PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG-19047-NP-A
Revision 0

WESTINGHOUSE NON-PROPRIETARY CLASS 3
FRAMATOME INC. NON-PROPRIETARY

North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis

Materials Committee

PA- MSC-1481, R3

September, 2021

PWROG-19047-NP-A
Revision 0

North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis

PA- MSC-1481, R3

Gordon Z. Hall *
Structural Design and Analysis

September, 2021

Reviewer: James Andrachek*
PWR Owners Group PMO

Reviewer: Ashok D. Nana
Framatome, Component Analysis and Fracture Mechanics

Approved: Stephen P. Rigby*, Manager
Structural Design and Analysis

Approved: Jim Molkenthin*, Program Director
PWR Owners Group PMO

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

Framatome Inc.
3315 Old Forest Road
Lynchburg, VA 24501

NON-PROPRIETARY INFORMATION

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR THE PRESSURIZED WATER REACTOR OWNERS GROUP

TOPICAL REPORT PWROG-19047-P/NP, REVISION 0,

"NORTH ANNA UNITS 1 AND 2 REACTOR VESSELS LOW UPPER-SHELF FRACTURE

TOUGHNESS EQUIVALENT MARGIN ANALYSIS"

EPID: L-2020-TOP-0028

NON-PROPRIETARY INFORMATION

TABLE OF CONTENTS

1.0 INTRODUCTION

By letter dated May 27, 2020 (Ref. 1), as supplemented by letter dated November 19, 2020 (Ref. 2), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) PWROG-19047-P/NP, Revision (Rev.) 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (Ref. 3), to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

The purpose of this TR is to document the equivalent margins analysis (EMA) for the North Anna Power Station, Units 1 and 2 (North Anna) reactor vessel (RV) inlet and outlet nozzle welds, nozzle forgings, and nozzle belt forgings (a.k.a., upper shell forgings). These locations were chosen for their upper-shelf energy (USE) potentially falling below the 50 foot- pound (ft-lb) limit at 80-years (72 effective full power year (EFPY)) for subsequent license renewal (SLR). In accordance with the requirements of Appendix G, "Fracture Toughness Requirements," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 as listed in Paragraph IV.A.1.a, the RV beltline materials must maintain Charpy USE 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 NRC, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G to Section XI of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code) (Ref. 4).

The EMA documented in the TR for the nozzle and upper shell forgings uses the multivariable model for RV base metal reported in NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data" (Ref. 5). Although the nozzle-to-shell welds are projected to have USE greater than 50 ft-lbs at 72 EFPY, they are evaluated proactively in this EMA for asset management consideration. The EMA documented in the TR for the Rotterdamsche Droogdok Maatschappij (Rotterdam) welds utilizes the Babcock & Wilcox Owners Group (B&WOG) J-integral resistance (J-R) Model 6B reported in Appendix A of BAW-2192, Revision 0, Supplement 1P/NP-A, 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" (Ref. 6). The justification for using the B&WOG Model 6B for the North Anna Rotterdam welds is addressed in BAW-2192, Supplement 2P/NP-A, 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" (Ref. 7).

2.0 REGULATORY EVALUATION

The regulations in Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The NRC staff reviews licensee evaluations to ensure that they provide a realistic or conservative assessment of the RPV such that it can be demonstrated that the licensee complies with these requirements.

Specifically, 10 CFR Part 50, Appendix G, Paragraph IV.A.1.a requires RV 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 (i.e., EMA) that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G to

Section XI of the ASME Code. 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 RV surveillance capsule data that are reported through implementation of a plant's RV 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" (Ref. 8) 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" (Ref. 9) provides an acceptable methodology for the performance of analyses intended to meet the provisions for additional analysis specified in Appendix G to 10 CFR Part 50 Paragraph IV.A.1.a.

In accordance with 10 CFR 54.3, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," under the current licensing basis (CLB) USE is considered a time-limited aging analysis (TLAA). 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.

RG 1.161 provides guidance for acceptable methods of evaluating low USE, and states that the analytical methods described in Appendix K of Section XI of the ASME Code (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 detailed information on the selection of transients and gives very little details 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 RV materials, including Linde 80 welds.

The NRC staff considered 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," when evaluating the TR with respect to General Design Criteria (GDC). The NRC staff review of the fluence assumptions was performed in consideration with the requirements contained in GDC 14, "Reactor Coolant Pressure Boundary;" GDC 30, "Quality of Reactor Coolant Pressure Boundary;" and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary." These GDCs require the design, fabrication, and maintenance of the reactor coolant pressure boundary with adequate margin to assure that the probability of rapidly propagating failure of the boundary is minimized. In particular, GDC 31 explicitly requires consideration of the effects of irradiation on material properties.

RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (Ref. 10), provides guidance on methods for determining reactor pressure vessel fluence that are acceptable to the NRC staff, based on the requirements identified above.

3.0 TECHNICAL EVALUATION

This technical evaluation section documents the NRC staff's evaluation of the TR against the relevant criteria identified in Section 2.0 above.

3.1 Overview of PWROG-19047-P/NP

The TR covers the following major topics:

Section 1.0 discusses the scope and purpose of the TR. Specifically, the TR documents the EMA for the North Anna RV inlet and outlet nozzle Rotterdam welds, nozzle forgings, and nozzle belt forgings. These locations were chosen for USE potentially falling below the 50 ft-lb limit at 80-years (72 EFPY) for SLR.

Section 2.0 of the TR addresses regulatory requirements and ASME Code requirements. Specifically, Appendix G to 10 CFR Part 50 Paragraph IV.A.1 specifies the RV Charpy USE requirements. TR Section 2.2 states the EMAs are performed in accordance with Appendix K of the 2013 Edition of Section XI of the ASME Code (i.e., an edition incorporated by reference in 10 CFR 50.55a and Appendix G), and confirms that the material properties used in this analysis are based on the original RV construction code, ASME *Boiler and Pressure Vessel Code*, Section III, 1968 Edition, with Addenda up to and including the Winter of 1968.

Section 3.0 of the TR provides the inputs for the EMAs and a description of the RVs for North Anna. Specifically, the finite element stress model and analysis; and the J-R models used to address the RV inlet and outlet nozzle Rotterdam welds, nozzle forgings, and nozzle belt forgings were described.

Section 4.0 of the TR describes the fracture mechanics evaluation method used, which is in accordance with Appendix K. Section 4.0 also provides the detailed results of the fracture mechanics evaluation, which included two analyses per component: an evaluation of flaw extension and an evaluation of flaw stability.

Section 5.0 of the TR also contains the summary and conclusions. The PWROG concluded that all components assessed by the EMAs for North Anna met the acceptance criteria of Appendix K for Service Level A and B loadings, and Service Level D loadings. The applicant further concluded that the EMAs in this TR demonstrate that the North Anna RV nozzle-to-shell welds, nozzle forgings, and upper shell forgings were evaluated for equivalent margins of safety per ASME Code Section XI, and that the flaw extension and stability criteria of Appendix K were satisfied.

3.2 Staff Evaluation

The NRC staff's review focused on:

Verification that the EMAs for North Anna were performed in accordance with the Appendix K to Section XI of the ASME Code

Verification that the EMA results for North Anna satisfy the acceptance criteria in Appendix K to Section XI of the ASME Code

3.2.1 Inputs

3.2.1.1 Material chemistry, RT_{NDT}, and Upper-Shelf Energy Value

Section 3.2 of the TR states that "the maximum reported copper content and the fluence value at the nozzle-to-shell weld is utilized" as summarized in Table 3-3 of the TR. In its RAI responses dated November 19, 2020, the PWROG explained that the copper content value is the generic value in RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" (Ref. 11), because the weld heat and type for these nozzle-to-shell welds could not be determined, only that these welds were fabricated by Rotterdam. Consistent with the PWROG-17090-NP-A, "Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination" (Ref. 12), when the weld heat and type are not known for a Rotterdam fabricated weld, the generic values of 0.35% and 1.13% from RG 1.99, Rev. 2 can be used for copper content and nickel content, respectively. Thus, the staff finds the use of 0.35% and 1.13% for copper content and nickel content in the TR to be acceptable. Furthermore, the PWROG clarified that the RT_{NDT} value of 208.3°F is applicable to the inlet nozzle to upper shell welds and outlet nozzle to upper shell welds and discussed how it calculated the Adjusted Reference Temperature (ART). Based on its review, the staff confirmed that the $RT_{NDT(u)}$ values, and material chemistry values (i.e., copper and nickel) for these components are consistent with the CLB for North Anna and PWROG-17090-NP-A, respectively. Furthermore, the staff confirmed that the ART value of 208.3°F used in the TR was determined to be consistent with the RG 1.99, Rev 2.

Section 3.2 of the TR provides the basic form of J-R as expressed in NUREG/CR-5729, and RG 1.161. Furthermore, the TR provides the different parameters, variables and assumed values used in the EMAs for North Anna. In its RAI responses dated November 19, 2020, the PWROG explained that the 80-year USE values for North Anna are documented in WCAP-18364-NP, Rev. 1, "North Anna Units 1 and 2 Time-Limited Aging Analysis on Reactor Vessel Integrity for Subsequent License Renewal (SLR)" (Ref 13), and also provided the Charpy values used in the TR for the nozzle and intermediate forgings. The PWROG also confirmed in its supplement that the input parameters represent weak orientation material properties in all cases, which the staff finds to be a conservative approach. The staff reviewed the 80-year USE values for the North Anna RV inlet and outlet nozzle-to-shell welds, nozzle forgings, and intermediate shell forging, in WCAP-18364-NP for the purposes of verifying that the reference of these values were appropriate in the TR. Based on its review, the staff finds the 80-year USE values for the North Anna RV inlet and outlet nozzle-to-shell welds, nozzle forgings, and intermediate shell forging were calculated consistent with applicable regulations in Appendix G to 10 CFR Part 50 and guidance in RG 1.99, Rev. 2. Additionally, the staff confirmed that the Charpy values used in the TR to calculate J-R curves for the nozzle and intermediate forgings were conservative compared to the 80-year USE values for the North Anna, nozzle forgings and intermediate shell forging.

3.2.1.2 Neutron Fluence Values

Section 3.2 of the TR indicates that the neutron fluence value at the nozzle-to-shell weld is summarized in Table 3-3 of the TR. In its RAI responses dated November 19, 2020, the PWROG clarified that revised fluence values were generated and documented in WCAP-18015-NP, Rev. 2 (Ref. 14). This document was submitted in support of the NRC staff review of a change to the North Anna, Units 1 and 2 surveillance capsule withdrawal schedule (the staff's assessment is documented in Ref. 15). The methods used to calculate the fluence are consistent with those described in Chapter 2.2 of WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" (Ref. 16). In the surveillance capsule withdrawal schedule review, the NRC staff determined that the fluence estimates contained therein were acceptable, because the methods used to calculate the fluence were adherent to RG 1.190, and thus consistent with GDC 14, 30, and 31. This conclusion applies to the fluence estimates, insofar as they support the additional evaluations provided in the TR.

In the safety evaluation referenced above, the NRC staff noted that the application of the fluence methods to RV components that are significantly above or below the core, i.e., nozzle elevations, would require more detailed evaluation. In that review, the NRC staff considered various contributors to the generic uncertainty estimate provided in WCAP-14040-A that would likely increase as a result of the increased transport distance to the nozzle elevations, and determined that significant increases could be accommodated and the overall uncertainty could still remain within the ±20-percent criterion specified in RG 1.190.

In its RAI responses dated November 19, 2020, the PWROG provided additional justification for the treatment of the fluence values assumed for the North Anna, Units 1 and 2 RV inlet and outlet nozzles. Among other things, the licensee stated:

- The maximum projected fluence values of the North Anna RV inlet/outlet nozzles… could increase by 60% before exceeding the fluence value utilized in the [equivalent margins analysis (EMA) for subsequent license renewal].
- …the limiting [upper shelf energy] nozzle for both Units 1 and 2… would need to increase by a factor of greater than 4 prior to reaching the fluence utilized in the EMA.
- …the maximum projected fluence values of the North Anna inlet and outlet nozzle-to-shell welds would need to increase by 3 times the current value before exceeding the fluence value utilized in the EMA for SLR.

The additional information indicates that the nozzle fluence values assumed in the EMA bound, with substantial margin, the fluence estimates that were determined using the discrete ordinates transport methods to support the SLR. This margin provides assurance that, even if the uncertainty associated with the nozzle fluence estimates were to exceed ±20-percent by a modest amount, the EMA would remain bounding for those components. Based on this consideration and on the previous staff determination considering the acceptability of the fluence values documented in WCAP-18015-NP, the NRC staff determined that the fluence assumptions in the TR are acceptable.

3.2.1.3 Mechanical Properties

Section 2.2 of the TR indicates that the material properties used in the EMAs are based on the original RV construction code, ASME *Boiler and Pressure Vessel Code*, Section III, 1968 Edition, with Addenda up to and including the Winter of 1968. Specifically, Section 3.1 of the TR provides the material properties of SA-508 Class 2 used for the RV shell and nozzle forging base metal, and the material properties for Type 304 stainless steel used for the cladding. The PWROG indicated that the nozzle-to-shell weld mechanical properties were assumed to be identical to the forging material for the purpose of stress analysis. The staff noted that this assumption is typical of finite element analysis (FEA) and is reasonable because the objective of the FEA in the TR was not to determine residual stress due to the welding process of the nozzle-to-shell welds. The staff reviewed Updated Final Safety Analysis Report (UFSAR) (Ref. 17) Section 5.4.3.6 and UFSAR Tables 5.2-3, 5.2-22, 5.2-26, and 5.2-27 and confirmed that the base metal and cladding materials are consistent with those addressed in the TR; thus, the staff finds the material properties used in the TR are appropriate for North Anna.

3.2.2 Applicability of the J-Integral Resistance Models

K-3000 of Appendix K does not contain a specific model for determining the J-R material curve. Instead, K-3300 of Appendix K 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. The staff's review of the applicability of the J-R model used for the RV nozzle-to-shell welds and RV nozzle and upper shell forging is documented below.

RV Nozzle-to-Shell Welds

Section 3.2 of the TR states that the nozzle-to-shell welds were fabricated by Rotterdamsche Droogdok Maatschappij and that the nozzle weld EMA utilizes the B&WOG J-R model 6B reported in BAW-2192, Rev. 0, Supplement 1P-A, Rev. 0, Appendix A. The PWROG explained that the justification for the use of Model 6B for the North Anna nozzle-to-shell welds is provided in BAW-2192, Rev. 0, Supplement 2P-A, Rev. 0.

Based on its review of BAW-2192, Rev. 0, Supplement 2P-A, Rev. 0, including the data set used to develop the Model 6B curve and the new Rotterdam J-R data in the report, the staff determined that the PWROG J-R Model 6B: (a) bounds most of the original and new data used in its development, and (b) bounds all of the new Rotterdam J-R data provided in the report (see Ref. 5). Thus, the staff finds the use of the Model 6B curve acceptable for use in the 80-year EMAs for the North Anna Rotterdam weld material within the limits of the B&WOG Model 6B explanatory variables. The staff noted the application of the Model 6B curve for the 80-year EMA for the North Anna Rotterdam weld material is dependent, in part, on the acceptability and consistency with the plant's CLB of the material chemistry values (i.e., Cu and Ni values) used in the EMA for each RV weld material.

The staff's review of the acceptability and consistency with the plant's CLB of material chemistry values used in the TR for nozzle-to-shell welds for North Anna, is documented in Section 3.2.1.1 of this SE. Based on the staff's review, the staff noted the material chemistry values used in the TR for nozzle-to-shell welds are within the B&WOG Model 6B explanatory variables for copper content; thus, the staff finds that for the inlet and outlet nozzle-to-shell welds at North Anna

(a) the use of the Model 6B curve to be appropriate and acceptable, and (b) K-3300 of Appendix K has been appropriately addressed.

RV Nozzle and Shell Forgings

Section 3.2 of the TR states that the J-R curves for the RV nozzle and shell forgings were developed in accordance with Charpy model without fluence in Table 11 of NUREG/CR-5729. The PWROG explained that this Charpy model is the same model described in Section 3.3 of RG 1.161 and is applicable to the North Anna nozzle and shell forgings because they are bounded by the range of explanatory variables (fluence, copper content, etc.) used to develop the J-R model.

