FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO PRESSURIZED WATER REACTOR OWNERS' GROUP

LICENSING TOPICAL REPORT BAW-2178, SUPPLEMENT 1NP, 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"

PROJECT NO. 694

Enclosure 2

Contents

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO PRESSURIZED WATER REACTOR OWNERS' GROUP

LICENSING TOPICAL REPORT BAW-2178, SUPPLEMENT 1NP, 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"

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-2178, Supplement 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" (Ref. 4, the topical report (TR)) for review and approval by the Nuclear Regulatory Commission (NRC).

The TR is a supplement to BAW-2178PA, "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. 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-2178 by a safety evaluation (SE) dated March 29, 1994 (Refs. 6 and 7) which is included in BAW-2178PA.

2.0 REGULATORY ANALYSIS

The regulation at 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 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 American Society of Mechanical Engineers (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 Generic Aging Lessons Learned Report for Subsequent License Renewal (GALL-SLR), NUREG-2191, provides guidelines for review of the USE TLAA in Section 4.2.2.1.2.

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 (referred to as Appendix K for simplicity), 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 Summary of the TR Content

The TR covers the following major topics:

Section 1.0 of the TR discusses the scope and purpose of the TR, and also identifies the current licensing basis of the in-scope plants, or EMA analysis of record.

Section 2.0 of the TR addresses regulatory requirements and ASME Code requirements. 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.

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 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. Section 3.0 also states that 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 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, Surry, and TP reactor vessel fabrication reports.

TR 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-2178PA. Section 4.0 refers to Appendix A of BAW-2192, Supplement 1, Rev. 0, 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. The TR indicates the methodology and acceptance criteria are consistent with Appendix K.

The results are reported for each plant in TR Section 5.3, for the RPV shell welds and the RPV transition welds and 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 C and D loadings. The conclusions for the limiting weld are detailed for each plant, including the ratio of $J_{0.1}$ to J_1.

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 Appendix K, which is an NRC approved method.

3.2.1 Applicability of the Original Model to 80 Years

Details of the staff's review of the justification for applicability of the original J-R model to the EMA for 80 years are contained in the NRC staff's SE for BAW-2192, Supplement 1, Rev. 0 (Ref. 8). The NRC staff found that the PWROG has demonstrated the applicability of the original Model 4B for 80 years of operation.

3.2.2 Consistency with Appendix K

Summary of the Appendix K methodology:

Appendix K provides acceptance criteria and evaluation procedures for determining acceptability for operation of a RPV when the vessel metal temperature is in the upper shelf range. The methodology is based on the principles of elastic-plastic fracture mechanics.

For Service Level C and D conditions, for weld materials, an interior semi-elliptical flaw with an a/t ratio of 0.1 or less is postulated, with the flaw's major axis oriented along the weld of concern, and the flaw plane oriented in the radial direction. For Service Level C, Appendix K, Paragraph K-2300 specifies the following acceptance criteria:

a) The applied J-integral shall be less than the J-integral of the material at a ductile flaw extension of 0.10 in. (2.5 mm), using a structural factor of 1 on loading.

b) Flaw extensions shall be ductile and stable, using a structural factor of 1 on loading.

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

For Service Level D, Appendix K, paragraph K-2400 specifies the following acceptance criteria:

a) When evaluating adequacy of the upper shelf toughness for Level D Service Loadings, flaws as specified for Level C Service Loadings in K-2300 shall be postulated, and toughness properties for the corresponding orientation shall be used. Flaws of various depths, ranging up to the maximum postulated depth, shall be analyzed to determine the most limiting flaw depth. Smaller maximum flaw sizes may be used when justified. Flaw extensions shall be ductile and stable, using a structural factor of 1 on loading.

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

c) The total flaw depth after stable flaw extension shall be less than or equal to 75% of the vessel wall thickness, and the remaining ligament shall not be subject to tensile instability.

