ENCLOSURE

ATTACHMENT 3

AREVA NP, INC., DOCUMENT NO. 51-9057252-001, "CR3 RECONCILIATION OF 60-YEAR FLUENCE PROJECTIONS USED IN REACTOR VESSEL BELTLINE EMBRITTLEMENT CALCULATIONS FOR LICENSE RENEWAL" (NON-PROPRIETARY)

AREVA

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Title CR3 Reconciliation of 60-year Fluence Projections Used in Reactor Vessel Beltline Embrittlement Calculations for License Renewal

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Remarks:

Purpose:

Revision 000: CR3 plans to submit a license renewal application to the NRC in the 1st quarter of 2009 and has recently updated 60-year fluence projections at 54 Effective Full Power Years (EFPY). These fluence projections will be used as inputs for new calculations of RT_{PTS}, ART, and P-T limits that will be completed later. The purpose of this document is to reconcile the 60-year fluence projections with the 48 EFPY fluence projections used in previous 60-year reactor vessel beltline equivalent margins analysis (Reference 3) and underclad cracking analysis (Reference 4)—also reported in BAW-2251A, Appendices B and C--and demonstrate that each of these analyses are valid for 60 years.

Section 2.1 of this report compares the fluence values at 54 EFPY from Reference 1 to the 48 EFPY fluence values used in the 60-year/48 EFPY equivalent margins and underclad cracking analyses (References 3 and 4). The 54 EFPY/60-year fluence exceeds the 48 EFPY fluence for all RV locations with the exception of the RV nozzles and RV closure flange.

Section 2.2 of this report shows that the USE values for plates and forgings remain above 50 foot-pounds at 54 EFPY. Section 2.2 also shows that the equivalent margins analysis for the reactor vessel welds reported in Reference 3 remains acceptable for CR3 at 54 EFPY. Calculation of material J-integrals at 54 EFPY shows that TMI-1 weld SA-1526 continues to be limiting, and CR3 is still bounded by the Reference 3 analysis.

Section 2.3 reports pressurized thermal shock and the limiting longitudinal welds are WF-8 and WF-18 with an RT_{PTS} of 231.3°F, which is below the screening criterion of 270°F. The limiting circumferential weld is WF-70 with an RT_{PTS} of 253.8°F, which is below the screening criterion of 300°F.

Section 2.4 reports that the current P-T limits are valid through 32 EFPY and will be updated to incorporate the Reference 1 fluence analysis.

Section 2.5 of this report shows that the underclad cracking evaluation reported in Reference 4 (approved by the NRC in BAW-2251A, Appendix C) remains acceptable at 54 EFPY for the closure flange and beltline regions, but CR3 is not bounded for evaluation of underclad cracks for the nozzle belt regions. Additional evaluation is required.
16 total pages in the document. Section 1.0 (pages 5-6), Section 2.0 (pages 6-15), Section 3.0 (pages 15-16), Section 4.0 (page 16).
Revision 001: Corrects copper value of beltline welds reported in Table 2-2. Section 2.2 revised to note that USE evaluation for beltline welds based on Reference 3 copper content is conservative. 16 total pages in the document. Section 1.0 (pages 5-6), Section 2.0 (pages 6-15), Section 3.0 (pages 15-16), Section 4.0 (page 16). Revision 001 supersedes Revision 000.

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Record of Revisions

Revision	Date	Pages/Sections Changed	Brief Description
000	January 2008	All	Original Issue
001	September 2008	8/Section 2.2	Noted that copper contents for the beltline plates and forgings were obtained from Reference 10.
		9/Section 2.2	Revised copper content of beltline welds reported in Table 2-2 to agree with Reference 10.
		10/Section 2.2	Third paragraph. Noted that analysis in this section assumes higher copper content for welds WF-70, WF-8, and WF-18 reported in Reference 3 versus the copper content reported in Reference 10. This is conservative for calculation of J-integral material.