In its RAI responses dated November 19, 2020, the PWROG clarified that the Charpy model without fluence in Table 11 of NUREG/CR-5729 is applicable to the North Anna nozzle and shell forgings since this J-R model was validated against representative Rotterdam forgings with measured J-R data. Furthermore, in its supplement, the PWROG provided a comparison of measured test data from Rotterdam forgings to the Charpy model in Table 11 of NUREG/CR-5729 used in the TR. The staff noted that the Charpy model used in the TR was representative of the measured test data from these Rotterdam forgings. Based on its review, the staff finds (1) K-3300 of Appendix K has been addressed and (2) the use of the Charpy model without fluence in Table 11 of NUREG/CR-5729 for the nozzle and shell forgings at North Anna to be appropriate because the material fabrication of these components are applicable to the database of materials used to develop this J-R model.

3.2.3 Consistency with Appendix K to Section XI of the ASME Code

Nozzle-to-Shell Welds and Upper Shell Forgings

Section 4.1.1 of the TR explains that for axial or circumferential flaws the stress intensity factor (SIF) due to radial thermal gradients can be calculated per K-4210(c) of Appendix K. However, since the thermal stresses are based on a FEA, the PWROG explained that the procedure in A-3320 of 2013 Edition of Section XI of the ASME Code were modified and used to calculate the SIFs. The PWROG explained that this modified approach used in the TR is consistent with the methods in the 2015 Edition of Section XI of the ASME Code (i.e., A-3212 and A-3411(c)). The staff noted that the 2015 Edition of Section XI of the ASME Code has been incorporated by reference in to 10 CFR 50.55a, “Codes and Standards;” thus, the staff finds the use of these modified procedures used in the TR for the nozzle-to-shell welds and upper shell forging to be acceptable.

The PWROG stated that the method described in A-3200 of Section XI of the ASME Code, including crack face pressure, with an actual FEA pressure stress profile will be used for the SIF calculations. The staff finds the use of procedures consistent with A-3200 of Section XI of the ASME Code for the nozzle-to-shell welds and upper shell forging to be acceptable.

Nozzle Forging

Section 4.1.1 of the TR states that the nozzle corner is the bounding location for the nozzle forging and the nozzle corner flaws are considered using the quarter circular crack as shown in Figure 4-1 of the TR. Furthermore, the PWROG provided the closed form solutions to compute the crack tip K_I values. The staff finds selecting the nozzle corner to be the bounding location for the nozzle forging to be reasonable because of the increased stress levels at this location caused by the structural discontinuities of the nozzle corner. The staff also finds the PWROG's use of closed form solutions for the nozzle forging in the TR to be acceptable because they are consistent with procedures for determining SIFs in Section XI of the ASME Code that have been incorporated by referenced in 10 CFR 50.55a.

3.2.4 Selection of Transients

Appendix K contains acceptance criteria and evaluation procedures for determining acceptability for operation of RVs when the vessel metal temperature is in the upper shelf range. The methodology is based on the principles of elastic-plastic fracture mechanics. Flaws are to be postulated in RV locations of predicted low USE, and the applied J-integral for these flaws are to be calculated and compared with the J-R curve of the material to determine acceptability. K-1100 of Appendix K indicates that all transients for the RV are to be considered. Further, Section 4 of RG 1.161 provides additional guidance for the selection of Service Level C and D transients. The staff's review of the transients selected in the TR for (a) Service Level A and B Transients and (b) Service Level C and D Transients are documented below.

Selection of Service Level A and B Transients

Section 2.1 of the TR indicates that the applicant selected the cooldown transient for North Anna with a constant pressure of 2750 psia assumed throughout the transient, which bounds all Levels A/B conditions and is consistent with and based on the Appendix K, 100°F/hour cooldown rate guidance coincident with the use of a high pressure value.

In its RAI responses dated November 19, 2020, the PWROG confirmed that it considered the design transients defined by the ASME Section III Certified Design Specification, specifically the Service level A/B (i.e., normal/upset conditions) in UFSAR Table 5.2-4. The PWROG explained that the cooldown transient was chosen to bound all Levels A/B service loading conditions, and a constant accumulation pressure of 2750 psia was used (i.e., 1.1 times design pressure) per K-1300 and K-4220 of Appendix K. Furthermore, staff noted that the plant cooldown transient with a 100°F/hour cooldown rate is consistent with K-4210(c) of Appendix K. The staff finds the TR adequately addressed K-1100 of Appendix K to Section XI of the ASME Code because the CLB design transients at North Anna were assessed and the transient selected is consistent with the guidelines in Appendix K.

Selection of Service Level C and D Transients

Section 2.1 of the TR indicates that the Level C/D transient selection is based on the guidance in RG 1.161 and explained that the Level D transient is the steam line break (SLB). Furthermore, Section 4.1 of the TR indicates that only the SLB transient is specified for Level D conditions.

In Section 4.1 of the TR and in its RAI responses dated November 19, 2020, the PWROG states that there is no applicable emergency (Level C) transient defined in the Westinghouse RV design specification. The staff reviewed UFSAR Table 5.2-4 Section and confirmed there are no applicable emergency (Level C) transient as part the CLB for North Anna. Thus, the staff finds that it is appropriate that Level C transients need not be addressed in the TR.

The staff noted that K-1100 of Appendix K states, in part, that "[a]ll specified design transients for the reactor vessel shall be considered." The staff reviewed North Anna UFSAR Table 5.2-4 and noted that beyond the "steam pipe break" transient there are additional Service Level D conditions (i.e., Faulted Conditions) that are applicable to the CLB for North Anna, such as the "Main reactor coolant pipe break" and "Design-basis earthquake." Section 4 of RG 1.161 states that selection of the limiting transients for Service Levels C and D is a key aspect of evaluating the integrity of reactor pressure vessels that contain materials with Charpy USE less than 50 ft- lb. In its RAI responses dated November 19, 2020, the PWROG explained that as documented in the UFSAR Section 5.2.4, the North Anna Leak-Before-Break (LBB) analysis was performed on the main coolant loop piping and was approved by the NRC on August 31, 1999; thus, the staff noted a main reactor coolant pipe break is not applicable. Further, the PWROG also confirmed that the Level D condition included the pipe rupture and Safe Shutdown Earthquake (SSE) loads on the nozzle and support pad for the EMA. Thus, the staff finds that the use of the Steam Line Break (SLB) transient in the TR adequately addressed K-1100 of Appendix K with respect to plant-specific Service Level C and D transients for North Anna.

3.2.5 Evaluation and Acceptance Criteria Per ASME Code, Section XI, Appendix K

3.2.5.1 Service Level A and B Transients

K-2200 states that when evaluating the weld material and the base material for Service Level A and B loadings, the postulated flaws (e.g., axial and/or circumferential) are interior semi-elliptical surface flaws with a depth of ¼ of the wall thickness and a length to depth (*l/a*) aspect ratio of 6. The J-integral resistance versus flaw extension curve shall be a conservative representation for the vessel material under evaluation. Furthermore, two criteria must 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.

The staff noted that flaw stability at a given applied load is verified when the slope of the $J_{applied}$ curve is less than the slope of the J-R curve at the point on the J-R curve where the two curves intersect, as described in K-4310 of Appendix K. The staff's review of the methodology, including inputs, for determining the $J_{applied}$ are documented in SE Section 3.2.1 through 3.2.3.

RV Nozzle-to-Shell Welds

The staff reviewed the results for the nozzle-to-shell welds and noted that the $J_{applied}$ at 0.1-inch flaw extensions (J_1) for the inlet nozzle-to-shell weld ([ ] lbf/in) and outlet nozzle-to-shell weld ([ ] lbf/in) are below the J-R at 0.1 inch flaw extension ($J_{0.1}$) of [ ] lbf/in. Thus, the staff finds the acceptance criteria in K-2200(a)(1) of Appendix K is satisfied for the inlet nozzle-to-shell weld and outlet nozzle-to-shell weld. Furthermore, the staff reviewed the Figure 4-2 and Figure 4-3 of the TR and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves for the inlet and outlet nozzle-to-shell welds, respectively; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-2200(a)(2) of Appendix K is satisfied.

RV Upper Shell Forging

The staff reviewed the results for the upper shell forging and noted that the $J_{applied}$ at 0.1-inch flaw extensions (J_1) for the circumferential flaw ([ ] lbf/in) and axial flaw ([ ] lbf/in) are below $J_{0.1}$ at the respective crack tip temperatures ([ ] lbf/in - circumferential flaw and [ ] lbf/in - axial flaw). Thus, the staff finds the acceptance criteria in K-2200(a)(1) of Appendix K is satisfied for the upper shell forging. Furthermore, the staff reviewed the Figure 4-8 and Figure 4-9 of the TR, for the circumferential and axial flaw, respectively, and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves for the upper shell forging; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-2200(a)(2) of Appendix K is satisfied.

RV Nozzle Forging

The staff reviewed the results for the inlet nozzle corner and noted that the $J_{applied}$ at 0.1-inch flaw extensions (J_1) for the circumferential flaw ([ ] lbf/in) and axial flaw ([ ] lbf/in) are below $J_{0.1}$ at the respective crack tip temperatures ([ ] lbf/in - circumferential flaw and [ ] lbf/in - axial flaw). Thus, the staff finds the acceptance criteria in K-2200(a)(1) of Appendix K is satisfied for the inlet nozzle corner. Furthermore, the staff reviewed the Figure 4-12 and Figure 4-14 of the TR, for the inlet nozzle forging circumferential and axial flaw, respectively, and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-2200(a)(2) of Appendix K is satisfied.

The staff reviewed the results for the outlet nozzle corner and noted that the $J_{applied}$ at 0.1-inch flaw extensions (J_1) for the circumferential flaw ([ ] lbf/in) and axial flaw ([ ] lbf/in) are below $J_{0.1}$ at the respective crack tip temperatures ([ ] lbf/in - circumferential flaw and [ ] lbf/in - axial flaw). Thus, the staff finds the acceptance criteria in K-2200(a)(1) of Appendix K is satisfied for the outlet nozzle corner. Furthermore, the staff reviewed the Figure 4-13 and Figure 4-15 of the TR, for the outlet nozzle forging circumferential and axial flaw, respectively, and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-2200(a)(2) of Appendix K is satisfied.

3.2.5.2 Service Level C and D Transients

As discussed in SE Section 3.2.4 Service Level C transients are not applicable to the CLB for North Anna; thus, K-2300 of Appendix K is not applicable.

In accordance with K-2400(a), the Level D postulated flaws shall be the same as those specified for Level C in K-2300. Specifically, the postulated flaws (e.g., axial and/or circumferential) are interior semi-elliptical surface flaws with depths up to 1/10 of the wall thickness of the base metal plus cladding, with total depth not exceeding 1 inch. For cases where 1/10 wall thickness plus cladding exceeded 1 inch, 1 inch is used for the postulated flaws for Level D, and the length to depth (l/a) aspect ratio is 6. K-2400(a) further states flaw extensions shall be ductile and stable, using a structural factor of 1 on loading. K-2400(b) and (c) states that (1) the J- integral resistance versus flaw extension curve shall be a best estimate representation (i.e., mean curve) for the vessel material under evaluation, and (2) 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, respectively. The staff noted that flaw stability at a given applied load is verified when the slope of the $J_{applied}$ curve is less than the slope of the J-R curve at the point on the J-R curve where the two curves intersect, as described in K-4310 of Appendix K. The staff's review of the methodology, including inputs, for determining the $J_{applied}$ are documented in SE Section 3.2.1 through 3.2.3.

Section 4.1.4 of the TR states the procedures for the J-applied calculation for Levels C/D described in K-5000 of Appendix K are the same as those for Levels A/B described in K-4000, except that the effect of cladding/base metal differential thermal expansion needs to be considered for Levels C/D per K-5210(a). In its RAI responses dated November 19, 2020, the PWROG confirmed that the FEA for Level C/D explicitly modeled the effect of the cladding/base metal differential thermal expansion by incorporating the cladding with appropriate material properties such as thermal expansion coefficients per ASME Section II. Based on the PWROG's confirmation, the staff finds that the TR addresses K-5210(a) of Appendix K by including the cladding material properties consistent with ASME Section II.

<u>RV Nozzle-to-Shell Welds</u>

The staff reviewed the results for the nozzle-to-shell welds and noted that the all applied J_1 at various flaw depths for the inlet nozzle-to-shell weld and outlet nozzle-to-shell weld are below $J_{0.1}$ ([ ] lbf/in). Thus, the staff finds the acceptance criteria in K-2400(a) of Appendix K is satisfied for the inlet nozzle-to-shell weld and outlet nozzle-to-shell weld. Furthermore, the staff reviewed the Figure 4-5 and Figure 4-7 of the TR and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves for the inlet and outlet nozzle-to-shell welds, respectively; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-3400 of Appendix K is satisfied. Further based on its review of these figures, the staff noted that the flaw depth after stable flaw extension (i.e., point at which the applied J-integral intersects the mean J-R curve) is a small percentage of the inlet and outlet nozzle-to-shell weld thickness; thus, the staff finds the acceptance criteria in K-2400(c) related to 75% of the vessel wall thickness is satisfied. The staff reviewed Table 4-5 of the TR and noted that the maximum internal pressure during a SLB transient (i.e., 2.5 ksi) is significantly less than the tensile instability pressures calculated per K-5300 of Appendix K; thus, the staff finds the acceptance criteria in K-2400(c) related to tensile instability of the remaining ligament is satisfied.

<u>RV Upper Shell Forging</u>

The staff reviewed the results for the upper shell forging and noted that the all applied J_1 at various flaw depths for the circumferential and axial flaws are below $J_{0.1}$ ([ ] lbf/in). Thus, the staff finds the acceptance criteria in K-2400(a) of Appendix K is satisfied for the upper shell forging. Furthermore, the staff reviewed the Figure 4-10 and Figure 4-11 of the TR, for the circumferential and axial flaw, respectively, and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves for the upper shell forging; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-3400 of Appendix K is satisfied for the upper shell forging. Further based on its review of these figures, the staff noted that the flaw depth after stable flaw extension (i.e., point at which the applied J-integral intersects the mean J-R curve) is a small percentage of the upper shell forging thickness; thus, the staff finds the acceptance criteria in K-2400(c) related to 75% of the vessel wall thickness is satisfied. The staff reviewed Table 4-8 of the TR and noted that the maximum internal pressure during a SLB transient (i.e., 2.5 ksi) is less than the tensile instability pressures calculated per K-5300 of Appendix K; thus, the staff finds the acceptance criteria in K-2400(c) related to tensile instability of the remaining ligament is satisfied.

RV Nozzle Forging

The staff reviewed the results for the inlet and outlet nozzle corner and noted that the all applied J_1 at various flaw depths for the circumferential and axial flaws are below $J_{0.1}$ ([ ] lbf/in). Thus, the staff finds the acceptance criteria in K-2400(a) of Appendix K is satisfied for the inlet and outlet nozzle forging. Furthermore, the staff reviewed the Figure 4-16 through Figure 4-19 of the TR, for the circumferential and axial flaws of the applicable forging, and noted that the slope of $J_{applied}$ is less than the J-R curve at the intersection of curves for both the inlet and outlet nozzle corners; thus, the staff finds that flaw stability has been verified and the acceptance criteria in K-3400 of Appendix K is satisfied for the inlet nozzle forging and outlet nozzle forging. Further based on its review of these figures, the staff noted that the flaw depth after stable flaw extension (i.e., point at which the applied J-integral intersects the mean J-R curve) is a small percentage of the inlet and outlet nozzle forging thickness; thus, the staff finds the acceptance criteria in K-2400(c) related to 75% of the vessel wall thickness is satisfied. The staff reviewed Table 4-11 of the TR and noted that the maximum internal pressure during a SLB transient (i.e., 2.5 ksi) is less than the tensile instability pressures calculated per K-5300 of Appendix K; thus, the staff finds the acceptance criteria in K-2400(c) related to tensile instability of the remaining ligament is satisfied.