3.2.3 Methodology of Determination of Applied Loads

3.2.3.1 Selection of Level C and D Transients

Regulatory Guide 1.161 indicates that for selection of Level C and D transients, either plant-specific transients or a generic bounding transient may be used. The TR considers plant-specific transients.

For a plant-specific approach, RG 1.161 states that the Service Level C and D design transients and events that are necessary to demonstrate compliance with Standard Review Plan 3.9.3 Rev. 1 should be used.

TR Section 4.3 identified the Level C and D transients for each plant. Table 1 below compares the transients evaluated in the TR versus those originally evaluated in BAW-2178PA, Rev. 0.

Table 1 - Comparison of Transients in TR vs. Previous EMA Analyses

Plant	Level	BAW-2178, Rev. 0 Suppl. 1	60-year EMA	BAW-2178PA, Rev. 0
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)	**[ ]**	**[ ]**

Table 1 also includes the information on the transients considered in the 60-year EMAs, which was provided in response to RAI 7. Since the transients are not consistent in the licensee's current and previous EMA evaluations, the staff issued RAI 7 for clarification.

In its June 15, 2018, response to RAI 7a (Ref. 2), the PWROG indicated the 40-year EMA used the limiting TP SLB combined with the Zion RPV geometry to bound all the B&W RPVs within scope of BAW-2178PA, Rev. 0. For the 60-year EMA, the response indicates ONS used a separate analysis in BAW-2251A, which reevaluated all the transients for 60 years except the RDP, which was determined in BAW-2178PA, Rev. 0 to have no safety impact. However, to ensure the most limiting Service Level C & D transients were chosen, **[ ]**

The response indicates plant-specific 60-year EMAs were developed for Surry and TP.

With respect to the 80 year EMA, the response indicates that for ONS, all the Level C and D transients were reevaluated **[ ]** For Surry, the limiting Level C and D transient (SLB) was evaluated for 80 years as reported in the TR, Section 4.3.2. The SLB case evaluated in the 60-year EMA was replaced with a revised SLB case and supplemented with an SLB case from the Surry ASME Section III Design Specification.

For TP, the response indicates that the limiting Level C and D transients (SLB) were evaluated for 80 years as reported in the TR, Section 4.3.3. The SLB case evaluated in the 60-year EMA (identical to the SLB used in BAW-2178PA, Rev. 0 for 40-years), was supplemented with an SLB case from the Surry ASME Section III Design Specification; the TP ASME Section III Design Specification did not include any level C and D transients.

The NRC staff finds the response to RAI 7a acceptable because it explains the differences in the limiting transients chosen in the TR versus the limiting transients in the 40-year EMA. Some of the differences are due to the approach of selecting a bounding transient for all the RPVs in BAW-2178PA, Rev. 0, versus the use of plant-specific limiting transients in the 80-year EMA. The NRC staff finds that based on the response, the transients for ONS evaluated in the TR include all those evaluated in BAW-2178PA, Rev. 0 except RDP, plus some additional transients. For TP, the TR evaluates the same Turkey Point SLB with Loss of Offsite Power as was evaluated in BAW-2178PA, Rev. 0, plus the additional SSDC 1.3 SLB. **[** **]** The Zion SLB has been dropped since Zion is not within scope of the TR.

The PWROG also provided the source/reference of all the limiting Level C and D transients. The majority of the transients were from the ASME Section III Reactor Vessel Design Specifications for the individual plants, with some exceptions.

Regulatory Guide 1.161, Section 4.1 states that to provide reasonable assurance that the limiting service loading conditions have been identified on a plant specific basis, the Service Level C and D design transients and events that are necessary to demonstrate compliance with Standard Review Plan 3.9.3 should be used. SRP 3.9.3, Appendix A states that Code Class 1, 2, and 3 components shall meet Level C loading under sustained loads and the design basis pipe break (DBPB), and that for Level D loading, sustained loads plus either 1) the DBPB or 2) main steam/feedwater pipe break (MS/FWPB), or LOCA, plus SSE should be considered. Therefore, since the limiting transients used in the TR are consistent with the above guidance, reasonable assurance has been demonstrated that the most limiting Level C and D transients have been selected for the EMA reported in the TR.

