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Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads, Response to NRC RAIs Set 2 (RAI 9)

BAW-2178NP-00-Supplement 1-00-000_Q2NP
Revision 0

June 2018

Framatome Inc.

Nature of Changes

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

Contents

List of Tables

List of Figures

Nomenclature

(If applicable)

Acronym	Definition
EMA	Equivalent Margins Analysis
SLB	Steam Line Break
TR	Topical Report
TP	Turkey Point

1.0 NRC RAIS

The NRC staff is reviewing the Pressurized Water Reactor Owner's Group (PWROG) topical reports (TRs) BAW-2192, Rev. 0 Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level A&B Service Loads," and BAW-2178, Rev. 0, Supplement 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group For Level C&D Service Loads," which provides the EMAs applicable to 80 years of operation for several plants. In order to verify the technical adequacy of the PWROG's Equivalent Margins Analysis (EMA), the NRC staff sent RAIs 1-10 to the PWROG by e-mail dated May 7, 2014. RAIs 1-10 are presented in Section 3.0 of this document.

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

2.0 RESPONSE TO NRC REQUESTS FOR ADDITIONAL INFORMATION

2.1 *RAI 9*

2.1.1 NRC RAI Text (from Section 4.0)

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

2.1.2 PWROG Response to RAI 9

The stress intensity factor due to cladding for the Surry longitudinal seam weld, SA-1526, at the critical time point of the SSDC 1.3 steam line break was calculated using the procedure described in Section 5.2.3 BAW-2178P, Supplement 1, Revision 0 (hereafter referred to as TR). Table 2—1 shows the nodal coordinates and temperature distribution obtained from the Framatome fracture mechanics analysis code PCRIT (TR, Section 5.2.1.1), along with the calculated thermal hoop stresses with and without cladding, stress intensity factors versus through-wall thickness depth with and without cladding, and the stress intensity factor correction for cladding effects, K_{Iclad}. The thermal stress distributions both with and without cladding effects are also plotted in Figure 2—1. With cladding effects included, high stresses ranging from approximately [ ] are predicted in the stainless steel cladding under a large thermal gradient load. [

]

Table 2—1
K_{Iclad} Calculation Data

[ ]

Figure 2—1
Thermal Hoop Stresses with and without Cladding Effect

[ ]

2.1.3 References for Response to RAI 9

Not Applicable.

3.0 NRC RAIS—MAY 7, 2018

REQUEST FOR ADDITIONAL INFORMATION FROM THE

OFFICE OF NUCLEAR REACTOR REGULATION

BAW-2192, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP

FOR LEVEL A&B SERVICE LOADS TOPICAL REPORT

BAW-2178, SUPPLEMENT 1, REVISION 1,

LOW UPPER-SHELF TOUGHNESS FRACTURE MECHANICS ANALYSIS OF

REACTOR VESSELS OF B&W OWNERS REACTOR VESSEL WORKING GROUP

FOR LEVEL C&D SERVICE LOADS TOPICAL REPORT

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

Regulatory Basis

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

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

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

RAI 1

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

RAI 2

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

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

]

RAI 4

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

RAI 5

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

RAI 6

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

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

RAI 7

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

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

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

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

The staff therefore requests the following information:

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

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

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

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

RAI 8

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

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

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

RAI 9

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

RAI 10

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

References

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

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

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

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

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

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

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

RAI-1

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

Revision:

Deletion.

RAI-5

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

Revision:

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

RAI-9

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