4.0 CONCLUSIONS

The NRC staff concludes that the TR demonstrates that for the components in the scope of this report for North Anna, there is adequate margin of safety against ductile fracture for Service Level A and B loads, and Service Level D loads, through the subsequent period of extended operation (i.e., 60-80 years of plant operation). The NRC staff also concludes that the TR may be referenced in the SLR for North Anna, as a basis for demonstrating that the USE TLAA for the components in the scope of this report has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.1 Summary of Regulatory Compliance

The NRC has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered, and (2) there is reasonable assurance that such activities will be conducted in compliance with the NRC's regulations.

5.0 CONDITIONS, LIMITATIONS, AND/OR ACTION ITEMS

Based on the review of this TR, NRC staff concludes that there is no condition or limitation for the EMA at North Anna that needs to be imposed at this time.

6.0 REFERENCES

1. Letter from Michael Powell, Pressurized Water Reactor Owners Group, to USNRC Document Control Desk, May 27, 2020, Submittal of PWROG-19047-P/NP Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20149K668)

2. Letter from Michael Powell, Pressurized Water Reactor Owners Group, to USNRC Document Control Desk, December 3, 2020, Transmittal of the Response to Request for Additional Information, RAIs 1-7 Associated with PWROG-19047-P/NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (ADAMS Accession No. ML20338A520), and Enclosure - Westinghouse Response to U.S. NRC Request for Additional Information (RAI) for PWROG-19047-P/NP, Rev. 0, November 19, 2020 (ADAMS Accession No. ML20338A522)

3. Topical Report PWROG-19047-NP Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis," May 2020 (ADAMS Accession No. ML20149K671)

4. American Society of Mechanical Engineers, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," *of the ASME Boiler and Pressure Vessel Code*, New York

5. Eason, E. D.; Wright, J. E.; Nelson, E. E, NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991. (https://ntrl.ntis.gov/NTRL/dashboard/searchResults/titleDetail/NUREGCR5729.xhtml)

6. BAW-2192, Revision 0, Supplement 1P/NP-A, 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, April 19, 2019 (ADAMS Accession Nos. ML19101A355/ML19101A344)

7. BAW-2192, Revision 0, Supplement 2P/NP-A, 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, August 31, 2020 (ADAMS Accession Nos. ML20234A361/ML20234A363)

8. NUREG 0800 Section 5.3.2, "Pressure Temperature Limits, Upper Shelf Energy, and Pressurized Thermal Shock" (ADAMS Accession No. ML070380185)

9. Regulatory Guide1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb," June 1995 (ADAMS Accession No. ML003740038)

10. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (ADAMS Accession No. ML010890301)

11. RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988 (ADAMS Accession No. ML003740284)

12. PWROG-17090-NP-A Revision 0 - Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination (PA-MSC-1367, Task 3), January 2020 (ADAMS Accession No. ML20024E238)

13. WCAP-18364-NP, Rev. 1 - North Anna Units 1 and 2 Time-Limited Aging Analysis on Reactor Vessel Integrity for Subsequent License Renewal (SLR), March 31, 2020 (ADAMS Accession No. ML20246G701)

14. WCAP-18015-NP, Revision "Extended Beltline Pressure Vessel Fluence Evaluations Applicable to North Anna 1 & 2," September 2018 (ADAMS Accession No. ML20140A336)

15. North Anna Power Station, Unit Nos. 1 and 2 - RE: Request To Revise Reactor Vessel Material Surveillance Capsule Withdrawal Schedules (EPID L-2019-LLL-0038), August 6, 2020 (ADAMS Accession No. ML20216A299)

16. WCAP 14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 31, 2004 (ADAMS Accession No. ML050120209)

17. North Anna Power Station, Units 1 & 2, Revision 56 to Updated Final Safety Analysis Report (ADAMS Accession No. ML20309A590)

Principal Contributors: On Yee, NRR/DNRL
David Dijamco, NRR/DNRL
Benjamin Parks, NRR/DSS

Date:

NON-PROPRIETARY INFORMATION

TOPICAL REPORT PWROG-19047-P/NP, REVISION 0 COMMENT DISPOSITION TABLE						
Comment No.		Comment Text Location in the DSE		Comment Type (Clarification, Editorial, Accuracy, Proprietary)	Suggested Revision	NRC Response
PWROG	NRC	Page No.	Line No.			
1		6	25	Editorial	Please add: “Rev. 0, ” after BAW-2192.	NRC staff finds the PWROG comment acceptable and the revisions have been incorporated.
2		6	27	Editorial	Please add: “Rev. 0, ” after BAW-2192; and Please add: “-A” after Supplement 2P.	NRC staff finds the PWROG comment acceptable and the revisions have been incorporated.
3		6	29	Editorial	Please add: “Rev. 0, ” after BAW-2192; and Please add: “-A” after Supplement 2P.	NRC staff finds the PWROG comment acceptable and the revisions have been incorporated.
	4	9	39-40	Proprietary	Add [ ] and yellow highlight to identify proprietary values of J_1 and $J_{0.1}$.	NRC has provided add’l comments. Revisions have been incorporated.
	5	10	2-4	Proprietary	Add [ ] and yellow highlight to identify proprietary values of J_1 and $J_{0.1}$.	NRC has provided add’l comments. Revisions have been incorporated.
	6	10	14-16	Proprietary	Add [ ] and yellow highlight to identify proprietary values of J_1 and $J_{0.1}$.	NRC has provided add’l comments. Revisions have been incorporated.
	7	10	24-26	Proprietary	Add [ ] and yellow highlight to identify proprietary values of J_1 and $J_{0.1}$.	NRC has provided add’l comments. Revisions have been incorporated.
	8	11	18	Proprietary	Add [ ] and yellow highlight to identify proprietary value of $J_{0.1}$.	NRC provided add’l comments. Revisions have been incorporated.
	9	11	35	Proprietary	Add [ ] and yellow highlight to identify proprietary value of $J_{0.1}$.	NRC has provided add’l comments. Revisions have been incorporated.

NON-PROPRIETARY INFORMATION

NON-PROPRIETARY INFORMATION

TOPICAL REPORT PWROG-19047-P/NP, REVISION 0 COMMENT DISPOSITION TABLE						
Comment No.		Comment Text Location in the DSE		Comment Type (Clarification, Editorial, Accuracy, Proprietary)	Suggested Revision	NRC Response
PWROG	NRC	Page No.	Line No.			
	10	12	3	Proprietary	Add [ ] and yellow highlight to identify proprietary value of $J_{0.1.}$	NRC has provided add'l comments. Revisions have been incorporated.

NON-PROPRIETARY INFORMATION

ACKNOWLEDGEMENTS

This topical report was developed and funded by the PWR Owners Group under the leadership of the participating utility representatives of the Materials Committee. The author would like to thank the following people and/or organizations for their valuable contributions to this report:

Mark Rinckel, Framatome

Martin Kolar, Framatome

Chuck Tomes, Dominion

LEGAL NOTICE

This report was prepared as an account of work performed by Westinghouse Electric Company LLC and Framatome Inc. Neither Westinghouse Electric Company LLC nor Framatome Inc. nor its affiliates, nor any person acting on its behalf:

Makes any warranty or representation, express or implied including the warranties of fitness for a particular purpose or merchantability, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this report.

COPYRIGHT NOTICE

DISTRIBUTION NOTICE

This report was prepared for the PWR Owners Group. This Distribution Notice is intended to establish guidance for access to this information. This report (including proprietary and non-proprietary versions) is not to be provided to any individual or organization outside of the PWR Owners Group program participants without prior written approval of the PWR Owners Group Program Management Office. However, prior written approval is not required for program participants to provide copies of Class 3 Non-Proprietary reports to third parties that are supporting implementation at their plant, and for submittals to the NRC.

PWR Owners Group
United States Member Participation* for PA- MSC-1481, R3

Utility Member	Plant Site(s)	Participant Yes	Participant No
Ameren Missouri	Callaway (W)		X
American Electric Power	D.C. Cook 1 & 2 (W)		X
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Dominion Energy	Millstone 2 (CE)		X
	Millstone 3 (W)		X
	North Anna 1 & 2 (W)	X	
	Surry 1 & 2 (W)		X
	V.C. Summer (W)		X
Duke Energy Carolinas	Catawba 1 & 2 (W)		X
	McGuire 1 & 2 (W)		X
	Oconee 1, 2, & 3 (B&W)		X
Duke Energy Progress	Robinson 2 (W)		X
	Shearon Harris (W)		X
Entergy Palisades	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)		X
Entergy Operations South	Arkansas 1 (B&W)		X
	Arkansas 2 (CE)		X
	Waterford 3 (CE)		X
Exelon Generation Co. LLC	Braidwood 1 & 2 (W)		X
	Byron 1 & 2 (W)		X
	Calvert Cliffs 1 & 2 (CE)		X
	Ginna (W)		X
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)		X
	Davis-Besse (B&W)		X
Florida Power & Light \ NextEra	St. Lucie 1 & 2 (CE)		X
	Turkey Point 3 & 4 (W)		X
	Seabrook (W)		X
	Pt. Beach 1 & 2 (W)		X
Luminant Power	Comanche Peak 1 & 2 (W)		X

PWR Owners Group
United States Member Participation* for PA- MSC-1481, R3

Utility Member	Plant Site(s)	Participant	
		Yes	No
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)		X
PSEG – Nuclear	Salem 1 & 2 (W)		X
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2 (W)		X
	Vogtle 1 & 2 (W)		X
Tennessee Valley Authority	Sequoyah 1 & 2 (W)		X
	Watts Bar 1 & 2 (W)		X
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)		X
Xcel Energy	Prairie Island 1 & 2 (W)		X

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group
International Member Participation* for PA- MSC-1481, R3

Utility Member	Plant Site(s)	Participant	
		Yes	No
Asociación Nuclear Ascó-Vandellòs	Asco 1 & 2 (W)		X
	Vandellos 2 (W)		X
Centrales Nucleares Almaraz-Trillo	Almaraz 1 & 2 (W)		X
EDF Energy	Sizewell B (W)		X
Electrabel	Doel 1, 2 & 4 (W)		X
	Tihange 1 & 3 (W)		X
Electricite de France	58 Units		X
Elektriciteits Produktiemaatschappij Zuid-Nederland	Borssele 1 (Siemens)		X
Eletronuclear-Eletrobras	Angra 1 (W)		X
Emirates Nuclear Energy Corporation	Barakah 1 & 2		X
Hokkaido	Tomari 1, 2 & 3 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 3 (W)		X
	Ohi 1, 2, 3 & 4 (W & MHI)		X
	Takahama 1, 2, 3 & 4 (W & MHI)		X
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 (W)		X
	Hanbit 1 & 2 (W)		X
	Hanbit 3, 4, 5 & 6 (CE)		X
	Hanul 3, 4 , 5 & 6 (CE)		X
Kyushu	Genkai 2, 3 & 4 (MHI)		X
	Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Shikoku	Ikata 2 & 3 (MHI)		X
Taiwan Power Co.	Maanshan 1 & 2 (W)		X

* **Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1 INTRODUCTION

The purpose of this topical report is to document the equivalent margins analysis (EMA) for the North Anna Units 1 and 2 reactor vessel (RV) inlet and outlet nozzle Rotterdam welds, nozzle forgings and nozzle belt forgings (a.k.a., upper shell forgings). These locations were chosen for their upper-shelf energy (USE) potentially falling below the 50 ft-lb limit at 80-years (72 EFPY) for subsequent license renewal (SLR). Materials with end-of-license-extension (EOLE) USE below the 50 ft-lb limit are required to be evaluated per paragraph IV.A.1.a of 10 CFR 50, Appendix G, for equivalent margins of safety specified in ASME Code Section XI, Appendix K [8].

The North Anna Units 1 and 2 reactor vessels are Westinghouse-designed vessels whose subject nozzle welds were fabricated by the Rotterdam Shipyards. There are only two locations with projected SLR USE at or below the required 50 ft-lbs:

- North Anna Unit 1, Inlet Nozzle Forging 11, Heat #990268-21
- North Anna Unit 2, Intermediate Shell Forging 04, Heat #990496 / 292424

The EMA for the nozzle and upper shell forgings utilizes the multivariable model for RV base metal reported in NUREG/CR-5729. Although the Rotterdam nozzle-to-shell welds are projected to have USE greater than 50 ft-lbs, they are evaluated proactively in this EMA for asset management consideration. The EMA for the Rotterdam welds utilizes the B&WOG J-integral resistance (J-R) Model 6B reported in BAW-2192, Revision 0, Supplement 1P-A, Rev. 0, Appendix A [1]. The justification for using the B&WOG Model 6B for the North Anna Rotterdam welds is addressed in BAW-2192, Supplement 2P, Revision 0 [2].

This low upper-shelf toughness EMA is based on the projected RV neutron fluence at 80 years of operation for SLR at the RV inlet and outlet nozzle regions, which are projected to exceed 1.0 E+17 n/cm^2, E > 1.0 MeV, and are qualified as extended beltline materials. The general configuration of the North Anna Units 1 and 2 RVs, and the locations evaluated in this EMA are shown in Figure 3-1. There are no longitudinal welds on the North Anna Units 1 and 2 RVs.

2 REGULATORY REQUIREMENTS

2.1 REGULATORY REQUIREMENTS

In accordance with 10 CFR 50, Appendix G, IV.A.1, [4] 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, 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."

When the RVs within the scope of this topical report were fabricated, the Charpy V-notch testing of the RV welds was in accordance with the original construction code, which did not require Charpy V-notch tests on the upper shelf. The original construction code for the RV shell and nozzles for both units is ASME Section III, 1968 Edition through the Winter 1968 Addenda, as discussed in the North Anna Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) [5, Table 5.2-3].

In accordance with NRC Regulatory Guide 1.161 [6], 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 USE falls below the 50 ft-lb limit of Appendix G of 10 CFR Part 50. However, the staff noted that Appendix K does not provide information on the selection of transients and provides very little detail on the selection of material properties. Consistent with BAW-2192, Revision 0, Supplement 1P-A, Revision 0 [1], the cooldown transient for North Anna Units 1 and 2 with a constant pressure of 2750 psia assumed throughout the transient bounds all Levels A/B conditions. This is consistent with and based on the ASME Section XI, Appendix K 100°F/hour cooldown rate guidance coincident with the use of a high pressure value. The Level C/D transient selection is based on the guidance in Regulatory Guide

1.161 Section 4.0 [6]. There are no applicable emergency (Level C) transients in the RV design specifications. The Level D transient is the steam line break (SLB). Additional transient discussions are contained in Section 4.1. Physical properties for the forging and weld materials are from construction ASME Code [7]. The J-Resistance (J-R) models are discussed in detail in Section 3.2.

2.2 COMPLIANCE WITH 10 CFR 50 APPENDIX G AND ACCEPTANCE CRITERIA

The analyses reported herein are performed in accordance with the 2013 Edition of Section XI of the ASME Code, Appendix K [8]. The edition of ASME Section XI discussed in 10 CFR 50.55a is the 2013 Edition. The material properties used in this analysis are based on the original RV construction code, ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, with Addenda up to and including the Winter of 1968 [7].

2.2.1 Acceptance Criteria

ASME Section XI [8], Appendix K provides the acceptance criteria for the Level A, B, C and D conditions. These criteria summarized in the following subsections are consistent with Regulatory Guide 1.161.

2.2.1.1 K-2200 Levels A and B Service Loadings

(a) Postulated axial and circumferential flaws are interior semi-elliptical surface flaws with a depth of ¼ of the wall thickness and a length to depth (*l/a*) aspect ratio of 6.

   (1) $J_{applied}$ with a SF of 1.15 for pressure and a SF of 1.0 for thermal (cooldown) shall be less than the J-integral of the material (J-R curve) at a ductile flaw extension of 0.1 inch.

   (2) $J_{applied}$ with a SF of 1.25 for pressure and a SF of 1.0 for thermal (cooldown) shall be ductile and stable.

(b) The J-R curve shall be a conservative representation for the vessel material under evaluation.

   The flaw stability criteria is per K-3400: $\frac{\partial J}{\partial a} < \frac{dJ_R}{da}$ at J ≡ JR. This is further explained in K-4310. The J-R curve shall be plotted on the crack driving force diagram and shall intersect the horizontal axis at the initial flaw depth, a_0. Flaw stability at a given applied load is verified when the slope of the $J_{applied}$ curve is less than the slope of the J-R curve at the point on the J-R curve where the two curves intersect.