The NRC staff finds that the PWROG has identified the limiting transients for each plant from appropriate plant-specific sources and has also evaluated other non-plant-specific transients for conservatism, such as the use of the Surry SSDC 1.3 SLB for TP.

Lastly, the PWROG stated that the difference in results between Surry and TP for the same transient is due to differences in flaw orientation and the residual stress. The Surry limiting weld is a longitudinal weld, while the limiting weld for TP is a circumferential weld. In addition, the Surry longitudinal weld has **[** **]**

Based on the above, RAI 7 is resolved.

3.2.3.2 <u>Level C and D Service Loadings</u>

Applied K_I for each transient for each plant for the RPV shell welds was determined using the PCRIT computer code, which includes a one-dimensional shell thermal model, closed-form expressions for through-wall thermal and pressure stresses, and closed-form applied K_I models.

To determine the K_I for the nozzle-to-shell welds and transition welds, PCRIT was used for ONS-1, -2, and -3, and three-dimensional finite element analyses were used for Surry and TP. TR Section 5.2.3 states that **[ ]**

To gain confidence in the cladding stress evaluation methodology, in RAI 9, the staff requested that the PWROG provide the actual stress distribution and the cladding stress distribution and their associated applied K_I values for the axial flaw in the RPV with and without cladding. In its June 29, 2018, response (Ref. 3), the PWROG provided, for an axial flaw in the Surry limiting axial weld at the limiting time in the SLB transient, a table showing the nodal coordinates and temperature distribution, along with the calculated thermal hoop stresses and applied K_I values with and without cladding, and the applied K_I for cladding effects, K_{Iclad}. The NRC staff noted that the cladding-induced stress distribution through the cladded RPV thickness (the stress in the table with cladding minus the stress without cladding) is very similar to the cladding-induced stress distribution in NUREG/CR-4486, "VISA-II-A A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure," March 1986. The difference in the cladding-induced stresses in the base metal between the RAI response and the NUREG is minor because the cladding-induced base metal stresses are about one tenth the magnitude of the cladding stresses. Based on the NUREG K_{Iclad} values and the maximum cladding stress provided by the RAI response, the NRC staff found that the K_{Iclad} for a/t = 0.1 is 16.89 ksi√inch, about 25 percent more than the corresponding value in the RAI response. This verification is acceptable, considering that: (1) the NUREG stress model is analytical, applying many simplifications, while the TR is finite element analysis, applying fewer simplifications; (2) the NUREG fracture mechanics model is two dimensional, while the TR is three dimensional; and (3) the NUREG stress and fracture mechanics models are generic, while the TR is plant-specific. These three factors would remove the conservatism in the NUREG approach and have thus verified the TR K_{Iclad} values. RAI-9 is resolved.

TR 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 Appendix K do not consider residual stresses, in RAI 10 the NRC staff requested the PWROG to confirm that the assumed residual stress distribution is conservative such that it would have resulted in a positive contribution to total K_I. In its June 15, 2018, response to RAI 10 (Ref. 2), the PWROG confirmed that the residual stresses determined in the PCRIT analysis are either tensile, resulting in a positive K_I, or were compressive, **[ ]** The response also provided details for each specific weld type for which PCRIT was used. The NRC staff finds the response to RAI 10 acceptable since it confirms that only positive contributions of residual stress to K_I were considered.

The most limiting transient for each plant was determined by plotting K_I for the design basis transients for each plant versus the K_{JC} and K_{IC} as a function of temperature. The intersection of the K_{IC} and K_{JC} curves defines the start of the upper shelf region. Since an EMA analysis is an analysis of ductile tearing in the upper shelf region, the only purpose for plotting K_{IC} is to define the beginning of the upper shelf temperature range.