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1.0 INTRODUCTION

CR3 plans to submit a license renewal application to the NRC in the 1st quarter of 2009, and has updated calculations to address reduction of fracture toughness of reactor vessel beltline materials for 60 years of operation. NRC Regulations 10 CFR 50.60 and 10 CFR 50.61 provide fracture toughness requirements and acceptance criteria applicable to the CR3 nuclear power reactor. NRC Regulation 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation," requires that all light water nuclear power reactors meet the requirements of 10 CFR 50 Appendix G, "Fracture Toughness Requirements," and 10 CFR 50 Appendix H, "Reactor Vessel Material Surveillance Program Requirements." Appendix G specifies fracture toughness requirements for the reactor coolant pressure boundary to provide margins of safety against fracture during any condition of normal plant operation, including anticipated operational occurrences and system hydrostatic tests. The Appendix H Reactor Vessel Surveillance Program is required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors resulting from exposure of these materials to neutron irradiation and the thermal environment.

NRC Regulation 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," provides requirements for computing the reference temperature, RT_{PTS}, for the end-of-life (EOL) fluence for each of the reactor vessel beltline materials, which is a measure of the fracture toughness after exposure to EOL fluence. It also provides a pressurized thermal shock screening criterion for each type of beltline material, which is a temperature above which the plant cannot continue to operate without justification. The RT_{PTS} reference temperatures are a function of material composition and neutron fluence, and they increase as cumulative fluence increases, possibly approaching the screening criterion if the material is highly susceptible to neutron embrittlement. If the RT_{PTS} value is projected to exceed the screening criterion using the EOL fluence, licensees are required to implement flux reduction programs that are reasonably practical to prevent this from occurring.

The 40-year reactor vessel embrittlement calculations for CR3 are reported in the NRC RVID2 database and are based on various Progress Energy and AREVA NP calculations. In June 1996, the B&WOG Generic License Renewal Program (GLRP) submitted topical report BAW-2251, Demonstration of the Management of Aging Effects for the Reactor Vessel, to the NRC for review and approval. BAW-2251 included 60-year calculations of fluence (48 EFPY), pressurized thermal shock, upper shelf energy, and underclad cracking. The NRC reviewed and approved BAW-2251 in April 1999. CR3 withdrew from the GLRP in 1995, and all references to CR3 were removed from the topical report prior to submittal to the NRC. While CR3 may not directly reference BAW-2251A, the supporting calculations for adjusted reference temperature (ART), pressurized thermal shock (PTS), upper shelf energy and underclad cracking were completed prior to CR3 withdrawal from the GLRP and may be used to support the CR3 LRA. However, these supporting calculations were based on 48 EFPY fluence values, which must be reevaluated to determine if they bound 60 years of operation at current capacity factors.

CR3 has recently prepared 60-year fluence projections at 54 Effective Full Power Years (Reference 1) that meet the uncertainty requirements of Regulatory Guide 1.190. These 54 EFPY fluence projections will be used to calculate RT_{PTS}, ART, and P-T limits to support the CR3 LRA submittal; these analyses are scheduled to be completed in the 4th quarter of 2007 and 1st quarter of 2008.

End-of-life (60-year/48 EFPY) reactor vessel embrittlement calculations, prepared as part of the GLRP for CR3, are contained in the following AREVA NP documents:

- Reference temperatures for pressurized thermal shock, adjusted reference temperatures at the 1/4T and 3/4T locations, and upper shelf energy reduction, 32-1240132-03, B&W 177-FA Reactor Vessel Fracture Toughness Properties (Reference 2).

- Equivalent margins analysis for beltline welds, 32-1245770-00/01, Low Upper Shelf Fracture Analysis - Levels A, B, C, and D (Reference 3).

- Fracture mechanics analysis of underclad cracks, 32-1245893-000, FM Analysis of Postulated Underclad Cracks in B&W Designed RV for 48 EFPY (Reference 4).

The EFPY assumed for the evaluations in References 2 through 4 is 48 EFPY and the fluence projections are based on methodology that pre-dates Regulatory Guide 1.190. The 60-year fluence projections have recently been updated at 54 EFPY. ARTs, RT_{PTS} and USE will be updated using the new 60-year fluence at 54 EFPY, and these evaluations will supersede the Reference 2 evaluations at 48 EFPY.