2.2.1.2 K-2300 and K-2400, Levels C and D Service Loadings

Per K-2400, the Level D postulated flaws shall be the same as those specified for Level C in K-2300.

(a) Postulated axial and circumferential flaws are interior semi-elliptical surface flaws with depths up to 1/10 of the wall thickness of the base metal plus cladding, with total depth not exceeding 1 inch. For cases where 1/10 wall thickness plus cladding exceeded 1 inch, 1 inch is used for the postulated flaws for Level D. The length to depth (l/a) aspect ratio is 6.

(1) $J_{applied}$ with a SF of 1.0 for thermal and pressure shall be less than the J-R curve at a ductile flaw extension of 0.1 inch.

(2) $J_{applied}$ with a SF of 1.0 for thermal and pressure shall be ductile and stable.

(b) The J-R curve shall be a conservative 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.

The flaw stability criteria is detailed in K-5300.

(a) Stability is verified per K-3400: $\frac{\partial J}{\partial a} < \frac{dJ_R}{da}$ at J ≡ JR.

(b) For Level D Service Loadings, demonstrate that total flaw depth after stable flaw extension is less than or equal to 75% of the vessel wall thickness, and the remaining ligament is not subjected to tensile instability. The internal pressure shall be less than the instability pressure (P_I), calculated by the equations below:

(1) For axial flaw, $P_I = 1.07\sigma_o \left[\frac{1 - A_c/A}{\frac{R_i}{t} + \frac{A_c}{A}}\right]$

(2) For circumferential flaw, $P_I = 1.07\sigma_o \left[\frac{1 - A_c/A}{\frac{R_i^2}{2R_m t} + \frac{A_c}{A}}\right]$

P_I is limited to $P_I = 1.07\sigma_o \left[\frac{t}{R_i}\right]$

where,

σ_o = Flow stress, average of yield strength and ultimate tensile strength

A = An area parameter = $t(l + t)$

A_c = Area of the flaw = $\pi a l / 4$

R_i = Inner radius of the vessel

R_m = Mean radius of the vessel

t = Wall thickness of the vessel

a = Flaw Depth

l = Flaw length

3 EQUIVALENT MARGINS ANALYSIS INPUTS

3.1 FINITE ELEMENT STRESS ANALYSIS

The general procedures for J-integral calculation are described in Appendix K of [8]. As discussed in Section 2.1, the cooldown transient was analyzed to bound Levels A/B. The Level D transient is SLB. Figure 3-1 is a sketch illustrating the North Anna RV upper shell, intermediate shell, inlet and outlet nozzles, and nozzle to shell welds. The finite element model (FEM) is illustrated in Figure 3-2. Geometry and dimensions are taken from design drawings. The applied loadings consist of pressure, thermal and attached piping and support reactions at RV nozzles.

Table 3-1 lists the material properties of SA-508 Class 2 used for the RV shell and nozzle forging base metal. The nozzle-to-shell weld mechanical properties were assumed to be identical to the forging material for the purpose of stress analysis. Table 3-2 lists the material properties for Type 304 stainless steel were used for the cladding.

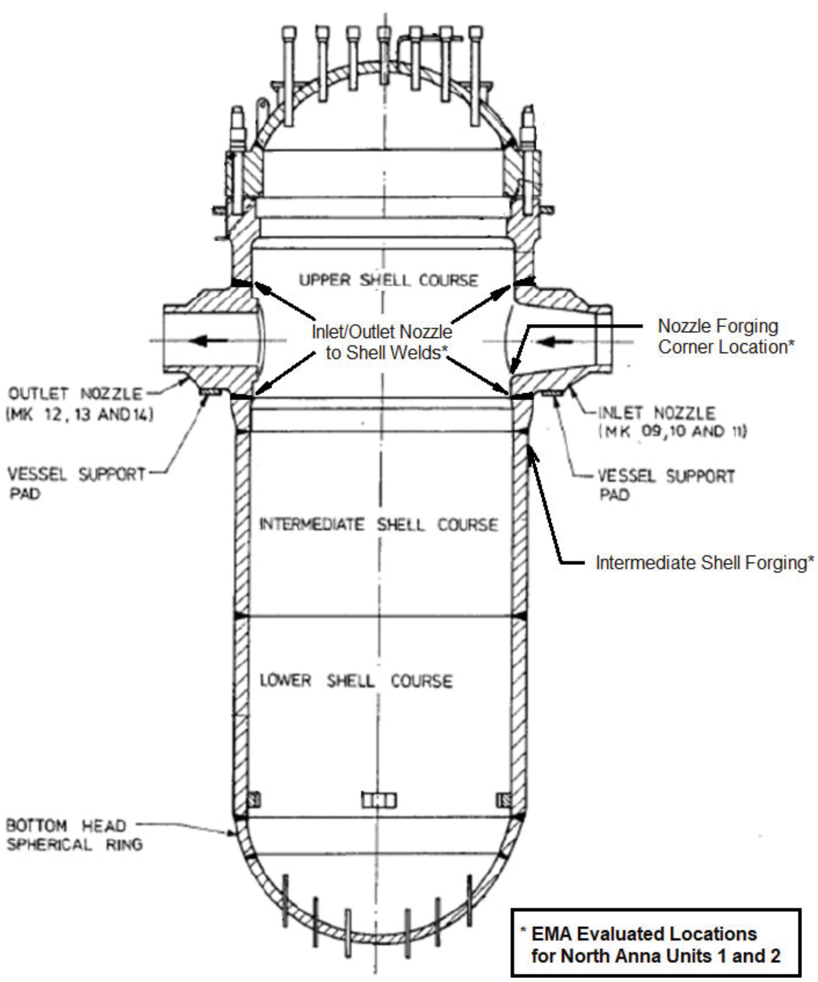

Figure 3-1: North Anna Units 1 and 2 Reactor Vessel Generic Configuration

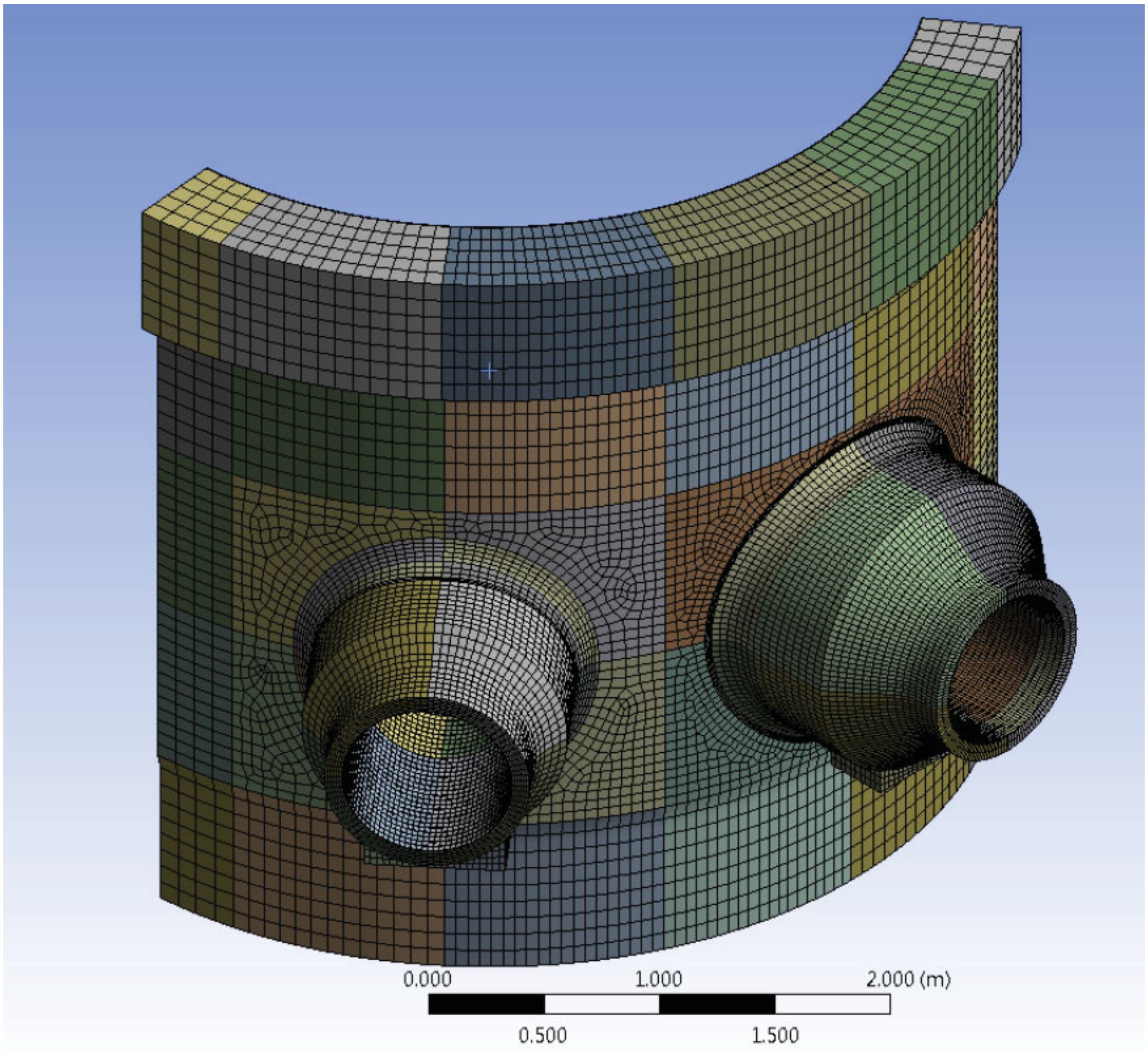

Figure 3-2: Overview of FEM

Table 3-1: Base Metal Material Properties (SA-508 Class 2)

Temp [°F]	E [x10^6 psi]	Temp [°F]	Thermal Expansion, α [x10^{-6} in/(in·°F)]	Conductivity, K [BTU/(hr·ft·°F)]	Thermal Diffusivity, [ft^2/hr]	Heat Capacity, C_p [BTU/(lb$_m$·°F)]	Density, ρ [lb$_m$/in^3]
70	27.9	70	6.07	31.50	0.5692	0.1144	0.28
200	27.7	100	6.13	31.00	0.5509	0.1163	0.28
300	27.4	150	6.25	30.50	0.5421	0.1163	0.28
400	27.0	200	6.38	30.00	0.5246	0.1182	0.28
500	26.4	250	6.49	29.50	0.5075	0.1201	0.28
600	25.7	300	6.60	29.10	0.4928	0.1220	0.28
700	24.8	350	6.71	28.60	0.4770	0.1239	0.28
		400	6.82	28.10	0.4616	0.1258	0.28
		450	6.92	27.60	0.4467	0.1277	0.28
		500	7.02	27.20	0.4338	0.1296	0.28
		550	7.12	26.70	0.4198	0.1315	0.28
		600	7.23	26.20	0.4061	0.1333	0.28
		650	7.33	25.80	0.3915	0.1362	0.28
		700	7.41	25.30	0.3763	0.1390	0.28

Table 3-2: Cladding Material Properties (Type 304 Stainless Steel)

Temp [°F]	E [x10^6 psi]	Temp [°F]	Thermal Expansion, α [x10^{-6} in/(in·°F)]	Conductivity, K [BTU/(hr·ft·°F)]	Thermal Diffusivity, [ft^2/hr]	Heat Capacity, C_p [BTU/(lb$_m$·°F)]	Density, ρ [lb$_m$/in^3]
70	27.4	70	9.11	8.35	0.1498	0.1112	0.29
200	27.1	100	9.16	8.40	0.1495	0.1121	0.29
300	26.8	150	9.25	8.67	0.1525	0.1135	0.29
400	26.4	200	9.34	8.90	0.1548	0.1147	0.29
500	26.0	250	9.41	9.12	0.1568	0.1160	0.29
600	25.4	300	9.47	9.35	0.1589	0.1174	0.29
700	24.9	350	9.53	9.56	0.1601	0.1192	0.29
		400	9.59	9.80	0.1630	0.1200	0.29
		450	9.65	10.00	0.1639	0.1218	0.29
		500	9.70	10.23	0.1659	0.1231	0.29
		550	9.76	10.45	0.1684	0.1238	0.29
		600	9.82	10.70	0.1707	0.1251	0.29
		650	9.87	10.90	0.1721	0.1264	0.29
		700	9.93	11.10	0.1736	0.1276	0.29

3.2 J-INTEGRAL RESISTANCE MODELS

North Anna Units 1 and 2 RV are Westinghouse-design vessels whose subject nozzle-to-shell welds were fabricated by Rotterdam Shipyards. The nozzle weld EMA utilizes the B&WOG J-R model 6B reported in BAW-2192, Revision 0, Supplement 1P-A, Revision 0, Appendix A [1]. The justification for the use of Model 6B for North Anna the nozzle-to-shell welds is provided in BAW-2192, Supplement 2P, Revision 0 [2]. For conservatism, the maximum cold leg temperature is utilized. Similarly, the maximum reported copper content and the fluence value at the nozzle-to-shell weld is utilized as summarized in Table 3-3 below. Table 3-4 lists the J-R curve for the nozzle-to-shell welds provided by Framatome.

Table 3-3: J-R Calculation Input Parameters for Nozzle Welds

[ ] f

Table 3-4: Model 6B J-R curve for Nozzle-to Shell Welds at 552°F for Levels A/B

[ ] f

The J-R curves for the RV nozzle and shell forgings were developed in accordance with NUREG/CR-5729 [3], Table 11, Charpy model without fluence. This Charpy model is the same model described in Section 3.3 of Regulatory Guide 1.161 [6]. The test specimen's net-section thickness, B_n of 1.0 inches is used per Section 3 of [6]. The Charpy model is applicable to North Anna nozzle and shell forgings because they are bounded by the range of explanatory variables (fluence, copper content, etc.) used to develop the J-R model. The calculated J-R curve data for the nozzle and shell forging is listed in Table 3-6 and Table 3-7.

The basic form J_R is expressed in [3 and 6] as:

$$\mathbf{J_R = (MF)C1(\Delta a)^{C2}exp[C3(\Delta a)^{C4}]} \qquad \textbf{Equation 1}$$

where,

MF = margin factors, = 0.749 for Levels A, B and C; MF = 1.0 for Level D

Δa = flaw extension, inches

For the Charpy model, C1 is defined in [3, Eq. 10] as:

$$\mathbf{lnC1 = a_1 + a_2 lnCVN + a_3 T + a_4 lnB_n} \qquad \textbf{Equation 2}$$

where,

a_1 through a_5 are defined in [3, Table 11], and shown in Table 3-5.

CVN = Charpy V-notch Impact Energy, ft-lbs. The 80-year projected USE value is used.

T = temperature, °F

B_n = test specimen net thickness, inches. B_n = 1.0 inch is used per Section 3 of [6].

C2 and C3 are defined in [3, Eq. 7 and 8] as:

$$\mathbf{C2 = d_1 + d_2 lnC1 + d_3 lnB_n} \qquad \textbf{Equation 3}$$

$$\mathbf{C3 = d_4 + d_5 lnC1 + d_6 lnB_n} \qquad \textbf{Equation 4}$$

where,

d_1 through d_6 are defined in [3, Table 11], and shown in Table 3-5.

C4 is defined in [3, Table 11], and shown in Table 3-5.

Table 3-5: Parameters for RPV Base Metals, J_d Model [3, Table 11]

Parameter		Variable	*Charpy* Model
lnC1	a1	(constant)	-2.44
	a2	ln CVN	1.13
	a3	T	-0.00277
	a4	$\ln B_n$	0.0801
	a5	ϕt	
C2	d1	(constant)	0.077
	d2	lnC1	0.116
	d3	$\ln B_n$	-0.0412
C3	d4	(constant)	-0.0812
	d5	lnC1	-0.0092
	d6	$\ln B_n$	-0.0295
C4	C4	(exponent)	-0.409

Table 3-6: RV Inlet and Outlet Nozzle Forgings J-R Curves

a, c, e

Note: NUREG/CR-5729 Charpy Model was used.