The TR states that the maximum 1/10T adjusted reference temperature (ART) for the ONS nozzle-to-shell welds is **[ ]**°F. For the Surry and TP limiting nozzle-to-shell welds, the TR uses a 1/10T ART of **[ ]** for both plants. This value did not match the value calculated by the staff from the material data in the submittals of record referenced for these plants. However, the value of **[ ]** is higher and therefore more conservative than the values calculated by the staff. Since the NRC staff could not independently confirm the two ART values discussed above, in RAI 8 the staff requested the PWROG to identify the material heat corresponding to the two ART values, and provide the $RT_{NDT(u)}$, σ_Δ and σ_I, chemistry factor (CF), fluence, and Cu and Ni content for these two, weld heats, plus the source/reference of this data. In its June 15, 2018, response to RAI 8 (Ref. 2), the PWROG provided the requested data. The PWROG provided the 1/10T fluence values. The material heats are 82102 and 8T1554B, respectively. Using the fluence, $RT_{NDT(u)}$, σ_Δ and σ_I, CF, and Cu and Ni values provided, the staff calculated the corresponding ART values, obtaining similar values to the PWROG values. The PWROG stated the $RT_{NDT(u)}$, and σ_I are generic values developed per the guidance of RG 1.99, Rev. 2, and the σ_Δ was determined per the guidance of RG 1.99, Rev. 2. The NRC staff notes that RG 1.99, Rev. 2 states that σ_I is the standard deviation of the set of values used to establish the mean if generic mean values of a class of material is used. RG 1.99, Rev. 2 states that σ_Δ is 28 °F for welds, except that σ_Δ need not exceed 0.50 times ΔRT_{NDT}. The staff finds the response to RAI 8 to be acceptable.

Upper shelf toughness K_{JC} is derived from the J-integral resistance model for a flaw depth of 1/10th the wall thickness, a crack extension of 0.10 inch, and the applicable fluence value at the crack tip:

$$K_{JC} = \sqrt{J_{0.1}E/(1000(1-v^2)}$$

where:

K_{JC} = upper-shelf region toughness, ksi√in
$J_{0.1}$ = J-integral resistance at Δa = 0.1 in.
E = elastic modulus
N = Poisson's Ratio

The transient with the smallest margin to the K_{JC} curve in the upper shelf region was determined to be the limiting transient. The temperature where the smallest margin occurs was used to define the point in the transient at which the evaluation of flaw extension and flaw stability was performed.

The K_I for the limiting transient were then converted to the applied J-integral (J_{app}) for the purposes of the flaw extension and flaw stability analyses, using the equation:

$$J_{app}(a) = 1000K_{Itotal}^2(a)(1-v^2)/E$$

Where a is the crack depth, K_{Itotal} is the total K_I, v is Poisson's Ratio, and E is the elastic modulus. The J_{app} curve as a function of a was then plotted with the J-R curves for the crack extension and crack stability analyses.

TR Figure 5-1 shows K_I as a function of temperature for the ONS-1, -2, and -3 RPV shell welds for all five transients versus the K_{IC} and K_{JC} as a function of temperature. The K_I for the HL-LOCA transient approaches the closest to the lower bound K_{JC}, thus was determined to be the limiting transient. TR Figure 5-7 shows K_I as a function of temperature for the RPV nozzle-to-shell weld for the HL-LOCA transient only versus the K_{IC} and K_{JC} as a function of temperature. The K_I for the HL-LOCA transient most closely approaches the lower bound K_{JC} at

5.5 minutes into the transient, which was determined to be the limiting evaluation point for the RPV nozzle-to-shell welds.

TR Figure 5-3 and 5-5 show the K_I as a function of temperature for the two transient cases versus the K_{IC} and K_{JC} for the limiting RPV shell welds for Surry and TP as a function of temperature. For both plants, the SSDC SLB is seen to be the most limiting transient since it most closely approaches the lower bound K_{JC}. TR Figures 5-9 and 5-11 are similar plots for the RPV outlet nozzle-to-shell welds for Surry and TP, and show the SSDC SLB is the most limiting transient for the outlet nozzle-to-shell welds.

For all the plants, the loadings from the limiting Service Level D transient were used to evaluate the Appendix K acceptance criteria for both the Service Level C and Service Level D.