The purpose of this document is to reconcile the recently completed 60-year CR3 fluence projections (Reference 1) and any affects of those fluence projections on the material properties of the reactor vessel equivalent margins and underclad cracking evaluations reported in References 3 and 4, and to identify any additional evaluations that may be required to disposition all RV embrittlement license renewal issues for CR3.

2.0 RV EMBRITTLEMENT

2.1 End-of-Life Fluence

End-of-life fluence is based on a projected value of effective full power years (EFPY) over the licensed life of the plant. For the current term of operation, end-of-life for CR3 is 40 years and reactor vessel embrittlement calculations for pressurized thermal shock and upper shelf energy are based on fluence projections at 32 EFPY. CR3 began operation in December 1976. The plant lifetime capacity factor through 2005 is 0.682 (Reference 2005 NEI Plant Capacity Factors). Assuming a plant capacity factor of 98.5% beyond 2005, CR3 is expected to accrue approximately 50.3 EFPY by December 2036. Therefore, a 54 EFPY fluence estimate used for calculating reactor vessel embrittlement for 60 years of operation is conservative for the period of extended operation.

The 48 EFPY fluence values reported in References 3 and 4 for CR3 are based on analytical (Discrete Ordinate Transport per BAW-2108, Revision 1—Reference 5) calculations for Cycles 1-7, hand adjoint calculations for Cycle 8, and extrapolation to end-of-life (48 EFPY) based on the average flux calculated for Cycle 8. As described in BAW-2251A (Reference 6), Appendix D, the hand adjoint method does not comply with Regulatory Guide 1.190 uncertainty requirements.

The 54 EFPY fluence values calculated in Reference 1 include cavity dosimetry data from Cycles 11 and 12 and plant operation through Cycle 14; these fluence projections do comply with RG 1.190 uncertainty requirements. To account for a potential measurement uncertainty recapture (MUR) and extended power uprate (EPU), the Cycle 14 fluxes were used for Cycle 15 and then increased by a factor of 1.02 for the MUR cycles (Cycles 16 and 17) and by a factor of 1.25 for the EPU cycles (Cycles 18+). A comparison of the fluence projections at the inside wetted surface reported in References 3

and 4 at 48 EFPY to the wetted surface fluence at 54 EFPY from Reference 1 is presented in Table 2-1 below.

Table 2-1--Comparison of Wetted Surface Fluence for Reactor Vessel Locations

Reactor Vessel Location	Material ID	48 EFPY Reference 2, Table 3-8	54 EFPY w/MUR and EPU (calculated in Reference 1) Table 3-5
Plates & Forgings			
1. Nozzle Belt Closure Flange (maximum fluence is next to the circumferential weld that connects the flange to the nozzle belt)	NA	6.75E+16[1]	4.38E+13[2]
2. Nozzle Belt Forging-Lower	AZJ 94	1.10E+19	1.48E+19
3. Upper Shell Plate	C4344-1	1.25E+19	1.60E+19
4. Upper Shell Plate	C4344-2	1.25E+19	1.60E+19
5. Lower Shell Plate	C4347-1	1.20E+19	1.62E+19
6. Lower Shell Plate	C4347-2	1.20E+19	1.62E+19
Welds			
7. NB to US Circ. Weld (Inside 40% is SA-1769 and outside 60% is WF-169-1)	SA-1769	1.10E+19	1.48E+19
8. US Longit. Weld (100%)	WF-8	1.16E+19	1.44E+19
9. US Longit. Weld (100%)	WF-18	1.16E+19	1.44E+19
10. US to LS Circ. Weld (100%)	WF-70	1.20E+19	1.56E+19
11. LS Longit. Weld (100% both)	SA-1580	1.02E+19	1.35E+19
12. RV Nozzle Weld	Linde 80	1.5E+18[3]	6.67E+16

Beltline at 60 Years

The beltline materials for CR3 for 60 years (54 EFPY) include items 2 through 11 in Table 2-1.