Table 3-7: RV Intermediate Shell Forgings J-R Curves

a, c, e

Note:

NUREG/CR-5729 Charpy Model was used. The limiting location for the upper shell is near the intermediate shell. Since the J-R curves for the intermediate shell are more limiting than the upper shell, they are used in the EMA for the upper shell. Additionally, since the upper shell forging stresses were taken from the upper shell to intermediate shell transition, the stress concentration effect was captured, therefore, the upper shell $J_{applied}$ results are also applicable to the intermediate shell.

4 FRACTURE MECHANICS ANALYSIS

The EMA methodology that was used for the North Anna Units 1 and 2 RV locations with projected USE below 50 ft-lbs is consistent with previously NRC approved methodologies for WCAP-13587, Rev. 1, BAW-2178 and BAW-2192. The respective NRC Safety Evaluation Reports are in [12 and 13]. The EMA methodology is discussed further in Section 4.1. Although the Rotterdam nozzle-to-shell welds are projected to have a USE greater than 50 ft-lbs, they are evaluated in this EMA proactively for asset management consideration.

4.1 METHODOLOGY DISCUSSION

The $J_{applied}$ are to be calculated per ASME Section XI, Appendix K [8], which is consistent with BAW-2192, Revision 0, Supplement 1P-A [1]. The maximum $J_{applied}$ values at the critical time points for service Levels A/B and Levels C/D, along with plots of $J_{applied}$ vs. flaw depth, will be compared with the J-R curves for the EMA. The Levels A/B service loadings required by ASME Section XI, Appendix K, are an accumulation pressure (internal pressure load) and a cooldown rate (thermal load). For Level A/B, K-1300 and K-4000 of [8] conservatively defined the accumulation pressure as 1.1 times the design pressure, which is a constant pressure of 2750 psia applied throughout the 100°F/hr cooldown transient.

The actual design thermal transients are used for finite element analysis (FEA) stress and input for the K and J calculations, instead of the generic design pressure and cooling rate in Appendix K [8]. As discussed in Section 2.1 of this topical report, the plant cooldown transient is used to bound all Level A/B conditions. This is also consistent with the Appendix K guidance of 100°F/hour cooldown rate. Based on the design specification, there is no Level C transient for North Anna Units 1 and 2, and only the SLB transient is specified for Level D conditions. The Level D thermal and pressure transients are defined in the design specification. Instead of lumping Levels C and D together as traditional EMA would do, this topical report will refer to it as Level D instead of Level C/D for clarity because there is no Level C conditions.

Appendix K of [8] provides various postulated flaw depths, locations, and orientations, as well as the $J_{applied}$ and stability criteria. Per K-2000 of [8], the postulated flaws shall be oriented along the major axis of the weld of concern. Therefore, only circumferential flaws are applicable to the inlet and outlet nozzle welds. Both axial and circumferential flaws will be postulated for the nozzle and upper shell forgings.

4.1.1 Nozzle-to-Shell Welds and Upper Shell Forging, K_I Using A-3200 [8]

For an axial or circumferential flaw of depth "a," the SIF due to radial thermal gradients can be calculated per K-4210(c) of [8]. However, since the thermal stresses are based on FEA, the procedure in ASME Section XI, Appendix A [8] is used to calculate the SIFs. This method accurately captures the stress states of the actual geometry. The stress profile representation prescribed in A-3200 of [8] is for a location over the flaw depth (x/a) for which the A_i coefficients need to be recalculated for every flaw depth analyzed. The term "x" is defined as the distance through the wall measured from the flawed surface. In order to simplify the calculation, the analysis herein uses through-wall stress profiles (x/t) in a similar fashion. The procedure in A-3320 of [8] is modified for the use of through-wall stress representation. This x/t approach is consistent with the methods prescribed in publications such as API-579 [10] and the 2015 Edition of ASME Section XI, A-3212 and A-3411(c). Note that the 2015 Edition of ASME Section XI, Appendix A specifies that G_i coefficient tables are applicable for both the “x/a” and “x/t” method.

The closed-form solution in K-4210 of [8] for K_{Ip} is generic for cylinder geometry, which is appropriate for the RV. However, preliminary results had determined it to be conservative for the nozzle weld locations. Therefore, the method described in A-3200 of [8], including crack face pressure, with an actual FEA pressure stress profile will be used for the SIF calculations.

The through-wall stress profile is represented as follows by a cubic polynomial:

$$\sigma = A_0 + A_1\left(\frac{x}{t}\right) + A_2\left(\frac{x}{t}\right)^2 + A_3\left(\frac{x}{t}\right)^3$$

$$K_I = \left[(A_0 + A_p)\, G_0 + A_1 G_1\left(\frac{a}{t}\right) + A_2 G_2\left(\frac{a}{t}\right)^2 + A_3 G_3\left(\frac{a}{t}\right)^3\right]\sqrt{\frac{\pi a}{Q}}$$

$$Q = 1 + 4.593\left(\frac{a}{l}\right)^{1.65} - q_y$$

$$q_y = \frac{1}{6}\left[\frac{A_0 G_0 + A_1 G_1\left(\frac{a}{t}\right) + A_2 G_2\left(\frac{a}{t}\right)^2 + A_3 G_3\left(\frac{a}{t}\right)^3}{\sigma_y}\right]^2$$

Where:

a = flaw depth, [in]

t = wall thickness, [in]

l = flaw length, [in]

A_i = coefficients from the cubic polynomial stress profile, i= 0, 1, 2, 3

A_p = 0 for thermal K_{It}; A_p = internal vessel pressure for pressure K_{Ip}

σ_y = material yield strength, ASME temperature-dependent value is used, [ksi]

G_i = free surface correction factors from Table A-3320-1 of [8] for point 1, the deepest point

q_y = plastic zone correction factor

The plastic zone correction factor, q_y, in this application is set to zero because K-4210 of [8] uses the effective flaw depth, "a_e," which includes ductile flaw extension and a plastic zone correction.

4.1.2 Nozzle Corner K_I Closed Form Solution per [11]

The nozzle corner is the bounding location for the nozzle forging. The nozzle corner flaws are considered using the quarter circular crack in a quarter space crack geometry shown in Figure 4-1 for which solutions are available in [11]. Crack tip K_I values are computed using:

$$\sigma = A_0 + A_1 x + A_2 x^2 + A_3 x^3$$

$$K_I = \sqrt{\pi a}\left[0.723A_0 + 0.551\left(\frac{2a}{\pi}\right)A_1 + 0.462\left(\frac{a^2}{2}\right)A_2 + 0.408\left(\frac{4a^3}{3\pi}\right)A_3\right]$$

Where:

σ = the stress perpendicular to the plane of the crack, and A_0, A_1, A_2, and A_3 are the polynomial coefficients for the stress profile

x = the distance from the inner surface where the crack initiates

a = crack depth

FUN 10 — QUARTER-CIRCULAR CRACK IN QUARTER-SPACE

$$K_1 = \sqrt{\pi a}\ [0.723\ A_0 + 0.551\ (\frac{2a}{\pi})\ A_1 + 0.462\ (\frac{a^2}{2})\ A_2 + 0.408(\frac{4a^3}{3\pi})\ A_3\ ]$$

Figure 4-1: K_I Solution for Quarter Circular Crack in Quarter Space [11, page 5]

4.1.3 Calculation of $J_{applied}$ for Small-Scale Yielding

The calculation of $J_{applied}$ due to applied loads accounts for a material's elastic-plastic behavior. When elastic fracture mechanics with small-scale yielding applies, $J_{applied}$ may be calculated using crack tip SIF formulae with a plastic zone correction.

The effective flaw depth for small-scale yielding, a_e, shall be calculated per K-4210 of [8]:

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

Where, K_{Ip} and K_{It} are SIF due to pressure and thermal stresses, respectively.

Both axial and circumferential K'_{Ip} and K'_{It} are calculated the same way as K_{Ip} and K_{It} as discussed in Section 4.1.1, except that the flaw depth, *a,* is substituted with the effective flaw depth, a_e. Then, the $J_{applied}$ for small-scale yielding is calculated using the following formula:

$$J_1 = 1000\frac{(K'_{Ip}+K'_{It})^2}{E'}, \text{ [in-lb/in}^2\text{]}$$

Where:

$E' = E/(1-\nu)^2$,[ksi]

E = Young's modulus, [ksi]

ν = Poisson's ratio = 0.3

4.1.4 Postulated Flaw

The procedures for the $J_{applied}$ calculation for Levels C/D described in K-5000 of [8] are the same as those for Levels A/B described in K-4000, except that the effect of cladding/base metal differential thermal expansion needs to be considered for Levels C/D per K-5210(a) of [8]. Therefore, stress data from the finite element model (FEM) with cladding is included for the Levels C/D evaluation. Additional details of the postulated flaw requirements per K-2200, K-2300 and K-2400 are summarized in Section 2.2.1.

4.1.5 Weld Residual Stress

The weld residual stress (WRS) is to be included for Level D to be consistent with the EMA performed in BAW-2192NP, Supplement 1 [13]. The normalized WRS profile is from [9, Section 4.1.3.4, Figure 30]. The WRS was directly added to the nozzle to shell weld FEA thermal stresses for the calculation of $K_{It.}$

4.1.6 Stress Due to Mechanical Loads

There are a total of four cases of mechanical piping and support load combinations: two cases for Levels A/B and two for Level D. Since the SF is only applicable to pressure, the mechanical stress is directly added to the FEA thermal stress. The maximum through-thickness mechanical stress of the design conditions is added to the corresponding thermal stresses. All K_I and $J_{applied}$ are calculated for all transient time points. The limiting $J_{appiled}$ at the 0.1 inch flaw extension is reported.

4.1.7 Temperature Range for Upper Shelf Fracture Toughness Evaluations

Upper-shelf fracture toughness is determined through use of Charpy V-notch impact energy versus temperature plots by noting the temperature above which the Charpy energy remains on a plateau, maintaining a relatively high constant energy level. Similarly, fracture toughness can be addressed in three different regions on the temperature scale, i.e., a lower-shelf toughness region, a transition region, and an upper-shelf toughness region. Fracture toughness of reactor vessel steel and associated weld metals are conservatively predicted by the ASME initiation toughness curve, K_{Ic}, in the lower shelf and transition regions. In the upper shelf region, for the Linde 80 and similar welds (i.e., Rotterdam welds), the upper shelf toughness curve, K_{Jc}, is derived from the upper-shelf J-integral resistance model described in Section 3.2. The upper-shelf toughness then becomes a function of fluence, copper content, temperature, and fracture specimen size. When upper-shelf toughness is plotted versus temperature, a plateau-like curve develops that decreases slightly with increasing temperature. Since the present analysis addresses the low upper-shelf fracture toughness issue, only the upper-shelf temperature range, which begins at the intersection of K_{Ic} and the upper-shelf toughness curves, is considered.

Transition region toughness is obtained from the ASME Section XI [8] equation for crack initiation, Section A-4200,

$$K_{Ic} = 33.2 + 20.734\exp[0.02(T-RT_{NDT})]$$

Using an RT_{NDT} value of 208.3°F (page 5 of [5]) for a flaw depth of 1/10 the wall thickness where:

K_{Ic} = transition region toughness, ksi√in

T = crack tip temperature, °F

Stress intensity factors (SIF) are converted to J-integrals by the plane strain relationship,

$K_{Jc} = \sqrt{\frac{J_{0.1} \cdot E}{1000(1-\upsilon^2)}}$, where K_{Jc} is upper shelf region toughness, in ksi·√inch, and $J_{0.1}$ is the J integral resistance at Δa = 0.1 inch.

4.2 APPLIED J-INTEGRAL RESULTS AND COMPARISON WITH J-R CURVES ALLOWABLES

As discussed in Section 2.1, the cooldown transient is evaluated for Levels A/B. FEA through-wall stress profiles were fitted to 3rd order polynomials, and A-3200 of [8] was used for the calculation of K_{It} and K_{Ip} instead of the generic closed-form solution in Appendix K of [8]. As discussed in the methodology Section 4.1, this is more accurate and is an NRC approved method. Unit pressure (1 ksi) FEA stress profiles were scaled to pressure transients and K_{Ip} was then calculated in the same manner as K_{It} using the 3rd order polynomial method. The crack face pressure was applied as discussed in Section 4.1.1. As discussed in Section 4.1.1, the double counting of the plastic zone correction was removed by setting the q_y term in A-3200 of [8] to zero. The plastic correction was accounted for in the a_e term per K-4210 of [8].

4.2.1 Nozzle-to-Shell Welds Level A/B

The $J_{applied}$ values for a 0.1 inch flaw extension with pressure SF = 1.15 and SF =1.25 for Levels A/B are contained in Table 4-1. The $J_{applied}$ at 0.1-inch flaw extensions (J_1) for both inlet and outlet nozzle welds are below the J-R $J_{0.1}$ = [ ][f], per Table 3-6. Therefore, the acceptance criteria in ASME Section XI, K-2200 (a)(1) [8] is satisfied. As shown in Figure 4-2 and Figure 4-3, the slope of $J_{applied}$ is less than the J-R curve at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-2200 (a)(2) [8] is satisfied.

Table 4-1: Inlet and Outlet Nozzle Welds Levels A/B, Circumferential Flaw, $J_{applied}$

[ ] a, c, e, f

a, c, e, f

Figure 4-2: Inlet Nozzle Weld, Circumferential Flaw, Levels A/B $J_{applied}$ vs. J-R, SF=1.25

[ ]$^{a, c, e, f}$

Figure 4-3: Outlet Nozzle Weld, Circumferential Flaw, Levels A/B $J_{applied}$ vs. J-R, SF=1.25

4.2.2 Nozzle-to-Shell Welds Level D

As discussed in Section 2.1, there is no applicable Level C transient. The Level D transient is the SLB. Per [2], the B&WOG model 6-B J-R mean curve is used for Level D loading as a best estimate representation of the toughness for the North Anna Unit 1 and 2 Rotterdam welds. The mean J-R curve values for the Rotterdam welds are listed in Table 4-3. Values of K_{Ic} and K_{Ic}^2/E' as a function of temperature are contained in Table 4-4. Temperature at which the mean J-R curve intersects K_{Ic}^2/E' is [ ]^{f}, establishing the start of the upper shelf temperature range.

Figure 4-4 to Figure 4-7 combine the SLB transient $J_{applied}$, J-R and K_{Ic}^2/E' curves as follows:

1. The B&WOG Model 6-B mean J-R curve is used.

2. Figure 4-4 and Figure 4-6 present $J_{applied}$ curves due to the Level D SLB transient, the mean J-R curve and K_{Ic}^2/E' curves as a function of crack tip temperature. The $J_{applied}$ curve is truncated at temperature point of [ ]^{f} (limiting or lowest temperature for upper shelf toughness. Figure 4-4 illustrates the inlet nozzle location case. Figure 4-6 illustrates the outlet nozzle location case. All points of the transient remain below the mean J-R curve.

3. Figure 4-5 and Figure 4-7 show the $J_{applied}$ and the mean J-R curve [ ]^{f} as a function of crack extension. Figure 4-5 presents the inlet nozzle $J_{applied}$ at temperature of the crack tip temperature of [ ]$^{a, c, e}$. Figure 4-7 shows the outlet nozzle $J_{applied}$ at the crack tip temperature of [ ]$^{a, c, e}$. These temperatures are conservative since $J_{applied}$ is lower at the lowest upper shelf temperature. The slope of the $J_{applied}$ is less than the slope of the mean J-R curve at the point of intersection, which demonstrates that the flaw is stable as required by ASME Section XI, Appendix K, K-3400.

The $J_{applied}$ values at a 0.1 inch flaw extension with SF = 1 for Level D are contained in Table 4-2. Since the 1/10 of the wall thickness plus cladding exceeded 1 inch for all evaluated locations, the postulated flaw depth is 1 inch. As shown in Table 4-2, all applied J_1 for the nozzle-to-shell welds are below the $J_{0.1}$ of [ ]^{f}. The acceptance criteria in ASME Section XI, K-2400 (a) [8] is satisfied.

