "Documents." :

- BAW-2192P-00 Supplement 1-00-000_Q1P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 1 (RAIs 2, 4, and 6)"
- BAW-2178P-00 Supplement 1-00-000_Q1P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 1 (RAIs 7, 8 and 10)"

Information contained in these Documents has been classified by Framatome Inc. as proprietary in accordance with the policies established by Framatome Inc. for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome Inc. and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome Inc. to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome Inc.'s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome Inc.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome Inc. in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome Inc., would be helpful to competitors to Framatome Inc., and would likely cause substantial harm to the competitive position of Framatome Inc.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c) and 6(d) above.

7. In accordance with Framatome Inc.'s policies governing the protection and control of information, proprietary information contained in these Documents has been made available, on a limited basis, to others outside Framatome Inc. only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome Inc.'s policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 13th

day of June, 2018.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2018

framatome

Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 1 (RAIs 2, 4, and 6)

BAW-2192NP-00-Supplement 1-00-000_Q1NP
Revision 0

June 2018

Framatome Inc.

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
EMA	Equivalent Margins Analysis
TR	Topical Report
TP	Turkey Point

1.0 NRC RAIS

The NRC staff is reviewing the Pressurized Water Reactor Owner's Group (PWROG) topical reports (TRs) BAW-2192, Rev. 0 Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A&B Service Loads," and BAW-2178, Rev. 0, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads," which provides the EMAs applicable to 80 years of operation for several plants. In order to verify the technical adequacy of the PWROG's Equivalent Margins Analysis (EMA), the NRC staff sent RAIs 1-10 to the PWROG by e-mail dated May 7, 2018. RAIs 1-10 are presented in Section 3.0 of this document.

A telecom was conducted with the NRC on 5/14/2018 to discuss clarifications to RAIs (1, 3, 5, 7, 8, and 9). During this telecom, the NRC Staff acknowledged that RAI 3 was not used by the Staff and no response for RAI 3 is provided. Following the telecom, the NRC deleted RAI 1 and revised RAIs 5 and 9; RAIs 2, 4, 6, 7, 8 and 10 were not revised based on the telecom. Revisions to RAIs 1, 5, and 9 were sent to Danielle Page Blair (Framatome) by e-mail dated May 16, 2018 from Brian Benney (NRC). The deletion of RAI 1 and revised RAIs 5 and 9 are presented in Section 4.0 of this document. Responses to RAIs 2, 4, and 6 that apply to BAW-2192, Supplement 1, are reported in this document. The response to RAI 5 will be provided in a revision to this document (BAW-2192P-00-Supplement 1-00-000_Q2P) at the end of June 2018. Responses to RAIs 7, 8, and 10 that apply to BAW-2178, Supplement 1, are reported in BAW-2178P-00-Supplement 1-00-000_Q1P. The response to RAI 9 will be provided in BAW-2178P-00-Supplement 1-00-000_Q2P at the end of June 2018.

2.0 RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION

2.1 *RAI 2*

2.1.1 NRC RAI Text (from Section 3.0)

Figure A-3 plots original and new data and model fit normalized at standardized conditions versus Δa (change in crack size). It shows that [

] Explain why it is valid to use the proposed J-R curve, [

]

2.1.2 PWROG Response to RAI 2

BAW-2192PA, Figure B-13, is a plot of normalized Jd (Kip-in/in^2) versus delta a (in) for the B&WOG dataset. Figure B-13 also includes the Model 4B mean (at standard conditions) and bounds of ± 3 standard errors (Se). Note that BAW-2192PA does not specifically identify that ± 3 Se are reported for all scatter plots; this was confirmed by review of the source reference for BAW-2192PA, Figure B-13.

[

]

Figure 2—1
Original and New Data and Model Fit Normalized at Standardized Conditions vs Δa with ± 3Se

2.1.3 References for Response to RAI 2

Not applicable.

2.2 *RAI 4*

2.2.1 NRC RAI Text (from Section 3.0)

Section 5.2 of the TR states, "For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis." Identify the plant, for which the three-dimensional finite element model was developed, and explain why this plant-specific finite element analysis is applicable to Surry and Turkey Point reactor vessels.

2.2.2 PWROG Response to RAI 4

The stress analysis consists of two separate three-dimensional FEA models: one applicable for the Turkey Point Units 3 and 4 and the other applicable for Surry Units 1 and 2. Each model used appropriate plant specific geometry, material property, loading conditions (nozzle loads), and plant operating conditions (transient temperature/pressure). Thus, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from plant-specific detailed three-dimensional finite element analysis for Turkey Point and Surry Reactor Vessels.

2.2.3 References for Response to RAI 4

Not Applicable.

