FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO PRESSURIZED WATER REACTOR OWNERS' GROUP

LICENSING TOPICAL REPORT BAW-2192, SUPPLEMENT 1NP, 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

Contents

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO PRESSURIZED WATER REACTOR OWNERS' GROUP

LICENSING TOPICAL REPORT BAW-2192, SUPPLEMENT 1NP, 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 $1.0x10^{17}$ 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 <u>CONCLUSIONS</u>

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).

Attachment: Comment Resolution Table

Principal Contributor: Jeffrey Poehler, NRR/DMLR/MVIB

Date: April 29, 2019