Table 4-2: Inlet and Outlet Nozzle Welds Level D, Circumferential Flaw, Limiting $J_{applied}$

[ ]$^{a, c, e, f}$

Table 4-3: B&WOG Model 6-B Mean J-R Curve with Δa = 0.1 inch

[ ] f

Table 4-4: K_{Ic}^2/E' Curve

[ ] a, c, e, f

a, c, e, f

Figure 4-4: Inlet Nozzle-to-Shell Weld, J-integral vs. Temperature

a, c, e, f

Figure 4-5: Inlet Nozzle-to-Shell Weld, J-integral vs. Flaw Extension

a, c, e, f

Figure 4-6: Outlet Nozzle-to-Shell Weld, J-integral vs. Temperature

a, c, e, f

Figure 4-7: Outlet Nozzle-to-Shell Weld, J-integral vs. Flaw Extension

Additionally, as discussed in Section (a), K-5300(b) also requires that the remaining ligament is not subject to tensile instability. Table 4-5 contains the necessary inputs and a sample calculation for tensile instability pressure using a flaw depth a=1.181 inch. Additionally, a range of flaw depths from 0.098 to 1.968 inches were calculated to be in excess of 10 ksi, which is significantly greater than the 2.5 ksi pressure expected during a SLB transient.

Table 4-5: K-5300 Tensile Instability Check for RV Nozzle-to-Shell Weld

[ ]a, c, e, f

4.2.3 Upper Shell Forging Level A/B

The $J_{applied}$ values for a 0.1 inch flaw extension with pressure SF = 1.15 and SF =1.25 for Level A/B are contained in Table 4-6. The J-R $J_{0.1}$ values from Table 3-7 are interpolated to the respective actual crack tip temperatures of [ ]a, c, e in Table 4-6 to instead of using a conservative $J_{0.1}$ from a higher temperature. Both the circumferential and axial flaw applied J_1 are below the $J_{0.1}$ at their respective temperatures. Therefore, the acceptance criteria in ASME Section XI, K-2200 (a)(1) [8] is satisfied. As shown in Figure 4-8 and Figure 4-9, the slope of $J_{applied}$ is less than the J-R curve at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-2200 (a)(2) [8] is satisfied. Since the upper shell forging FEA stresses were taken from the thicker upper shell to the thinner intermediate shell region, it captured the stress concentration effect, therefore, the upper shell forging results are also applicable to the intermediate shell forgings.

Table 4-6: Upper Shell Forging Level A/B, Circumferential Flaw, Limiting $J_{applied}$

[ ]a, c, e

a, c, e

Figure 4-8: RV Upper Shell, Circumferential Flaw, Level A/B $J_{applied}$ vs. J-R, SF=1.25

a, c, e

Figure 4-9: RV Upper Shell, Axial Flaw, Level A/B $J_{applied}$ vs. J-R, SF=1.25

4.2.4 Upper Shell Forging Level D

The upper shell forging $J_{applied}$ values for a 0.1 inch flaw extension with SF = 1 Level D are contained in Table 4-7. The J-R $J_{0.1}$ at 600°F from Table 3-7 are used for conservatism. Both the circumferential and axial flaw applied J_1 are below the $J_{0.1}$. Therefore, the acceptance criteria in ASME Section XI, K-2400 (a) [8] is satisfied. As shown in Figure 4-10 and Figure 4-11, the slope of $J_{applied}$ is less than the J-R curve at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-3400 [8] is satisfied. As shown in Table 4-8, the Level D, SLB transient internal pressure of 2.5 ksi is significantly less than the tensile instability pressures calculated per K-5300; therefore, the remaining ligament is not subjected to tensile instability. Since the upper shell forging FEA stresses were taken from the thicker upper shell to the thinner intermediate shell region, it captured the stress concentration effect, therefore, the upper shell forging results are also applicable to the intermediate shell forgings.

Table 4-7: Upper Shell Forging Level D, Limiting $J_{applied}$

[ ]a, c, e

[ ]a, c, e

Figure 4-10: RV Upper Shell, Circumferential Flaw, Level D $J_{applied}$ vs. J-R

[ ]a, c, e

Figure 4-11: RV Upper Shell, Axial Flaw, Level D $J_{applied}$ vs. J-R

Table 4-8: K-5300 Tensile Instability Check for RV Upper Shell

[ ]a, c, e

4.2.5 Nozzle Forging Level A/B

The nozzle corner is the limiting location for the nozzle forging due to the wall thickness and the stress concentration effect. The $J_{applied}$ values for a 0.1 inch flaw extension with pressure SF = 1.15 and SF =1.25 for Level A/B are contained in Table 4-9. All the applied J_1 values are less than the respective $J_{0.1}$ from Table 3-6. Therefore, the acceptance criteria in ASME Section XI, K-2200 (a)(1) [8] is satisfied. As shown in Figure 4-12 to Figure 4-15, the slope of $J_{applied}$ is less than the J-R curve at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-2200 (a)(2) [8] is satisfied.

Table 4-9: Nozzle Corner Level A/B, Limiting $J_{applied}$

a, c, e

a, c, e

Figure 4-12: Inlet Nozzle Corner, Circumferential Flaw, Level A/B $J_{applied}$ vs. J-R, SF=1.25

[ ] a, c, e

Figure 4-13: Outlet Nozzle Corner, Circumferential Flaw, Level A/B $J_{applied}$ vs. J-R, SF=1.25

a, c, e

Figure 4-14: Inlet Nozzle Corner, Axial Flaw, Level A/B $J_{applied}$ vs. J-R, SF=1.25

[ ]$^{a, c, e}$

Figure 4-15: Outlet Nozzle Corner, Axial Flaw, Level A/B $J_{applied}$ vs. J-R, SF=1.25

4.2.6 Nozzle Forging Level D

The $J_{applied}$ values for a 0.1 inch flaw extension for Level D are contained in Table 4-10. All the applied J_1 for the nozzle forgings are less than the $J_{0.1}$ of [ ]$^{a, c, e}$ from Table 3-6. Therefore, the acceptance criteria in ASME Section XI, K-2400 (a) [8] is satisfied. As shown in Figure 4-16 to Figure 4-19, the slope of $J_{applied}$ is less than the J-R curve at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-3400 [8] is satisfied. As shown in Table 4-11, the Level D internal pressure of 2.5 ksi is significantly less than the tensile instability pressures calculated per K-5300, therefore, the remaining ligament is not subjected to tensile instability.

Table 4-10: Nozzle Corner Level D, Limiting $J_{applied}$

a, c, e

a, c, e

Figure 4-16: Inlet Nozzle Corner, Circumferential Flaw, Level D $J_{applied}$ vs. J-R, SF=1.0

a, c, e

Figure 4-17: Outlet Nozzle Corner, Circumferential Flaw, Level D $J_{applied}$ vs. J-R, SF=1.0

a, c, e

Figure 4-18: Inlet Nozzle Corner, Axial Flaw, Level D $J_{applied}$ vs. J-R, SF=1.0

a, c, e

Figure 4-19: Outlet Nozzle Corner, Axial Flaw, Level D $J_{applied}$ vs. J-R, SF=1.0

Table 4-11: K-5300 Tensile Instability Check for RV Nozzle Corner

a, c, e

5 CONCLUSIONS

The North Anna Units 1 and 2 RV nozzle-to-shell welds, nozzle forgings and upper shell forgings were evaluated for equivalent margins of safety per ASME Code Section XI [8]. The flaw extension and stability criteria of ASME Section XI, Appendix K are satisfied.

Levels A/B

For all evaluated locations, the $J_{applied}$ at 0.1-inch flaw extension with a structural factor (SF) of 1.15 for pressure and SF of 1.0 for thermal are less than the J-material at the 0.1-inch flaw extension. Therefore, the acceptance criteria in ASME Section XI, K-2200 (a)(1) [8] is satisfied. The slope of $J_{applied}$ (SF=1.25) is less than the J-material (J-R curve) at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-2200 (a)(2) [8] is satisfied.

Level D

For all evaluated locations, the $J_{applied}$ at 0.1-inch flaw extension with a SF of 1.0 are less than the J-R at the 0.1-inch flaw extension. Therefore, the acceptance criteria in ASME Section XI, K-2400 (a) [8] is satisfied. The slope of $J_{applied}$ is less than the J-R curve at the intersection of both curves (i.e., $J_{applied}$ = J-R). Therefore, the stability acceptance criteria in ASME Section XI, K-3400 [8] is satisfied. All flaws evaluated for Level D assumed 1/10 of the wall thickness (including cladding and limited to 1 inch) plus a 0.1 inch flaw extension. The results demonstrate that flaw growth is stable at less than 75% of the wall thickness since the applied J-integral intersects the mean J-integral resistant curve at a small fraction of the thickness. This satisfies the 75% of wall thickness requirement per K-2400 (c). Additionally, the maximum Level D internal pressure is less than the tensile instability pressures calculated per K-5300 (b) for all evaluated locations and flaws.

6 REFERENCES

1. BAW-2192, Revision 0, Supplement 1P-A, 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, 2018.

2. BAW-2192, Revision 0, Supplement 2P, 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.”

3. NUREG/CR-5729, “Multivariable Modeling of Pressure Vessel and Piping J-R Data,” May 1991.

4. Code of Federal Regulations, 10 CFR Part 50, Appendix G, “Fracture Toughness Requirements,” U.S. Nuclear Regulatory Commission, Washington D.C., Federal Register, Volume 77, No. 14, January 23, 2012.

5. North Anna Power Station Updated Final Safety Analysis Report, Revision 55.

6. Regulatory Guide 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb,” U.S. Nuclear Regulatory Commission, June 1995.

7. ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, with Addenda up to and including Winter 1968.

8. ASME Boiler and Pressure Vessel Code, Section XI, 2013 Edition.

9. Fracture Analysis of Vessels – Oak Ridge FAVOR, v05.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations, ORNL/NRC/LTR-05/18 (ADAMS Accession Number ML063350323).

10. API 579-1/ASME FFS-1, “Fitness-For-Service,” Annex C, “Compendium of Stress Intensity Factor Solutions,” June, 2016.

11. S. A. Delvin and P. C. Riccardella, “Fracture Mechanics Analysis of JAERI Model Pressure Vessel Test,” ASME Paper No. 78-PVP-91, Proceedings of the 1978 ASME Pressure Vessels and Piping Conference, June 25-30, 1978, Montreal, Quebec, Canada.

12. NRC SE Report, “Safety Assessment of Report WCAP-13587, Revision 1, “Reactor Vessel Upper Shelf Energy Bounding Evaluation For Westinghouse Pressurized Water Reactors,” September 1993”, April 21, 1994.

13. NRC SE Report, "Final Safety Evaluations for BAW-2192, Supplement 1NP, Revision, "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 1NP, 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," ML19106A196, April 29, 2019.

APPENDIX A : CORRESPONDENCE WITH U.S. NRC REGARDING THE REVIEW OF PWROG-19047-P, REVISION 0

PWROG
PWR Owners Group

Program Management Office
20 International Drive
Windsor, Connecticut 06095

PWROG-19047-P/NP, Revision 0
Project Number 99902037

May 27, 2019

OG-20-167

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

Subject: PWR Owners Group
Submittal of PWROG-19047-P/NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (PA-MSC-1481)

Reference 1: NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991.

The purpose of this letter is to submit PWROG-19047-P/NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis." to support the North Anna Units 1 and 2 Subsequent License Renewal (SLR) application as requested during the meeting held on April 9, 2020, between Dominion and the NRC (Accession Number ML20104A039 contains the NRC meeting summary).

The purpose of this topical report (TR) is to document the equivalent margins analysis (EMA) for the North Anna Units 1 and 2 reactor vessel (RV) inlet and outlet nozzle Rotterdam welds, nozzle forgings and nozzle belt forgings (a.k.a., upper shell forgings). These locations were chosen due to the potential that their upper-shelf energy (USE) maybe less than the 50 ft-lb limit at 80-years (72 EFPY) for SLR. Two forging locations were determined to be less than or equal to the 50 ft-lb limit at 80-years (72 EFPY) and the equivalent margin analysis (EMA) for these locations utilized the multivariable model for RPV base metal contained in Reference 1.

The PWROG requests that the NRC review the TR for the two forging locations that were determined to be less than or equal to the 50 ft-lb limit at 80-years (72 EFPY).

As Enclosures 1 and 2 contain information proprietary to Westinghouse Electric Company LLC ("Westinghouse") and Framatome Inc., the information contained herein is supported by two Affidavits: one each signed by Westinghouse and Framatome Inc., the owners of the information. The Affidavits set forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of 10CFR Section 2.390 of the Commission's regulations.

U.S. Nuclear Regulatory Commission May 27, 2020
Document Control Desk Page 2 of 3
OG-20-167

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse and/or Framatome Inc. be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations. Each affidavit should be consulted to identify the appropriate justifications for withholding of the respective information.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-20-5039 and should be addressed to Camille T. Zozula, Manager, Infrastructure & Facilities Licensing, Westinghouse Electric Company, 1000 Westinghouse Drive, Suite 165, Cranberry Township, Pennsylvania 16066.

Correspondence with respect to the copyright or proprietary aspects of the item listed above or the supporting Framatome Inc. Affidavit should be addressed to Philip Opsal, OF-34, Framatome Inc., 3315 Old Forest Road, Lynchburg, VA 24501.

TR Classification: As discussed above, this TR addresses the multivariable modeling of the Pressure Vessel per Reference 1 for the two forging locations at or below the 50 ft-lb limit at 80-years (72 EFPY). Additionally, the J-integral resistance Model 6B is used for some locations where the upper shelf energy was determined to be greater than 50 ft-lbs. These were evaluated proactively in this EMA for asset management considerations.

Specialized Resource Availability: This TR is 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.

This letter transmits PWROG-19047-P Revision 0 (Enclosure 1), and PWROG-19047-NP (Enclosure 2). Notarized Affidavits for Withholding proprietary information are provided as Enclosures 3 and 4.

Applicability: This TR is applicable to the reactor vessels for North Anna Units 1 and 2 add discussed in the TR.

The PWROG requests that the NRC complete their review of the TR by August 15, 2020.

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 172
Cranberry Township, PA 16066

U.S. Nuclear Regulatory Commission
Document Control Desk
OG-20-167

May 27, 2020
Page 3 of 3

If you have any questions, please do not hesitate to contact me at (602) 999-2080 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Michael Powell, Chief Operating Officer and Chairman
PWR Owners Group

MP:JPM:am

Enclosure 1: (One Copy) PWROG-19047-P, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (Proprietary)
Enclosure 2: (One copy) PWROG-19047-NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (non-proprietary)
Enclosure 3: Westinghouse Affidavit for Withholding Proprietary Information, CAW-20-5039
Enclosure 4: Framatome Affidavit for Withholding Proprietary Information

cc: PWROG PMO
PWROG Steering and Management Committee
L. Fields, US NRC
C. Tomes, Dominion
P. Aitken, Dominion
J. Andrachek, Westinghouse
T. Zalewski, Westinghouse
G. Hall, Westinghouse
S. Rigby, Westinghouse
D. Page Blair, Framatome
N. Ashok, Framatome
M. Rinckel, Framatome

Electronically Approved Records are Authenticated in PRIME

PWROG
PWR Owners Group

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

PWROG-19047-P/NP, Revision 0
Docket 99902037
Project 694

December 3, 2020

OG-20-284

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 1-7 Associated with PWROG-19047-P/NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis" (PA-MSC-1481)

References:

1. Letter OG-20-167, Submittal of PWROG-19047-P/NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis", PA-MSC-1519, dated May 27, 2020

2. Email from the NRC (Fields) to the PWROG (Holderbaum), Request for Additional Information, RAIs 1-7, RE: PWROG-19047-P/NP, Revision 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis", dated September 25, 2020

On May 27, 2020, 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 PWROG-19047-P & NP, Revision 0 for referencing in regulatory actions (Reference 1). The NRC Staff has determined that additional information is needed to complete the review per the email dated September 25, 2020 (Reference 2).

Enclosure 1 to this letter provides formal responses to NRC RAIs 1-7 (Reference 2) associated with PWROG-19047-P/NP, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis".