2.3 *RAI 6*

2.3.1 NRC RAI Text (from Section 3.0)

For Turkey Point, Unit 3 (TP3), and Turkey Point, Unit 4 (TP4), the staff verified the copper (Cu) and nickel (Ni) content is consistent with License Amendment Request for Extended Power Uprate, Attachment 4, L-2010-113, Attachment 4 (Ref. 4), except the staff noted the Cu and Ni values of 0.23 % Cu and 0.59 % Ni for Heat No. 71249, used in the TP3 inlet nozzle to RPV weld according to TR Table 3-1, are not consistent with Ref. 10, Table 2.1.2-1. In Reference 10, Table 2.1.2-1 the three heats of material associated with the inlet/outlet nozzle welds contain 0.34 % Cu, 0.68 % Ni , 0.16 % Cu, 0.57 % Ni, and 0.19 % Cu, 0.57 % Ni. The first one is not consistent with the TR, while the last two are consistent. Therefore, the staff requests the PWROG to resolve this discrepancy. If the TR is incorrect, please provide an update to the TR to correct these Cu and Ni values and the associated calculations.

2.3.2 PWROG Response to RAI 6

The Staff noted that a weld wire chemistry of 0.34 % Cu, 0.68 % Ni was listed in Table 2.1.2-1 of the EPU license amendment request (LAR) [1] for the TP3 inlet nozzle weld, but not in Table 3-1 of BAW-2192P, Supplement 1, Revision 0 (hereafter referred to as the TR) [2]. Several weld wire heats were used when fabricating the TP3 and TP4 inlet and outlet nozzle welds. In Table 2.1.2-1 of the EPU LAR [1], because the weld wire heats applicable to each nozzle weld were unknown at the time of the EPU submittal, all the weld wire heats were listed for all nozzle welds for both TP3 and TP4. The EPU LAR listed a weld wire chemistry of 0.34 % Cu, 0.68 % Ni for the TP3 inlet nozzle welds when, in fact, this weld wire chemistry is only present at TP4. In Table 3-1 of the TR [2], only the specific weld wire heats used to fabricate each nozzle weld type (inlet and outlet) at each unit (TP3 and TP4) were included as they are now identified.

The Staff also noted that a weld wire chemistry of 0.23 % Cu, 0.59 % Ni was listed in Table 3-1 of the TR [2] for the TP3 inlet nozzle welds, but not in Table 2.1.2-1 of the EPU LAR [1]. The presence of this weld wire heat used to fabricate TP3 inlet nozzle welds was identified in a 2017 records search while developing the TR, so it is appropriate to include a weld wire chemistry of 0.23 % Cu, 0.59 % Ni in Table 3-1 of the TR for the TP3 inlet nozzle welds.

2.3.3 References for Response to RAI 6

[1] Turkey Point, Units 3 and 4 - License Amendment Request for Extended Power Uprate, Letter L-2010-113, Attachment 4; Licensing Report, December 14, 2010 (ADAMS Accession No. ML103560177)

[2] BAW-2192P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," December 31, 2017 (ADAMS Accession No. ML17354A013)

3.0 NRC RAIS—MAY 7, 2018

REQUEST FOR ADDITIONAL INFORMATION FROM THE

OFFICE OF NUCLEAR REACTOR REGULATION

BAW-2192, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP

FOR LEVEL A&B SERVICE LOADS TOPICAL REPORT

BAW-2178, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP

FOR LEVEL C&D SERVICE LOADS TOPICAL REPORT

Note, proprietary information in this enclosure is denoted by bold brackets. **[ ]**

Regulatory Basis

10 CFR Part 50, Appendix G provides the NRC staff's criteria for maintaining acceptable levels of upper shelf energy (USE) for the reactor pressure vessel (RPV) beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires RPV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the licensed period of operation of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Such analyses are referred to as "equivalent margins analyses," or EMAs. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant RPV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H RV material surveillance program.

The NRC staff is reviewing the Pressurized Water Reactor Owner's Group (PWROG) topical reports (TRs) BAW-2192, Rev. 0 Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A&B Service Loads," (Ref. 1) and BAW-2178, Rev. 0, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads," (Ref. 2) which provides the EMAs applicable to 80 years of operation for several plants. In order to verify the technical adequacy of the PWROG's EMAs, the staff requires additional information as detailed below.

RAIs Related to BAW-2192P, Supplement 1 Rev. 0.

RAI 1

Section 4.1 of the BAW-2192, Rev. 0, Supplement 1 (the TR) states, "Consistent with BAW-2192PA, Revision 0 (Ref. 3), this J-R model is used for Linde 80 welds and Rotterdam welds." BAW-2192PA did not mention Rotterdam welds. Provide justification that the J-integral resistance (J-R) model based on Linde 80 welds is applicable to Rotterdam welds. This justification should consider other J-R models which has been used for RPV welds.