Item 1, Nozzle Belt Closure Flange, is not considered beltline material since the fluence is less than the 1.0E+17 n/cm^2 fluence specified in 10 CFR 50 Appendix H above which a material surveillance program is required, but is reported in Table 2-1 since the closure flange was evaluated for underclad cracking in Reference 4. See Section 2.5 below for additional discussion of the applicability of underclad cracking evaluation to CR3.

For reactor vessel beltline items (2 through 11 above), the 60-year (54 EFPY) fluence projections exceed 48 EFPY fluence values reported in Reference 2 and reconciliation to the applicable embrittlement calculations reported in References 3 and 4 is required. This reconciliation is reported in Sections 2.2 through 2.4 below.

For item 12, RV Nozzle Weld (i.e., the Linde 80 weld that connects the reactor vessel outlet nozzle to the nozzle belt shell), the calculated 54 EFPY fluence (6.67E+16 n/cm^2) is less than the 1.0E+17 n/cm^2 fluence specified in 10 CFR 50 Appendix H above which a material surveillance program is required. The outlet nozzle to nozzle belt weld will receive the highest fluence of all reactor vessel nozzles since

[1] Reference 4, Table 2-2
[2] Reference 1, Table 3-9, base metal outer surface value
[3] Reference 6, Page 3-5

it is lower in elevation relative to the active core than the inlet nozzles and core flood nozzles. Since the 60-year fluence at 54 EFPY is less than 1.0E+17 n/cm^2, the reactor vessel inlet nozzles, reactor vessel outlet nozzles, core flood nozzles and associated welds do not need to be considered in the beltline region for the period of extended operation. The nozzles, plates, forgings, and welds identified as beltline materials for the CR3 reactor vessel at 54 EFPY are the same as those identified at 32 EFPY. The beltline materials for CR3 for 60 years (54 EFPY) include items 2 through 11 in Table 2-1.

2.2 Upper Shelf Energy-Beltline Materials

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb...." Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides two methods for determining Charpy upper-shelf energy (C_VUSE). Position 1.2 applies for material that does not have surveillance data available, and Position 2.2 applies for material that does have surveillance data. For Position 1.2, the percent drop in C_VUSE, for a stated copper content and neutron fluence, is determined by reference to Figure 2 of Regulatory Guide 1.99, Revision 2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE. For Position 2.2, the percent drop in C_VUSE is determined by plotting the available data on Figure 2, and fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points.

USE for Plates and Forgings (Beltline)

Initial upper shelf energy for beltline plates and forgings are obtained from Reference 2. Copper content for beltline plates and forgings are obtained from Reference 10. Fluence at the 1/4T location is obtained from Reference 10. Upper shelf energies for these beltline plates and forgings at 54 EFPY, using Position 1.2, are reported in Table 2-2 and are all above 50 ft-lb, which is acceptable. Percentage reduction in USE is obtained from Figure 2 of Regulatory Guide 1.99, Revision 2. Position 2.2 could be applied to plate C4344-1, but Reference 2 indicates that Position 1.2 is conservative for plate C4344-1.

Table 2-2 CR3 Upper Shelf Energy at 54 EFPY

Material Description								
Reactor Vessel Beltline Region Location	Material ID	Heat Number	Type	Cu wt%	Initial CvUSE, ft-lbs	54 EFPY[1] Fluence 1/4T Location, n/cm^2	Estimated 54 EFPY CvUSE at 1/4T, ft-lbs	54 EFPY % Drop at 1/4T
Regulatory Guide 1.99, Revision 2, Position 1.2								
Nozzle Belt Forging	AZJ 94	123V190	A508 Cl. 2	0.13	109	8.66E+18	86	21.3
Upper Shell Plate	C4344-1	C4344-1	SA-533 Gr B1	0.20	88	9.36E+18	63	28.5
Upper Shell Plate	C4344-2	C4344-2	SA-533 Gr B1	0.20	88	9.36E+18	63	28.5
Lower Shell Plate	C4347-1	C4347-1	SA-533 Gr B1	0.12	119	9.47E+18	94	20.7
Lower Shell Plate	C4347-2	C4347-2	SA-533 Gr B1	0.12	86	9.47E+18	68	20.7
NB to US Circ. Weld (40% ID)	SA-1769	71249	ASA/Linde80	0.23	70	8.66E+18	EMA[2]	EMA
NB to US Circ. Weld (60% OD)	WF-169-1	8T1554	ASA/Linde80	0.16	70	-	-	
US Long. Weld (100%)	WF-8	8T1762	ASA/Linde80	0.19	70	8.42E+18	EMA[2]	EMA
US Long. Weld (100%)	WF-18	8T1762	ASA/Linde80	0.19	70	8.42E+18	EMA[2]	EMA
US to LS Circ Welds (100%)	WF-70	72105	ASA/Linde80	0.32	70	9.12E+18	EMA[2]	EMA
LS Long. Weld (Both 100%)	SA-1580	8T1762	ASA/Linde80	0.19	70	7.90E+18	EMA[2]	EMA