U.S. Nuclear Regulatory Commission December 3, 2020
OG-20-284 Page 2 of 2

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 (434) 832-2382 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Michael Powell
Chief Operating Officer & Chairman
Pressurized Water Reactor Owners Group

MP:JPM:am

cc: PWROG PMO
C. Tomes, DOM
D. Knee, DOM
P. Aitken, DOM
T. Hanna, DOM
M. Post, DOM
L. Fields, US NRC
J. Andrachek, Westinghouse
T. Zalewski, Westinghouse
G. Hall, Westinghouse
S. Rigby, Westinghouse

Enclosure 1: LTR-SDA-20-082, Revision 0 (nonproprietary), RAIs 1-7 Responses for PWROG-19047-P/NP, Revision 0 (PA-MSC-1481)

Electronically Approved Records are Authenticated in PRIME

To: Thomas Zalewski
cc: Stephen P. Rigby
From: Gordon Z. Hall
Ext: (860) 731-6114

Date: November 19, 2020
Your ref: N/A
Our ref: LTR-SDA-20-082, Rev. 0

Subject: **Westinghouse Response to U.S. NRC Request for Additional Information (RAI) for PWROG-19047-P/NP, Rev. 0**

References:

1. U.S. NRC Request for Additional Information for Pressurized Water Reactor Owners Group Topical Report, PWROG-19047-P/NP, Revision, 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis," Docket No. 99902037; EPID L-2020-TOP-0028.
2. Topical Report, BAW-2192, Revision 0, Supplement 1P-A, 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 2018.
3. ASME Boiler and Pressure Vessel Code, Section XI, 2013 Edition.
4. Topical Report, PWROG-18005-NP, Rev. 2, "Determination of Unirradiated RT_{NDT} and Upper-Shelf Energy Values of the North Anna Units 1 and 2 Reactor Vessel Materials," September 2019.
5. Topical Report, PWROG-17090-NP-A, Rev. 0, "Generic Rotterdam Forging and Weld Initial Upper-Shelf Energy Determination," January 2020. [ADAMS Accession Number ML20024E238]
6. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988. [ADAMS Accession Number ML003740284]
7. Westinghouse Report, WCAP-18363-NP, Rev. 1, "North Anna Units 1 and 2 Heatup and Cooldown Limit Curves for Normal Operation," March 2020.
8. Westinghouse Report, WCAP-18364-NP, Rev. 1, "North Anna Units 1 and 2 Time-Limited Aging Analysis on Reactor Integrity for Subsequent License Renewal (SLR)," March 2020.
9. Westinghouse Report, WCAP-18015-NP, Rev. 2, "Extended Beltline Pressure Vessel Fluence Evaluations Applicable to North Anna 1 & 2," September 2018.
10. U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001. [ADAMS Accession Number ML010890301]
11. Westinghouse Report, WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves WOG Programs: MUHP-5073 MUHP-3073," August 2005.
12. North Anna Power Station Units 1 and 2 Application for Subsequent License Renewal, August 2020. [ADAMS Accession Number ML20246G696]
13. Topical Report, PWROG-19047-P, Rev. 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis," May 2020.
14. NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," May 1991.
15. Westinghouse Report, WCAP-16333-NP, Rev. 0, "Fracture Toughness Testing of Compact Tension Specimens from Watts Bar Unit 1 Surveillance Capsule X," October 2004.

Background

By letter dated May 27, 2020 (Agencywide Document Access and Management System (ADAMS) Accession No. ML20149K668) the Pressurized Water Reactor Owner's Group (PWROG) submitted Topical Report (TR) PWROG-19047-Proprietary (P)/Non-Proprietary (NP), Revision (Rev.) 0, "North Anna Units 1 and 2 Reactor Vessels Low Upper-Shelf Fracture Toughness Equivalent Margin Analysis," ((ADAMS) Accession No. ML20149K670) for NRC review and approval.

The purpose of the TR is to document the equivalent margins analysis (EMAs) for the North Anna Units 1 and 2 reactor vessel (RV) inlet and outlet nozzle Rotterdam welds, nozzle forgings and nozzle belt forgings (a.k.a., upper shell forgings) to demonstrate compliance with Section IV.A.1 of Appendix G to 10 CFR Part 50 "Fracture Toughness Requirements." These locations were chosen for their upper-shelf energy (USE) potentially falling below the 50 ft-lb limit at 80-years (72 EFPY) for subsequent license renewal (SLR). The NRC staff (the staff) determined it needs additional information to complete its review of the TR.

Regulatory Basis

Title 10 of the Code of Federal Regulations (10 CFR), Part 50.60, "Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation," is the governing regulation for Reactor Vessel Charpy Upper-Shelf Energy of the reactor vessel. 10 CFR 50.60 imposes fracture toughness and material surveillance program requirements, which are set forth in Appendices G, "Fracture Toughness Requirements, " and H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50.

Section IV.A.1.a of Appendix G to 10 CFR Part 50 sets forth requirements that the 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, 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. Furthermore, 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. Section IV.A.1.c of Appendix G to 10 CFR Part 50 sets forth requirements that the analysis for satisfying the requirements of section IV.A.1 of Appendix G 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. The requirements in these sections of Appendix G to 10 CFR Part 50 and the ASME Code, therefore, form the regulatory bases for the staff's requests for additional information (RAIs).

REQUESTS FOR ADDITIONAL INFORMATION:

RAI 01

Issue

Section 2.1, "Regulatory Requirements," of PWROG-19047-P states, in part, the cooldown transient for North Anna Units 1 and 2 with a constant pressure of 2750 psia assumed throughout the transient bounds all Levels A/B conditions. Further, it states that the Level C/D transient selection is based on the guidance in Regulatory Guide 1.161, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb," (RG 1.161) and explained that the Level D transient is the steam line break (SLB). Section 4.1 of PWROG-19047-P indicates that only the SLB transient is specified for Level D conditions.

The staff noted that K-1100 of Appendix K to Section XI of the ASME Code states, in part, that "[a]ll specified design transients for the reactor vessel shall be considered." The staff reviewed North Anna Updated Final Safety Analysis Report Table 5.2-4 (ADAMS Accession No. ML11287A241) and noted that beyond the "steam pipe break" transient there are additional Service Level D conditions (i.e., Faulted Conditions) that are applicable to the current licensing basis for North Anna, Units 1 and 2, such as the "Main reactor coolant pipe break" and "Design-basis earthquake." Section 4 of RG 1.161 states that selection of the limiting transients for Service Levels C and D is a key aspect of evaluating the integrity of reactor pressure vessels that contain materials with Charpy upper-shelf energy less than 50 ft-lb.

The staff determined that it is not clear how the consideration of only the cooldown transient with a constant pressure of 2750 psia assumed throughout the transient, and the SLB transient satisfies K-1100 of Appendix K to Section XI of the ASME Code.

NRC Request

a) *Provide the justification for the selection of the cooldown transient for North Anna, Units 1 and 2, with a constant pressure of 2750 psia assumed throughout the transient satisfies K-1100 of Appendix K to Section XI of the ASME Code for Service A and B transients.*
b) *Provide the justification for the selection of the SLB transient for North Anna, Units 1 and 2, satisfies K-1100 of Appendix K to Section XI of the ASME Code for Service D transients.*

Response

(a) The ASME Section XI, K-1100 text regarding "design transients for the reactor vessel" is interpreted to mean the design transients defined by the ASME Section III Certified Design Specification. Considering the normal and upset conditions listed in the North Anna UFSAR, Table 5.2-4, the cooldown transient for North Anna Units 1 and 2 is chosen to bound all Levels A/B service loading conditions with a constant accumulation pressure of 2750 psia, i.e. 1.1 times design pressure per ASME Section XI, K-1300 and K-4220 [3]. Additionally, the plant cooldown transient with a 100°F/hour cooldown rate is consistent with K-4210 (c), where the simplified equation of thermal load, K_{It} using a CR (cooling rate) of 100°F/hour for Level A/B. The above approach is consistent with BAW-2192, Revision 0, Supplement 1P-A, Revision 0 [2].

(b) The Level C/D transient selection is based on the guidance in Regulatory Guide 1.161, Section 4, "TRANSIENT SELECTION." Plant-specific transient is used for the North Anna EMA. There is no applicable emergency (Level C) transient defined in the Westinghouse RV design specification, and the only Level D transient is the steam line break (SLB). As discussed in UFSAR Section 5.2.4, the North Anna Units 1 and 2 Leak-Before-Break (LBB) analysis was performed on the main coolant loop piping and was approved by the NRC on August 31, 1999. Therefore, a main reactor coolant pipe break need not to be considered. However, for conservatism, the Level D condition included the pipe rupture and Safe Shutdown Earthquake (SSE) loads on the nozzle and support pad for the EMA.

RAI 02

Issue

Section 3.2, "J-Integral Resistance Models" of PWROG-19047-P states that "the maximum reported copper content and the fluence value at the nozzle-to-shell weld is utilized" as summarized in Table 3-3 of PWROG-19047-P. However, the staff noted that the supporting reference from the current licensing basis for this assumption for copper content at the nozzle-to-shell weld and a description of the methodology for determining the neutron fluence at the nozzle-to-shell weld was not provided.

NRC Request

a) *Provide a reference to the current licensing basis document that supports the copper content value used in PWROG-19047-P that demonstrates its applicability for the nozzle-to-shell welds at North Anna, Units 1 and 2.*

b) *Provide the basis for the neutron fluence value used in PWROG-19047-P to bound the nozzle-to-shell welds for North Anna, Units 1 and 2.*

   i. *Provide an explanation for the methodology used to determine this projected neutron fluence value for the North Anna, Units 1 and 2, nozzle-to-shell welds at 80-years (72 Effective Full Power Years (EFPY) for subsequent license renewal).*

   ii. *Describe how this methodology either (1) adheres to the guidance contained NRC Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" (ADAMS Accession No. ML010890301); or (2) is otherwise suitably accurate or bounding for the downstream modeling. Furthermore, provide additional justification regarding how the calculational methods used to estimate fluence were suitable for application to the nozzle elevation.*

Response

(a) The reference for the copper content is PWROG-18005-NP, Rev. 2 [4]. Please refer to Table 9 and 10 in PWROG-18005-NP. Note that the Cu value is the generic value in Regulatory Guide 1.99, Rev. 2 [6]. The weld heat and type for these nozzle-to-shell welds could not be determined for North Anna Unit 1, only that these welds were fabricated by Rotterdam. Consistent with the NRC approved PWROG-17090-NP-A [5], when the weld heat and type are not known for a Rotterdam fabricated weld, the generic value from Regulatory Guide 1.99 can be used for Cu content.

(b) (i)

The reactor vessel fluence values utilized for the RVI evaluations in WCAP-18363-NP [7] and WCAP-18364-NP [8] are from WCAP-18015-NP [9] which uses DORT transport models based on methodologies consistent with the guidance in Regulatory Guide 1.190 [10]. These methodologies have been approved by the U.S. Nuclear Regulatory Commission (NRC) and are discussed in detail in WCAP 14040-A, Rev. 4 [11].

(ii)

Regulatory Guide 1.190 [10] contains an allowable neutron fluence uncertainty of ± 20% for the reactor vessel beltline region, but it does not define an allowable neutron fluence uncertainty for the extended beltline region of the reactor vessel. Currently, there is no NRC regulatory guidance for calculating reactor pressure vessel extended beltline fluences. Therefore, the same methodology that was used for the beltline region was used for the extended beltline region. In order to demonstrate the acceptability of the results, a margin assessment was performed and summarized in the Section 4.2.1 of the North Anna SLR Application [12]. The assessment concluded the following with respect to the EMA that:

- The maximum projected fluence values of the North Anna Units 1 and 2 reactor vessel inlet/outlet nozzles (3.14×10^{17} n/cm^2 corresponding to the nozzle-to-shell weld location) could increase by 60% before exceeding the fluence value utilized in the EMA for SLR (5×10^{18} n/cm^2). As shown in Tables 4.2.2-3 and 4.2.2-4 of the North Anna SLR Application [12], the limiting USE nozzle for both Units 1 and 2 has a projected fluence value of 1.20×10^{17} n/cm^2 and would need to increase by a factor of greater than 4 prior to reaching the fluence utilized in the EMA.
- As shown in Tables 4.2.1-1 and 4.2.1-2 of the North Anna SLR Application [12], the maximum projected fluence values of the North Anna Units 1 and 2 inlet and outlet nozzle to shell welds (3.14×10^{17} n/cm^2) would need to increase by 3 times the current value before exceeding the fluence value utilized in the EMA for SLR (1×10^{18} n/cm^2).

RAI 03

Issue
*Section 3.2, "J-Integral Resistance Models" of PWROG-19047-P states the J-R curves for the RV nozzle and shell forgings were developed in accordance with NUREG/CR-5729, "Multivariable Modeling of Pressure Vessel and Piping J-R Data," Table 11, Charpy model **without fluence**. Further, it states that the Charpy model is applicable to North Anna nozzle and shell forgings because they are **bounded** by the range of explanatory variables (**fluence**, copper content, etc.) used to develop the J-R model. **[emphasis added]***

The staff noted that NUREG/CR-5729 indicates that only two types of models were developed for the reactor pressure vessel base metals database (i.e., a Charpy model and a pre-irradiation Charpy (CVNp) model) and since copper content (Cu) was not measured for much of the base metal database, a Cu-fluence model was not investigated. Furthermore, Table 11 of NUREG/CR-5729 does not appear to incorporate Cu and neutron fluence as variables for the Charpy Model for RPV Base Metals. Given this discrepancy in PWROG-19047-P, it is not clear to the staff how the PWROG determined that the North Anna, Units 1 and 2, nozzle and shell forgings are bounded by the explanatory variables used in the development of the model it selected from NUREG/CR-5729.

NRC Request
a) Clarify the discrepancy in PWROG-19047 with regard to the explanatory variables (e.g., copper and fluence) of NUREG/CR-5729 model that are used in the EMAs for North Anna, Units 1 and 2.

*b) Provide the basis (with reference to the applicable current licensing basis documents) that the North Anna, Units 1 and 2, nozzle and shell forgings are **bounded** by the range of explanatory variables used to develop the J-R model (i.e., Table 11 of NUREG/CR-5729).*

Response

As stated in Section 3.2 of PWROG-19047 [13], the NUREG/CR-5729 [14], Table 11, Charpy model was used. This model does not include fluence and Cu as input parameters directly. However, values bounding the projected 80-year USE values contained in WCAP-18364-NP [8] were used for the CVN (Charpy) input. Note that WCAP-18364-NP was submitted to NRC for the North Anna SLR application. The projected 80-year values calculated in WCAP-18364-NP [8] include fluence and Cu as inputs consistent with Regulatory Guide 1.99 [6]. The projected SLR USE values are shown in Table 1 compared to the CVN values used to calculate the J-R curves using the NUREG/CR-5729, Table 11 Charpy model.

Table 1: Projected USE vs. CVN for Charpy Model J-R

Reactor Vessel Material	Limiting Projected SLR USE WCAP-18364-NP [8]		CVN values used to calculate J-R curves used in PWROG-19047-P [13] (ft-lbs)
	Unit 1 (ft-lbs)	Unit 2 (ft-lbs)	
Inlet and Outlet Nozzle Forgings	50.0	53.1	48.2
Intermediate Shell Forgings	63.7	48.2	47.5

The North Anna Units 1 and 2 reactor pressure vessels (RPV) were fabricated by Rotterdam Dockyard Company. Measured J-R curves for two other forgings from Rotterdam fabricated RPVs were compared to the NUREG/CR-5729, Table 11 Charpy model estimated J-R curves to validate the model for Rotterdam forgings. The Watts Bar Unit 1 surveillance forging from Capsules W and X was J-R tested and reported in WCAP-16333-NP [15]; the normalized results are shown in Figure 1. The predicted USE values for Watts Bar Unit 1 Capsules W and X determined in accordance with Regulatory Guide 1.99 [6] are 45.9 ft-lbs and 44.0 ft-lbs respectively. The predicted NUREG/CR-5729, Table 11 Charpy model at standard conditions is also shown in Figure 1. The standard conditions and normalization procedure defined in NUREG/CR-5729 was used. An unirradiated nozzle dropout A508 Class 2 forging from a different Rotterdam fabricated RPV was also J-R tested by Westinghouse in 1992. This forging had a USE of 100 ft-lbs. The predicted NUREG/CR-5729, Table 11 Charpy model at standard conditions is shown in Figure 2 and Figure 3, which are compared to the normalized 300°F and 550°F test data respectively. Based on the comparison of measured data from Rotterdam forgings to the NUREG/CR-5729, Table 11 Charpy model used for estimating J-R curves in PWROG-19047, the model adequately represents the measured data, therefore, the model is deemed acceptable to be used in the analysis. The irradiated Rotterdam forging has a Cu value (0.155%) that is representative of the North Anna Units 1 and 2 forgings and has the highest fluence available for a J-R tested Rotterdam forging in the U.S. The USE values (44.0, 45.9 and 100 ft-lbs) used as inputs to check the Charpy model against measured J-R data bound those used in the EMA.