RAI 2

Figure A-3 plots original and new data and model fit normalized at standardized conditions versus Δa (change in crack size). It shows that [

] Explain why it is valid to use the proposed J-R curve, [

]

RAI 4

Section 5.2 of the TR states, "For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis." Identify the plant, for which the three-dimensional finite element model was developed, and explain why this plant-specific finite element analysis is applicable to Surry and Turkey Point reactor vessels.

RAI 5

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the maximum error by using a flat plate solution instead of a cylindrical plate solution?

RAI 6

For Turkey Point, Unit 3 (TP3), and Turkey Point, Unit 4 (TP4), the staff verified the copper (Cu) and nickel (Ni) content is consistent with License Amendment Request for Extended Power Uprate, Attachment 4, L-2010-113, Attachment 4 (Ref. 4), except the staff noted the Cu and Ni values of 0.23 % Cu and 0.59 % Ni for Heat No. 71249, used in the TP3 inlet nozzle to RPV weld according to TR Table 3-1, are not consistent with Ref. 10, Table 2.1.2-1. In Reference 10, Table 2.1.2-1 the three heats of material associated with the inlet/outlet nozzle welds contain 0.34 % Cu, 0.68 % Ni , 0.16 % Cu, 0.57 % Ni, and 0.19 % Cu, 0.57 % Ni. The first one is not consistent with the TR, while the last two are consistent. Therefore, the staff requests the PWROG to resolve this discrepancy. If the TR is incorrect, please provide an update to the TR to correct these Cu and Ni values and the associated calculations.

<u>RAIs related to BAW-2178, Rev. 0 Supplement 1</u>

RAI 7

RG 1.161 requires identification of limiting Service Level C and D design transients in accordance with Standard Review Plan 3.9.3. The NRC noted that (1) RG 1.161 was issued after publication of BAW-2178PA, Rev. 0, and (2) the selected limiting transients for Service Level C and D design transients in the TR are not identical to those in BAW-2178PA, Rev. 0. The staff is concerned that the most bounding transient could be missed during the qualitative selection process.

Table 1 below compares the transients evaluated in the TR versus those originally evaluated in BAW-2178PA, Rev. 0. (Ref. 5) There were differences in the nomenclature used for some of the transients evaluated in the two reports, as well as some additional transients considered in the TR.

Table 1 - Comparison of Transients in TR vs. BAW-2178, Rev. 0

Plant	Level	BAW-2178, Rev. 0 Suppl. 1	BAW-2178PA, Rev. 0
Oconee Nuclear Station (ONS) 1,2,3	C	Stuck Open Turbine Bypass Valve (SOTBV)	[ ] [ ]
	D	Design Basis Steam Line Break (DB-SLB) Steam Line Break (ALT-SLB) Core Flood Line Break (CFLB) Hot Leg Large Break Loss of Coolant Accident (HL-LOCA)	[ ] [ ]
Surry 1, 2	C D	Steam Line Break (SM-0979) Steam Line Break (SSDC 1.3 SLB)	[ ] [ ]
TP 3, 4	C D	Steam Line Break (SLB Without Offsite Power) Steam Line Break (SSDC 1.3 SLB)	[ ] [ ]

The staff therefore requests the following information:

a) Reconcile the differences between the transients evaluated in BAW-2178PA, Rev. 0, and the TR.

b) Provide the source/reference for the transients evaluated, such as an Updated Final Safety Analysis Report (UFSAR) section, or other design basis document.

c) Demonstrate compliance with Standard Review Plan 3.9.3 for the selection of the Level C and D transients in this supplement to eliminate the NRC staff's concern that the truly bounding transient could be missed during the qualitative selection.

d) In TR Figures 4-5 ad 4-6, [ ] Considering that the two plants have very similar RPV geometry, explain why the applied loadings are so different for the same transient.

RAI 8

The TR states that the maximum one-tenth wall thickness (1/10T) adjusted reference temperature ART for the ONS 1, 2, and 3 nozzle-to-shell welds is [ ], and the TR uses a 1/10T ART of [ ] for the Surry and Turkey Point limiting nozzle-to-

shell welds. The staff could not independently confirm these ART values. The staff therefore requests that the PWROG:

a) Identify which material heats correspond to the two ART values above.

b) Provide the unirradiated reference temperature ($RT_{NDT(u)}$), σ_Δ , σ_I, fluence, chemistry factor, copper and nickel used to calculate the two ART values above, and the source/reference for these values.

RAI 9

In cladding stress calculation related to Section 5.2.3 of the Supplement, [ ] Address the sensitivity of this assumption.