1. Fluence values are obtained from Reference 10, Table 2-1
2. EMA-equivalent margins analysis required since CvUSE is less than 50 ft-lb at 40 years

USE for Welds

As is the case for the current term of operation, the C_vUSE values for all beltline welds are below 50 ft-lb, requiring an equivalent margins analysis (EMA) in accordance with the requirements of 10 CFR 50.60 for the period of extended operation.

An equivalent margins analysis for CR3 beltline welds at 48 EFPY is reported in Reference 3. The limiting CR3 welds evaluated in Reference 3 include WF-70 (circumferential weld) and WF-8,18 (longitudinal welds). In order to show the equivalent margins analysis remains valid for 60 years, the 48 EFPY fluence values used in Reference 3 for the limiting CR3 welds (WF-70 and WF-8,18) were compared to the 54 EFPY fluence projections from Reference 1, as presented in Table 2-1 of this document. For weld WF-70, the comparison shows that the 48 EFPY fluence value (1.20E19 n/cm^2) used in Reference 3 is lower than the 54 EFPY fluence (1.56E19 n/cm^2) from Reference 1. For welds WF-8 and WF-18, the comparison shows that the 48 EFPY fluence (1.16E19 n/cm^2) used in Reference 3 is lower than the 54 EFPY fluence (1.44E19 n/cm^2) from Reference 1. Therefore, an evaluation will be performed for the weld materials by considering the effect of the increased fluence on the J-integral of the material, which is a function of fluence.

Table 2-3 provides fluence estimates and associated fracture toughness properties for the limiting welds WF-70, WF-8, and WF-18. Since the fluence at 60-years (54 EFPY) has increased relative to the Reference 3 analysis at 48 EFPY, the impact on J-integral of the material must be assessed. The applied J-integral will not change since it is not a function of fluence. The J-integral of the material at 54 EFPY is calculated from Reference 3, Section 3.1.3. Copper, temperature, and net specimen thickness are as reported in Reference 3 for welds WF-70, WF-8, and WF-18. Note that the copper content of welds WF-70, WF-8, and WF-18 reported in Reference 3 are 0.35, 0.20, and 0.20, respectively. The copper content of welds WF-70, WF-8, and WF-18 reported in Reference 10 are 0.32, 0.19, and 0.19, respectively. Use of the higher copper content reported in Reference 3 is bounding for this evaluation since lower J-integral of the material values will be obtained for comparison to $J_{applied}$.

In accordance with Section 3 of Reference 3, the first acceptance criterion for J (ASME Section XI, Article K-2200(a)(1)) for Levels A and B service loadings is based on a ductile flaw extension of 0.10 inch and is satisfied when $J_1 < J_{0.1}$. J_1 equals the applied J-integral for a safety factor of 1.15 on pressure and a safety factor of 1.0 on thermal loading. $J_{0.1}$ equals the J-integral resistance at a ductile flaw extension of 0.1 inch.