3.2.4 Selection of Limiting Welds for Analysis

The TR reported the analysis for Service Level C and D for the limiting RPV shell, nozzle-to-RPV shell, and transition weld for each plant. The identification of the most limiting weld of each type was based on the most limiting weld identified in the Service Level A and B EMA in BAW-2192, Supplement 1 Rev. 0. The NRC staff finds this approach acceptable because the relative stresses for each weld of each type should be the same regardless of the magnitude of the stresses, and the relative J-R toughness should also be similar (e.g., those with the lowest toughness under Service Level A and B conditions should also have the lowest toughness under Service Level C and D conditions.)

For ONS, the TR did not include an analysis of the upper transition weld SA-1135 because (the TR stated) this weld was bounded by the limiting RPV shell weld for ONS. The staff verified that this weld was bounded by the results for the limiting RPV shell weld for ONS, ONS-1 weld SA-1073, in the analysis for Service Level A and B in BAW-2192, Supplement 1 Rev. 0.

3.2.5 Inputs

The same material chemistry, mechanical properties, and inner diameter neutron fluence values were used in both the Service Level A and B EMA and the Service Level C and D EMA. The staff's SE of BAW-2192, Supplement 1, Rev. 0 (Ref. 8) describes the staff's review of the material chemistry and mechanical property inputs.

The TR states that the fluence projections are reported for all reactor vessel weld locations that are expected to exceed 1×10^{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 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. The NRC staff notes that guidance on determination of neutron fluence for TLAA is contained in Section 4.2 of NUREG-2192.

3.2.6 Confirmatory Calculations

The NRC staff performed confirmatory calculations using the methodology of Appendix K, and the PWROG model for fracture toughness, for the limiting RPV shell welds, nozzle-to-RPV welds, and transition welds for each plant. 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. For the RPV shell welds, the staff calculated the margin with respect to flaw extension of $J_{0.1}/J_1$ with a structural factor of 1 on pressure, and found that the margin is acceptable (> 1.0), using the J-R curves from both the PWROG model and the NUREG/CR/5729 model.

In addition, for the RPV shell welds, the staff checked the flaw stability using the crack driving force diagram in accordance with K-4310, with a structural factor of 1.0 on pressure. The lower bound J-R curve was used for the Level C service loadings, and the mean J-R curve was used for the Level D service loadings, as required by Appendix K. The results showed the slope of the J_I curve to be less than the slope of the J-R curve at the point of intersection, using either the PWROG J-R model or the NUREG/CR-5729 model, therefore meeting the acceptance criteria of Appendix K.

The staff also calculated the stable flaw extension and remaining ligament stability for the Service Level D conditions. The staff's results were in good agreement with those in the TR.

4.0 CONCLUSIONS

The NRC staff concludes that BAW-2178, Supplement 1, Rev. 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 C and D loads, through 80 calendar years of operation. The 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, Rev. 0, Supplement 1 and BAW-2178-P, Rev. 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 BAW-2192P, Supplement 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," - RAIs 7, 8, and 10 Associated with BAW-2178P Supplement 1, Rev. 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, Revsion 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-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).

5. Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C&D Service Loads," April 30, 1994 (ADAMS Legacy Accession No. 9406290288).

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. 9404250026).

7. Safety Evaluation by the Office of Nuclear Reactor Regulation B&W Owners' Group Topical Report BAW-2178P on Upper-Shelf Energy Equivalent Margin Analysis Materials and Chemical Engineering Branch Division of Engineering, March 29, 1994 (Adams Legacy Accession No. 9404250171).

8. Final Safety Evaluation by the Office of Nuclear Reactor Regulation related to Pressurized Water Reactor Owners' Group Licensing Topical Report 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 Levels C&D Service Loads", December 7, 2018 (ADAMS Accession No. ML18333A129).

Attachment: Comment Resolution Table

Principal Contributor: Jeffrey Poehler, NRR/DMLR/MVIB

Date: April 29, 2019