RAI 10

Section 5.3.1.1 mentioned that, "The stress intensity factor K_I calculated by the PCRIT code is the sum of thermal, residual, and pressure terms." Since RG 1.161 and the ASME Code, Section XI, Appendix K do not consider residual stresses, please confirm that your assumed residual stress distribution is conservative such that it would resulted in a positive contribution to total applied K_I.

References

1. BAW-2192P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," December 31, 2017 (ADAMS Accession No. ML17354A013
2. BAW-2178P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads," December 31, 2017 (ADAMS Accession No. ML18029A200
3. "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads," April 30, 1994, ADAMS Legacy Accession No. 9406240263
4. Turkey Point, Units 3 and 4 - License Amendment Request for Extended Power Uprate, Attachment 4; Licensing Report, December 14, 2010 (ADAMS Accession No. ML103560177)
5. BAW-2178PA, Revision 00, "Low Upper Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level C and D Conditions," April 1994, ADAMS Accession (Legacy) 9406290288 (P)

4.0 NRC RAI 1, 5, AND 9 AMENDMENTS BASED ON 5-14-2018 TELECON

Following the clarification call with the NRC on 5/14/2018, the staff issued an e-mail from Brian Benney (NRC) to Danielle Page Blair dated May 16, 2018, 9:46 a.m. The attachment to the e-mail is provided below. RAI 3 was not used by the NRC.

RAI-1

Section 4.1 of the BAW-2192, Rev. 0, Supplement 1 (the TR) states, "Consistent with BAW-2192PA, Revision 0 (Ref. 3), this J-R model is used for Linde 80 welds and Rotterdam welds." BAW-2192PA did not mention Rotterdam welds. Provide justification that the J-integral resistance (J-R) model based on Linde 80 welds is applicable to Rotterdam welds. This justification should consider other J-R models which has been used for RPV welds.

Revision:

Deletion.

RAI-5

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the maximum error by using a flat plate solution instead of a cylindrical plate solution?

Revision:

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the percent difference in the stress intensity factor by using a flat plate solution instead of a cylindrical plate solution?

RAI-9

To gain confidence in the cladding stress evaluation methodology presented in Section 5.2.3 of the Supplement, please provide for the axial flaw in a clad RPV, the actual stress distribution and the cladding stress distribution and their associated stress

intensity factors. For the axial flaw in the same RPV without cladding, please provide the stress distribution and its corresponding stress intensity factor.

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

BAW-2178-P, Supplement 1, Revision 0
BAW-2192-P, Supplement 1, Revision 0
Project Number 99902037

June 29, 2018

OG-18-155

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Transmittal of the Response to Request for Additional Information: RAI 5 Associated with BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads", - RAI 9 Associated with BAW-2178-P Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" (PA-MSC-1481)

References:

1. Letter OG-17-329 Rev 1, Submittal of BAW-2192-P, Revision 0, Supplement 1 and BAW-2178-P, Revision 0, Supplement 1 (PA-MSC-1481), dated December 15, 2017

2. NRC Letter of Acceptance for Review of the Pressurized Water Reactor Owners Group Topical Report BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," dated March 6, 2018 (ML18053A154)

3. NRC Letter of Acceptance for Review of the Pressurized Water Reactor Owners Group Topical Report BAW-2178-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads," dated March 6, 2018 (ML18053A149)

4. AMRDEC Safe Access File Exchange from the NRC (Holonich for Benney) to the PWROG (Holderbaum), "Request for Additional Information from the Office of Nuclear Reactor Regulation BAW-2192, Supplement 1, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel

Working Group for Levels A & B Service Loads Topical Report, BAW-2178, Supplement 1, Revision 1, Low Upper-Shelf Toughness Fracture Mechanics Analysis for Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads Topical Report," dated April 25, 2018

5. Email from the NRC (Benney) to the PWROG (Holderbaum) and Framatome (Page Blair), "BAW-2192 and BAW-2178 Revisions to RAIs 1 5 9.docx", dated May 16, 2018

6. PWR Owner's Group letter OG-18-148, "Transmittal of the Response to Request for Additional Information, - RAIs 2, 4 and 6 Associated with BAW-2192-P, Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads", - RAIs 7, 8, and 10 Associated with BAW-2178-P Supplement 1, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" (PA-MSC-1481)

On December 15, 2017, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of BAW-2192-P, Supplement 1, Revision 0 and BAW-2178-P, Supplement 1, Revision 0 for referencing in regulatory actions (Reference 1). The NRC Staff has accepted the Topical Reports for review on March 6, 2018 (References 2 and 3).