As a result of the increased fluence, $J_{0.1}/J_1$ for WF-70 for Level A and B service loads is reduced from 3.21 to 3.16 (see Table 2-3) but remains greater than 1.0. For WF-8 and WF-18, $J_{0.1}/J_1$ is reduced from 1.32 to 1.31 (see Table 2-3) but remains greater than 1.0. Since each ratio remains greater than 1.0, the first acceptance criterion on the limit of the applied J-integral is satisfied. The base analysis (Reference 3) is a bounding analysis of all the reactor vessels of the BWOG, combining the worst weld with the worst loading condition to show that all reactor vessels included in the evaluation are acceptable. The limiting weld in the Reference 3 evaluation was TMI-1 weld SA-1526 with a 1/4T ratio of 1.09. Since the revised $J_{0.1}/J_1$ ratio for CR3 welds WF-8 and WF-18 (1.31) exceeds the limiting $J_{0.1}/J_1$ ratio for TMI-1 weld (SA-1526 with a ratio of 1.09), the assessment reported in Reference 3 remains bounding for CR3 at 54 EFPY.

In accordance with Section 3 of Reference 3, the second acceptance criterion for flaw stability relative to Level A and B service loadings states that flaw growth at a pressure of 1.25 times the accumulation

pressure shall be ductile and stable using a factor of safety of 1.0 on thermal loading. Since J_R, the J-integral resistance to ductile tearing for the material, has not changed, and TMI-1 weld SA-1526 remains the limiting weld (Section 3.2 of Reference 3), it is concluded that the flaw extension would be ductile and stable, thereby satisfying the second acceptance criterion of Appendix K for Level A and B service loads.

In accordance with Section 4 of Reference 3, the first acceptance criterion for Level C transients is that the crack driving force, J applied, must be less than the material toughness, J material. The acceptance criterion for Level C and D transients is that flaw extension must be ductile and stable using a factor of safety of 1.0 on loading.

For Level C and D service loads, TMI-1 weld SA-1526 was selected as the most limiting weld (Section 4.1 of Reference 3). Since the inside surface fluence of CR3 welds WF-70, WF-8, and WF-18 have increased at 54 EFPY relative to 48 EFPY, the J-integral resistance of the affected welds must be recomputed to determine if TMI-1 SA-1526 is still the most limiting weld. The reduction in material to applied J-integral ($J_{0.1}/J_1$) for the limiting CR3 welds are summarized in Table 2-4. Welds WF-8 and WF-18 are limiting for CR3 and the ratio of material to applied J-integral is reduced from 3.99 to 3.96, which exceeds the ratio of 3.26 (Reference 3, Table 4.1) for TMI-1 weld SA-1526. TMI-1 weld SA-1526 continues to be the most limiting weld.

In accordance with Reference 3, Section 4.4, for TMI-1 weld SA-1526 the values of J_R and $J_{applied}$ are 545 in-lb/in^2 and 241 in-lb/in^2, respectively, yielding a margin of 2.26. Since the updated fluence at the 1/10T location for CR3 does not affect the selection of the limiting weld (i.e., TMI-1 weld SA-1526), the conclusions of Reference 3, Section 4.4 are not affected and the margin of J_R to $J_{applied}$ will be 2.26, which is well above the acceptance criterion of 1.0. The conclusions relative to the evaluation of Level C service loads relative to J_R and $J_{applied}$ and to Level C and D service loads relative to ductile and stable flaw extension reported in Reference 3 remain valid for CR3 at 54 EFPY.

The second acceptance criterion for Level C transients and the first acceptance criterion for Level D transients is that flaw extension must be ductile and stable using a factor of safety of 1.0 on loading. Since the applied loading and limiting weld (SA-1526) have not changed, conclusions reported in Reference 3, Section 4.4, for the second acceptance criterion for Level C and D loadings remain valid for CR3 at 54 EFPY.

The last criterion for Level D conditions is that the extent of 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. Since the applied loading and limiting weld (SA-1526) have not changed, conclusions reported in Reference 3, Section 4.4, for the third acceptance criterion for Level D conditions remain valid for CR3 at 54 EFPY.