The statement in the RAI can be clarified as follows "The Charpy model is applicable to the North Anna Units 1 and 2 nozzle and shell forgings because the J-R model was validated against representative Rotterdam forgings with measured J-R data." The Topical Report PWROG-19047-P (and -NP) will be revised as such.

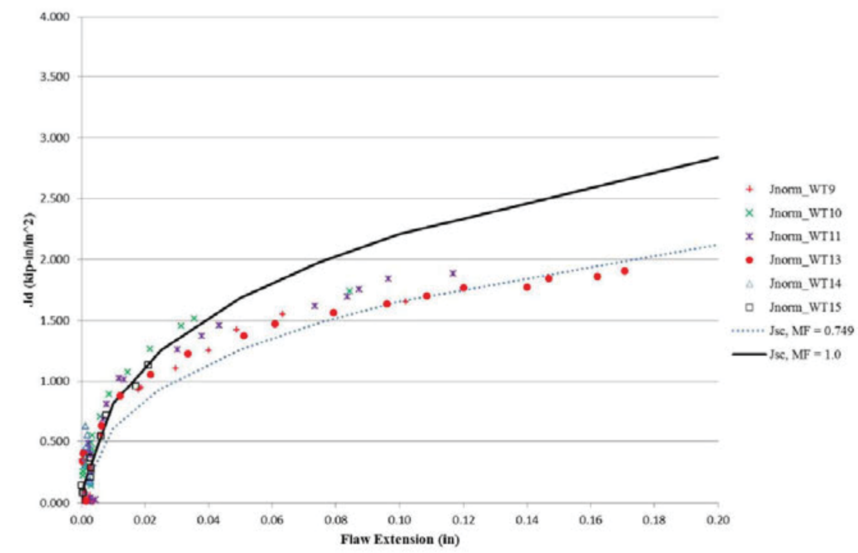

Figure 1: Charpy Model J-integral at Standard Conditions Vs. Normalized Watts Bar Unit 1 Data

Westinghouse Non-Proprietary Class 3

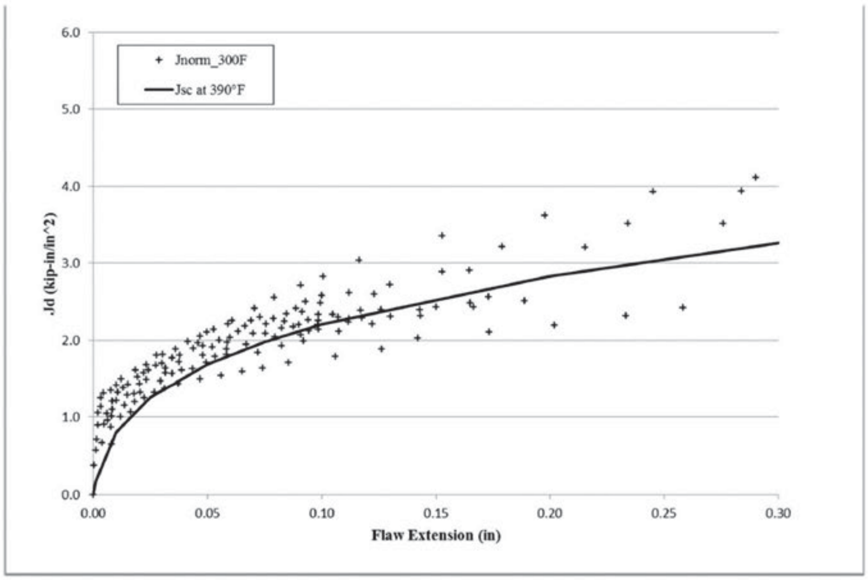

Figure 2: NUREG/CR-5729 Charpy Model vs. Normalized Measured Data at 300°F

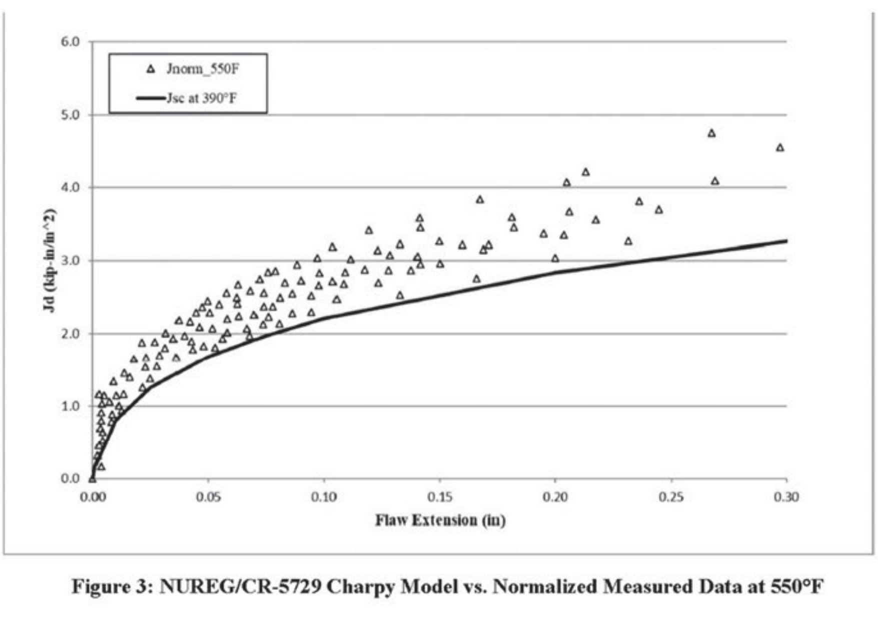

Figure 3: NUREG/CR-5729 Charpy Model vs. Normalized Measured Data at 550°F

RAI 04

Issue

Section 1, "Introduction" of PWROG-19047-P states that this TR documents the EMAs for the North Anna Units 1 and 2 reactor vessel inlet and outlet nozzle Rotterdam welds, nozzle forgings and nozzle belt forgings (a.k.a., upper shell forgings). The PWROG indicated that these locations were chosen for their upper-shelf energy (USE) potentially falling below the 50 ft-lb limit at 80-years (72 EFPY) for subsequent license renewal.

Section 3.2, "J-Integral Resistance Models" of PWROG-19047-P provides the basic form of J-R model as expressed in NUREG/CR-5729 and RG 1.161. Furthermore, PWROG-19047-P provides the different parameters, variables and assumed values used in the EMAs for North Anna, Units 1 and 2. In particular, PWROG-19047-P states that for the variable CVN = Charpy V-notch Impact Energy assumed the 80-year projected USE value. RG 1.161 and Appendix K of Section XI to the ASME Code indicate that toughness properties for the corresponding flaw orientation is used in the analysis (e.g., toughness properties corresponding to T-L or weak orientation for circumferential flaws, or L-T or strong orientation for axial flaws).

However, the staff noted that PWROG-19047-P does not provide the 80-year projected USE values for the components assessed in the EMAs, nor does it provide a supporting basis that EMAs for North Anna, Units 1 and 2. In addition, the PWROG did not provide the details for the variables/assumptions used to determine these 80-year projected USE values, nor did it provide a reference to the current licensing basis document that supports these values.

NRC Request

a) Provide the 80-year projected USE values for the components assessed in the PWROG 19047-P for North Anna, Units 1 and 2, using this model.

b) Explain and justify the methodology, including the variables/assumptions (with the appropriate reference to the current licensing basis document where applicable), used to derive the 80 year projected USE values for these components. If applicable, this justification should address the selection of toughness properties for the corresponding flaw orientation in the material (e.g., T-L or weak orientation for circumferential flaws, or L-T or strong orientation for axial flaws) used in EMAs for North Anna, Units 1 and 2.

Response

(a) The 80-year USE values for North Anna Unit 1 and 2 are documented in WCAP-18364-NP [8], Section 5.

(b) The 80-year USE projections, including the methodology utilized and input parameters are documented in WCAP-18364-NP [8], Section 5, which was submitted to USNRC by Dominion for the North Anna SLR application [12]. The methodology is consistent with that contained in Regulatory Guide 1.99 [6]. WCAP-18364-NP also refers to PWROG-18005-NP [4] as the original source of the input parameters. These input parameters represent weak orientation material properties in all cases.

RAI 05

Issue

Section 4.1.4 of PWROG-19047-P states the procedures for the J-applied calculation for Levels C/D described in K-5000 of Appendix K to Section XI of the ASME Code are the same as those for Levels A/B described in K-4000, except that the effect of cladding/base metal differential thermal expansion needs to be considered for Levels C/D per K-5210(a). Therefore, stress data from the finite element model (FEM) with cladding is included for the Levels C/D evaluation. However, it is not clear to the staff how the effects of the cladding (i.e., the stress due to the different thermal expansion between cladding and base metal) was captured in the FEM.

NRC Request

a) Provide a description of how the stress due to the different thermal expansion between cladding and base metal is captured in the FEM. For example, this stress due to the thermal expansion difference should be maximum at room temperature and decreases as temperature increases.

b) Describe and justify that the FEM addresses the cladding/base metal differential thermal expansion discussed in K-5210.

Response

(a) The FEA for Level C/D explicitly modeled the cladding with appropriate material properties such as thermal expansion coefficients per ASME Section II. Therefore, the differential thermal expansion between the cladding and base metal is considered.

(b) The effect of the cladding/base metal differential thermal expansion is addressed by explicit modeling of the cladding in the FEA.

RAI 06

Issue

Section 4.1.7, "Temperature Range for Upper Shelf Fracture Toughness Evaluations" of PWROG-19047-P states that the transition region toughness is obtained from the ASME Section XI equation for crack initiation (i.e., Section A-4200) and using an RT_{NDT} value of 208.3°F for a flaw depth of 1/10 the wall thickness. PWROG-19047-P cites page 5 of North Anna Power Station Updated Final Safety Analysis Report, Revision 55.

The staff was not able to independently confirm this Adjusted Reference Temperature value for the components in North Anna, Units 1 and 2, in the cited reference. In addition, it appears that the components being represented by this RT_{NDT} value of 208.3°F are the nozzle-to-shell welds; however, it was not specified in the PWROG-19047-P. The staff also noted that it is not clear how this RT_{NDT} value is representative or bounding for the components assessed in the EMAs for North Anna, Units 1 and 2.

NRC Request

a) *Clarify the components in the EMAs for North Anna, Units 1 and 2, that are being represented by the RT_{NDT} value of 208.3°F.*
b) *Provide the unirradiated reference temperature ($RT_{NDT(u)}$), sigma delta, sigma initial, fluence, chemistry factor, copper and nickel values used to calculate the ART and a reference to the applicable current licensing basis document for these values.*
c) *Justify that the ART value used in PWROG-19047-P is representative/bounding for the applicable components being evaluated in the EMAs for North Anna Units 1 and 2.*

Response

The reference "page 5 of [5]" in Section 4.1.7 of PWROG-19047 is a transcription error, refers to a table in a Westinghouse internal analysis input summary document that is shown below. The table shows the calculation of 1/10T ART calculation specifically for the EMA. This value is applicable to the nozzle-to-shell welds and uses a bounding fluence value, as discussed in the responses to RAI-02.

ART Values to be Used at the 1/10T Flaw Location in the EMA

Material	Copper Content(1) (wt. %)	Nickel Content(1) (wt. %)	CF(2)	Fluence (x 10^{19} n/cm², E > 1.0 MeV)	Surf. FF(3)	$RT_{NDT(U)}$(4) (°F)	Predicted ΔRT_{NDT}(5) (°F)	σ_I(6) (°F)	σ_Δ(7) (°F)	M(8) (°F)	ART(9) (°F)
Inlet Nozzle to Upper Shell Welds	0.35	1.13	293.45	0.10	0.417	30	122.3	0	28.0	56.0	208.3
Outlet Nozzle to Upper Shell Welds	0.35	1.13	293.45	0.10	0.417	30	122.3	0	28.0	56.0	208.3

Notes:
(1) Conservatively, the copper and nickel values shown are the limiting, i.e. highest, for the component listed in PWROG-18005-NP [4] between Units 1 and 2.
(2) Based on Table 1 and 2 of Regulatory Guide 1.99 [6] using the copper and nickel weight percent values.
(3) FF = fluence factor = $f^{(0.28 - 0.10*\log(f))}$ per Regulatory Guide 1.99 [6].
(4) Taken from PWROG-18005-NP [4].
(5) Predicted ΔRT_{NDT} = CF * FF per Regulatory Guide 1.99, Revision 2 [6].
(6) The initial RT_{NDT} values are based on measured values; therefore $\sigma_I = 0$°F.
(7) Per Regulatory Guide 1.99 [6], $\sigma_\Delta = 28$°F for welds when surveillance capsule data is not used; however, σ_Δ need not exceed 0.5*ΔRT_{NDT}.
(8) Margin = $2*(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$
(9) ART = $RT_{NDT(U)} + \Delta RT_{NDT}$ + Margin per Regulatory Guide 1.99 [6].

RAI 07

Issue

K-5400 of Appendix K to Section XI of the ASME Code states that when the applicable stress components vary with time during Level C or D Service Loadings, evaluation shall be performed at various times during the postulated loading to determine the limiting conditions for the flaw extension and flaw stability criteria. The J-integral resistance shall be determined at each of the times the evaluation is performed using the metal temperature associated with each flaw depth evaluated.

The staff noted that PWROG-19047-P does not explicitly discuss how the EMAs for North Anna, Units 1 and 2, address K-5400. It appears that K-5400 for the nozzle-to-shell welds for North Anna, Units 1 and 2, is addressed in Figure 4-4 and Figure 4-6; however, it is not clear how PWROG-19047-P addresses K-5400 for the upper shell forgings and nozzle forgings for North Anna, Units 1 and 2.

NRC Request

a) *Clarify/confirm that Figures 4-4 and 4-6 address K-5400 for the inlet nozzle-to-shell weld and outlet nozzle-to-shell weld, respectively. If not, describe and justify how K-5400 is addressed for the nozzle-to-shell welds for North Anna, Units 1 and 2.*
b) *Describe and justify how K-5400 of Appendix K to Section XI of the ASME Code is addressed for the upper shell forgings and nozzle forgings for North Anna, Units 1 and 2.*

Response

a) $J_{applied}$ as a function of time for the Level D SLB transient was calculated. The corresponding crack tip temperature for each time step was obtained from the FEA. PWROG-19047-P, Figures 4-4 and 4-6 plotted the $J_{applied}$ as well as the $J_{material}$ mean curve from all the transient time points vs. their corresponding crack tip temperatures for the inlet nozzle-to-shell weld and outlet nozzle-to-shell weld, respectively. Figures 4-5 and 4-7 plots the Level D $J_{applied}$ along with the associated $J_{material}$ mean curve at the most limiting time points for the inlet nozzle-to-shell weld and outlet nozzle-to-shell weld, respectively. Therefore, the inlet and outlet nozzle-to-shell weld EMAs addressed K-5400 of ASME Section XI.
b) The Level D $J_{applied}$ for all locations was calculated for all FEA time points for the SLB transient. The most limiting $J_{applied}$ for the entire transient is compared to the J-R at either the metal (crack tip) temperature or conservatively at 600°F. Therefore, the upper shell and nozzle forgings EMAs addressed K-5400 of ASME Section XI.

This letter was created and verified in accordance with Westinghouse Level 2 Procedures W2-8.3-101 and W2-8.4-102.

If you have any questions or require further information, please contact the undersigned.

Author Name(s)	Signature / Date	Scope
Gordon Z. Hall	***Electronically Approved****	All
Verifier Name(s)	Signature / Date	Scope
Jay D. White (Lead Verifier)	***Electronically Approved****	RAIs 01, 05 and 07
Benjamin Mays	***Electronically Approved****	RAIs 02, 04 and 06
J. Brian Hall	***Electronically Approved****	RAI 03
Manager Name	Signature / Date	
Stephen P. Rigby	***Electronically Approved****	

***Electronically approved records are authenticated in the electronic document management system.**

Westinghouse