The PWROG was notified on April 25, 2018 of the NRC staff's determination that additional information is needed to complete the review (Reference 4). The PWROG, supported by Framatome and participating utility members, held a call with the NRC staff to discuss clarification of the RAIs on May 14, 2018. Subsequently on May 16, 2018, the NRC staff provided an email communication to remove RAI-1 and revise RAI-5 and RAI-9 (Reference 5). NRC also acknowledged that RAI-3 placeholder was not used and thus RAI-3 did not require a response.

Enclosures 1 and 2 to this letter provide responses to RAI 5 (Reference 4) associated with BAW-2192-P, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads".

Enclosures 3 and 4 to this letter provide responses to RAI 9 (Reference 4) associated with BAW-2178-P, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads".

Responses to NRC RAIs 2, 4, 6, 7, 8 and 10 were submitted to the Document Control Desk on June 15th via PWR Owner's Group letter OG-18-148 (Reference 6). Therefore, this submittal completes the response to all NRC staff comments on the two subject Topical Reports.

The enclosed RAI responses (Enclosures 1 and 3) contain information proprietary to Framatome Inc (previously known as AREVA Inc); which is supported by an affidavit signed by Framatome, owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations. The affidavit is included as Enclosure 5.

Accordingly, it is respectfully requested that this information which is proprietary to Framatome be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the proprietary aspects of the information or supporting Framatome affidavit should reference this letter and should be addressed to Mr. Philip Opsal, Manager, Product Licensing, AREVA Inc., 3315 Old Forest Road, Lynchburg, Virginia 24506-0935.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken J Schrader

Ken Schrader, COO & Chairman
PWR Owners Group

JKS:am

Enclosure 1: BAW-2192P-00 Supplement 1-00-000_Q2P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 1 (RAI 5)" (Proprietary)

Enclosure 2: BAW-2192NP-00 Supplement 1-00-000_Q2NP, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 1 (RAI 5)"

Enclosure 3: BAW-2178P-00 Supplement 1-00-000_Q2P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 1 (RAI 9)" (Proprietary)

Enclosure 4: BAW-2178NP-00 Supplement 1-00-000_Q2NP, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 1 (RAI 9)"

Enclosure 5: Affidavit for Withholding Proprietary Information

cc with enclosures:
B. Benney, US NRC

cc with enclosures:
PWROG Materials Committee Representatives in MSC-1481 Revisions 0-2

cc without enclosures:
PWROG Steering and Management Committee
PWROG Licensing Committee
PWROG PMO
E. Dorman, Framatome
J. Fleck, Framatome
B. Haibach, Framatome
P. Thallapragada, Framatome
M. Rinckel, Framatome
P. Opsal, Framatome
G. Elliott, Framatome
T. Zalewski, Westinghouse
J. Andrachek. Westinghouse
A. Udyawar, Westinghouse

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Philip A. Opsal. I am Manager, Product Licensing, for Framatome Inc., (formally known as AREVA Inc.), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome Inc., to determine whether certain Framatome Inc. information is proprietary. I am familiar with the policies established by Framatome Inc. to ensure the proper application of these criteria.

3. I am familiar with the Framatome Inc. information contained in the following documents, referred to herein as "Documents." :

- BAW-2192P-00 Supplement 1-00-000_Q2P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 2 (RAI 5)"
- BAW-2178P-00 Supplement 1-00-000_Q2P, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 2 (RAI 9)"

Information contained in these Documents has been classified by Framatome Inc. as proprietary in accordance with the policies established by Framatome Inc. for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome Inc. and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome Inc. to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome Inc.'s research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome Inc.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome Inc. in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome Inc., would be helpful to competitors to Framatome Inc., and would likely cause substantial harm to the competitive position of Framatome Inc.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b), 6(c) and 6(d) above.

7. In accordance with Framatome Inc.'s policies governing the protection and control of information, proprietary information contained in these Documents has been made available, on a limited basis, to others outside Framatome Inc. only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome Inc.'s policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 25

day of June, 2018.

Heidi Hamilton Elder
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/2022
Reg. # 7777873

HEIDI HAMILTON ELDER
NOTARY PUBLIC
REGISTRATION # 7777873
COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES
DECEMBER 31, 2022

framatome

Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads, Response to NRC RAIs Set 2 (RAI 5)

BAW-2192NP-00-Supplement 1-00-000_Q2NP
Revision 0

June 2018

Framatome Inc.