This reconciliation demonstrates that welds WF-70, WF-8, and WF-18 satisfy the acceptance criteria of Appendix K of the Section XI of the ASME Code, and therefore, provide margins of safety equivalent to those of Appendix G of ASME Section XI. It may be concluded that welds WF-70, WF-8, and WF-18 have adequate upper-shelf toughness and satisfy the requirement of Appendix G to 10 CFR Part 50, Section IV.A.1.a at a reactor vessel life of 54 EFPY for CR3. The increased fluence at 60 years for CR3 does not change the selection of the limiting TMI-1 weld SA-1526 for the evaluation in Reference 3.

Table 2-3 Equivalent Margins Analysis for Level A and B Service Loads—J-Integral Resistance at a Flaw Depth of 1/4T at 54 EFPY

Beltline Weld ID	Surface Fluence (n/cm^2)	1/4T Fluence (n/cm^2)	$J_{0.1}$ material, Lower Bound (in-lb/in^2)	J_1 applied (in-lb/in^2)	$J_{0.1}/J_1$	Acceptance Criterion for $J_{0.1}/J_1$	Conclusion
48 EFPY Values from References 2 and 3							
WF-70	NA	6.72E+18	544	169	3.21	>1.0	Acceptable
WF-8,18	NA	6.50E+18	667	506	1.32	>1.0	Acceptable
54 EFPY 1/4T Fluence from Table 2-2 $J_{0.1}$ at 54 EFPY calculated in accordance with Reference 3, Section 3.							
WF-70	NA	9.12E18	534	169	3.16	>1.0	Acceptable
WF-8,18	NA	8.42E18	661	506	1.31	>1.0	Acceptable

Table 2-4 Equivalent Margins Analysis for Level C and D Service Loads—J-Integral Resistance at a Flaw Depth of 1/10T at 54 EFPY

Beltline Weld ID	Surface Fluence (n/cm^2)	1/10T Fluence (n/cm^2)	$J_{0.1}$ material, Lower Bound (in-lb/in^2)	J_1 applied (in-lb/in^2)	$J_{0.1}/J_1$	Acceptance Criterion for $J_{0.1}/J_1$	Conclusion
48 EFPY Values from References 2 and 3							
WF-70	1.20E19	9.80E+18	532	65	8.14	>1.0	Acceptable
WF-8,18	1.16E19	9.47E+18	658	165	3.99	>1.0	Acceptable
54 EFPY Values Based On Highest Fluence (54 EFPY) from Reference 1. $J_{0.1}$ at 54 EFPY calculated in accordance with Reference 3, Section 4.							
WF-70	1.56E19	1.27E19	523	65	8.05	>1.0	Acceptable
WF-8,18	1.44E19	1.18E19	653	165	3.96	>1.0	Acceptable

2.3 Pressurized Thermal Shock

The previous CR3 RT_{PTS} values for the reactor vessel beltline materials for the period of extended operation were found in Table 6-2 of AREVA NP document 32-1240132-03 (Reference 2). These values have been superseded by RT_{PTS} values (Reference 11) calculated using 54 EFPY fluence projections from Reference 1. From Reference 11, Table 2-1, the limiting longitudinal welds are WF-8 and WF-18 with an RT_{PTS} of 231.3°F, which is below the screening criterion of 270°F. The limiting circumferential weld is WF-70 with an RT_{PTS} of 253.8°F, which is below the screening criterion of 300°F.

2.4 P-T Limits at 60 Years

In accordance with the NRC SER of BAW-2251A, Applicant Action Item 12, an applicant must show that an operating window will be available between the pressure-temperature limits and the net positive suction curves for the reactor coolant pumps at 60 years. At present, the current Appendix G based uncorrected P-T limits for CR3 was developed by AREVA NP in September 2000 in 32-5000854-01, "CR3 Uncorrected P-T Limits at 32 EFPY." Even though CR3 was not included in BAW-2251A it is

recommended that 60-year P-T limits for CR3 be calculated using the Reference 1 fluence analysis at 54 EFPY to support the LRA submittal.