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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List of Tables

Table 2—1 K_I for Axial Flaw in Cylindrical Shell Compared to Flat Plate-Oconee RV Nozzle-to-Shell Weld 2-4

Nomenclature

Acronym	Definition
EMA	Equivalent Margins Analysis
ONS	Oconee Nuclear Station
TR	Topical Report
TP	Turkey Point

1.0 NRC RAIS

The NRC staff is reviewing the Pressurized Water Reactor Owner's Group (PWROG) topical reports (TRs) BAW-2192, Rev. 0 Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A&B Service Loads," and BAW-2178, Rev. 0, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads," which provides the EMAs applicable to 80 years of operation for several plants. In order to verify the technical adequacy of the PWROG's Equivalent Margins Analysis (EMA), the NRC staff sent RAIs 1-10 to the PWROG by e-mail dated May 7, 2018. RAIs 1-10 are presented in Section 3.0 of this document.

A telecom was conducted with the NRC on 5/14/2018 to discuss clarifications to RAIs (1, 3, 5, 7, 8, and 9). Following the telecom, the NRC deleted RAI 1 and revised RAIs 5 and 9; RAIs 2, 4, 6, 7, 8 and 10 were not revised based on the telecom. Revisions to RAIs 1, 5, and 9 were sent to Danielle Page Blair (Framatome) by e-mail dated May 16, 2018 from Brian Benney (NRC). The deletion of RAI 1 and revised RAIs 5 and 9 are presented in Section 4.0 of this document. Responses to RAIs 2, 4, and 6 that apply to BAW-2192, Supplement 1, are reported in BAW-2192P-00-Supplement 1-00-000_Q1P. The response to RAI5 is provided in this document. Responses to RAIs 7, 8, and 10 that apply to BAW-2178, Supplement 1, are reported in BAW-2178P-00-Supplement 1-00-000_Q1P. The response to RAI 9 is provided in BAW-2178P-00-Supplement 1-00-000_Q2P.

2.0 RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION

2.1 *RAI 5*

2.1.1 NRC RAI Text (from Section 4.0)

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the percent difference in the stress intensity factor by using a flat plate solution instead of a cylindrical plate solution?

2.1.2 PWROG Response to RAI 5

Section 5.3.2 (Page 5-8, 3rd paragraph) of BAW-2192P, Supplement 1, Revision 0 (hereafter referred to as the TR) applies only to the Oconee reactor vessel nozzle-to-shell welds. The R/t ratio for the Oconee reactor vessel nozzle weld is obtained from the TR, Table 5-3: Oconee Ri/t= [ ] The stress intensity factor calculation utilized a flat plate solution for membrane and bending stresses from Newman and Raju (Reference [1]).

As required by 10 CFR 50 Appendix G, the equivalent margins analysis must use the latest edition and addenda of the ASME Code incorporated by reference into 10 CFR 50.55a(b)(2) at the time the analysis is submitted (i.e., ASME Code Section XI, 2013 Edition, Reference [2]). As discussed in Section 2.2 of the TR, the EMA is in full compliance with this requirement. RAI 5 requests that a cylindrical solution be prepared for comparison to the flat plate solution reported in the TR.

For comparison of flat plate solution to a cylindrical solution for the Oconee reactor vessel nozzle-to-shell welds, the stress intensity factor solution for an axial flaw in a cylindrical shell from [

]

[

]

The results of the calculation demonstrate that flat plate solution is conservative by approximately [ ] compared to the cylindrical shell solution in this case. It is noted that the degree of conservatism on the applied J-integral will be larger since J is proportional to K_I^2 under the small scale yielding conditions.

Table 2—1
K_I for Axial Flaw in Cylindrical Shell Compared to Flat Plate-Oconee RV Nozzle-to-Shell Weld

2.1.3 References for Response to RAI 5

[1] Newman, J. C., Jr and Raju, I. S., "An Empirical Stress Intensity Factor Equation for the Surface Crack", Engineering Fracture Mechanics, Vol. 15, 1981, pp. 185-192.

[2] ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2013 Edition.

[3] ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2017 Edition.

3.0 NRC RAIS—MAY 7, 2018

REQUEST FOR ADDITIONAL INFORMATION FROM THE

OFFICE OF NUCLEAR REACTOR REGULATION

BAW-2192, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP

FOR LEVEL A&B SERVICE LOADS TOPICAL REPORT

BAW-2178, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP

FOR LEVEL C&D SERVICE LOADS TOPICAL REPORT

Note: proprietary information in this enclosure is denoted by bold brackets. **[ ]**

Regulatory Basis

10 CFR Part 50, Appendix G provides the NRC staff's criteria for maintaining acceptable levels of upper shelf energy (USE) for the reactor pressure vessel (RPV) beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires RPV beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the licensed period of operation of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Such analyses are referred to as "equivalent margins analyses," or EMAs. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant RPV surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H RV material surveillance program.

The NRC staff is reviewing the Pressurized Water Reactor Owner's Group (PWROG) topical reports (TRs) BAW-2192, Rev. 0 Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A&B Service Loads," (Ref. 1) and BAW-2178, Rev. 0, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads," (Ref. 2) which provides the EMAs applicable to 80 years of operation for several plants. In order to verify the technical adequacy of the PWROG's EMAs, the staff requires additional information as detailed below.

RAIs Related to BAW-2192P, Supplement 1 Rev. 0.

RAI 1

Section 4.1 of the BAW-2192, Rev. 0, Supplement 1 (the TR) states, "Consistent with BAW-2192PA, Revision 0 (Ref. 3), this J-R model is used for Linde 80 welds and Rotterdam welds." BAW-2192PA did not mention Rotterdam welds. Provide justification that the J-integral resistance (J-R) model based on Linde 80 welds is applicable to Rotterdam welds. This justification should consider other J-R models which has been used for RPV welds.

RAI 2

Figure A-3 plots original and new data and model fit normalized at standardized conditions versus Δa (change in crack size). It shows that [

] Explain why it is valid to use the proposed J-R curve, [

]

RAI 4

Section 5.2 of the TR states, "For both the Surry and Turkey Point reactor vessels, the applied J-integrals at the nozzle-to-shell welds and the upper transition welds were determined using stresses from a detailed three-dimensional finite element analysis." Identify the plant, for which the three-dimensional finite element model was developed, and explain why this plant-specific finite element analysis is applicable to Surry and Turkey Point reactor vessels.

RAI 5

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the maximum error by using a flat plate solution instead of a cylindrical plate solution?

RAI 6

For Turkey Point, Unit 3 (TP3), and Turkey Point, Unit 4 (TP4), the staff verified the copper (Cu) and nickel (Ni) content is consistent with License Amendment Request for Extended Power Uprate, Attachment 4, L-2010-113, Attachment 4 (Ref. 4), except the staff noted the Cu and Ni values of 0.23 % Cu and 0.59 % Ni for Heat No. 71249, used in the TP3 inlet nozzle to RPV weld according to TR Table 3-1, are not consistent with Ref. 10, Table 2.1.2-1. In Reference 10, Table 2.1.2-1 the three heats of material associated with the inlet/outlet nozzle welds contain 0.34 % Cu, 0.68 % Ni , 0.16 % Cu, 0.57 % Ni, and 0.19 % Cu, 0.57 % Ni. The first one is not consistent with the TR, while the last two are consistent. Therefore, the staff requests the PWROG to resolve this discrepancy. If the TR is incorrect, please provide an update to the TR to correct these Cu and Ni values and the associated calculations.

<u>RAIs related to BAW-2178, Rev. 0 Supplement 1</u>

RAI 7

RG 1.161 requires identification of limiting Service Level C and D design transients in accordance with Standard Review Plan 3.9.3. The NRC noted that (1) RG 1.161 was issued after publication of BAW-2178PA, Rev. 0, and (2) the selected limiting transients for Service Level C and D design transients in the TR are not identical to those in BAW-2178PA, Rev. 0. The staff is concerned that the most bounding transient could be missed during the qualitative selection process.

Table 1 below compares the transients evaluated in the TR versus those originally evaluated in BAW-2178PA, Rev. 0. (Ref. 5) There were differences in the nomenclature used for some of the transients evaluated in the two reports, as well as some additional transients considered in the TR.

Table 1 - Comparison of Transients in TR vs. BAW-2178, Rev. 0

Plant	Level	BAW-2178, Rev. 0 Suppl. 1	BAW-2178PA, Rev. 0
Oconee Nuclear Station (ONS) 1,2,3	C	Stuck Open Turbine Bypass Valve (SOTBV)	[ ] [ ]
	D	Design Basis Steam Line Break (DB-SLB) Steam Line Break (ALT-SLB) Core Flood Line Break (CFLB) Hot Leg Large Break Loss of Coolant Accident (HL-LOCA)	[ ] [ ]
Surry 1, 2	C D	Steam Line Break (SM-0979) Steam Line Break (SSDC 1.3 SLB)	[ ] [ ]
TP 3, 4	C D	Steam Line Break (SLB Without Offsite Power) Steam Line Break (SSDC 1.3 SLB)	[ ] [ ]

The staff therefore requests the following information:

a) Reconcile the differences between the transients evaluated in BAW-2178PA, Rev. 0, and the TR.

b) Provide the source/reference for the transients evaluated, such as an Updated Final Safety Analysis Report (UFSAR) section, or other design basis document.

c) Demonstrate compliance with Standard Review Plan 3.9.3 for the selection of the Level C and D transients in this supplement to eliminate the NRC staff's concern that the truly bounding transient could be missed during the qualitative selection.

d) In TR Figures 4-5 ad 4-6, [ ] Considering that the two plants have very similar RPV geometry, explain why the applied loadings are so different for the same transient.

RAI 8

The TR states that the maximum one-tenth wall thickness (1/10T) adjusted reference temperature ART for the ONS 1, 2, and 3 nozzle-to-shell welds is [ ], and the TR uses a 1/10T ART of [ ] for the Surry and Turkey Point limiting nozzle-to-shell welds. The staff could not independently confirm these ART values. The staff therefore requests that the PWROG:

a) Identify which material heats correspond to the two ART values above.

b) Provide the unirradiated reference temperature ($RT_{NDT(u)}$), σ_Δ , σ_I, fluence, chemistry factor, copper and nickel used to calculate the two ART values above, and the source/reference for these values.

RAI 9

In cladding stress calculation related to Section 5.2.3 of the Supplement, [ ] Address the sensitivity of this assumption.

RAI 10

Section 5.3.1.1 mentioned that, "The stress intensity factor K_I calculated by the PCRIT code is the sum of thermal, residual, and pressure terms." Since RG 1.161 and the ASME Code, Section XI, Appendix K do not consider residual stresses, please confirm that your assumed residual stress distribution is conservative such that it would resulted in a positive contribution to total applied K_I.

References

1. BAW-2192P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," December 31, 2017 (ADAMS Accession No. ML17354A013.
2. BAW-2178P, Supp. 1, Rev. 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads," December 31, 2017 (ADAMS Accession No. ML18029A200.
3. "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A&B Service Loads," April 30, 1994, ADAMS Legacy Accession No. 9406240263.

4. Turkey Point, Units 3 and 4 - License Amendment Request for Extended Power Uprate, Attachment 4; Licensing Report, December 14, 2010 (ADAMS Accession No. ML103560177).

5. BAW-2178PA, Revision 00, "Low Upper Shelf Toughness Fracture Analysis of Reactor Vessels of B&W Owners Group Reactor Vessel Working Group for Level C and D Conditions," April 1994, ADAMS Accession (Legacy) 9406290288 (P).

4.0 NRC RAI 1, 5, AND 9 AMENDMENTS BASED ON 5-14-2018 TELECON

Following the clarification call with the NRC on 5/14/2018, the staff issued an e-mail from Brian Benney (NRC) to Danielle Page Blair dated May 16, 2018, 9:46 a.m. The attachment to the e-mail is provided below. RAI 3 was not used by the NRC.

RAI-1

Section 4.1 of the BAW-2192, Rev. 0, Supplement 1 (the TR) states, "Consistent with BAW-2192PA, Revision 0 (Ref. 3), this J-R model is used for Linde 80 welds and Rotterdam welds." BAW-2192PA did not mention Rotterdam welds. Provide justification that the J-integral resistance (J-R) model based on Linde 80 welds is applicable to Rotterdam welds. This justification should consider other J-R models which has been used for RPV welds.

Revision:

Deletion.

RAI-5

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the maximum error by using a flat plate solution instead of a cylindrical plate solution?

Revision:

Section 5.3.2 of the TR (p. 5-8, 3rd paragraph) states that the stress intensity factor is conservatively calculated for the RV nozzle-to shell weld using a flat plate solution by Newman and Raju. What is the R/t ratio for this weld? What is the percent difference in the stress intensity factor by using a flat plate solution instead of a cylindrical plate solution?

RAI-9

To gain confidence in the cladding stress evaluation methodology presented in Section 5.2.3 of the Supplement, please provide for the axial flaw in a clad RPV, the actual

stress distribution and the cladding stress distribution and their associated stress intensity factors. For the axial flaw in the same RPV without cladding, please provide the stress distribution and its corresponding stress intensity factor.

Summary of Changes

No technical changes were made to the TR by the RAI process.

The following editorial change was made to the original BAW-2192P, Supplement 1, Revision 0 submittal:

- Table 3-1, Page 3-4, Superscript 3 associated with Surry Unit 2 Nozzle Shell (NS) to Outlet Nozzle Forging Welds, Rotterdam, was deleted. There was no associated footnote text and there is no technical change to the document that results from this editorial change.

The following editorial change was made to the RAI response BAW-2192P-00-Supplement 1-00-000_Q1P, Revision 0:

- Page 1-1, first paragraph, the date of May 7, 2014 was corrected to May 7, 2018. There is no technical change to the document that results from this editorial change.

The following editorial change was made to the RAI response BAW-2192P-00-Supplement 1-00-000_Q2P, Revision 0:

- Page 1-1, first paragraph, the date of May 7, 2014 was corrected to May 7, 2018. There is no technical change to the document that results from this editorial change.

In addition, the comment and resolution responses regarding BAW-2178 (from ADAMS ML18333A131) have been removed from this document since they are not applicable to this TR.