2.5 Underclad Cracking

BAW-2251A, Appendix C (also BAW-2274A) updates and supersedes the fracture mechanics analysis for underclad cracking as originally reported in BAW-10013A. The supporting calculation for BAW-2274A is Reference 4. The revised analysis concluded that postulated underclad cracking in the reactor vessel meets the acceptance criteria of the ASME Code, Section XI, IWB-3612. The maximum crack growth and applied stress intensity factor for normal and upset condition occurs in the nozzle belt region. The fracture toughness margin for normal and upset conditions was determined to be 3.63, which is greater than the required toughness margin of 3.16 (i.e., $\sqrt{10}$). The maximum applied stress intensity for the emergency and faulted condition occurs in the closure head to head flange regions. The fracture toughness margin for emergency and faulted condition was 2.42, which is greater than the required toughness margin of 1.41 (i.e., $\sqrt{2}$).

The Reference 4 analysis was based on 48 EFPY fluence estimates reported in References 2 and 4 and associated fracture toughness properties. Three vessel regions were evaluated: (1) nozzle belt, (2) closure flange, and (3) beltline.

Nozzle Belt

In accordance with Reference 4, the controlling nozzle belt forging used in the evaluation was Oconee Unit 3 forging 4680 with an adjusted RT_{NDT} at the inside surface and 1/4T locations of 175°F and 159°F, respectively (Reference 4, Table 2-1). The adjusted RT_{NDT} at the inside surface and 1/4T locations of CR3 nozzle belt forging AZJ 94 at 54 EFPY must be computed and compared to ONS-3 forging 4680 to ensure that the ONS-3 forging remains the limiting nozzle belt material.

From Reference 4, Section 2, the thickness of the nozzle belt is 12 inches. Using RG 1.99, Revision 2, Equation (3), the 1/4T fluence is 1.48E19 * e^(-.24*12/4) = 7.2E+18 n/cm^2. From Reference 4, Table 2-1, the ART at the inside surface and 1/4T locations is as follows:

ART = Initial RT_{NDT} + CF*ff + margin

Initial RT_{NDT} = +3°F

CF = 94.0

Margin = 71°F

ff = f^(0.28-0.10*log f) RG 1.99, Rev. 2, Equation (2), f is fluence E19 n/cm^2

ART at inside surface = 3°F + 94* 1.48^(0.28-0.10*log 1.48) + 71°F = 178°F

ART at 1/4T = 3°F + 94 *0.72* ^(0.28-0.10*log 0.72) + 71°F = 159°F

The ART at the inside surface of CR3 nozzle belt forging AZJ 94 is 3°F higher than the ART evaluated for ONS-3 forging 4680 in Reference 4. Therefore, the CR3 nozzle belt forging is not bounded by the

Reference 4 analysis. The impact of this differential is not expected to be significant but must be evaluated further.

Closure Flange

Evaluation of the closure flange in Reference 4 identified limiting closure flange material based on an inside surface fluence of 7.78E+16 n/cm^2 (BAW-2274-A, Table 2-2). Material properties were assumed to be identical (i.e., copper, nickel, initial RT_{NDT}, chemistry factor) for all B&W plants. For CR3, the fluence at 54 EFPY at the closure flange is 4.38E+13 n/cm^2 (Item 1 of Table 2-1) and is bounded by the Reference 4 analysis.

Beltline (upper and lower shells)

CR3 beltline upper and lower shell plates are fabricated from SA-533 Grade B1 and are not susceptible to underclad cracking. Since CR3 does not have A508 Class 2 forgings in the upper and lower shell region, the increase in ART due to increased fluence at 54 EFPY is not relevant for CR3 for the evaluation of underclad cracking.

Note: The limiting ART used in Reference 4 for the beltline upper and lower shell evaluation was conservatively taken from CR3 plate C4344-1, which had a higher adjusted reference temperature than the other B&W plants with A508 Class 2 forgings.

2.6 Reactor Vessel Integrity Program

A comprehensive B&WOG Reactor Vessel Integrity Program (RVIP) was established in 1977 in response to concerns relative to reduction of fracture toughness of B&W 177-Fuel Assembly (FA) reactor vessels. The RVIP was organized to obtain material property information and validate test and analysis methods to respond to the concerns related to the fracture toughness in beltline weld materials. The RVIP was later extended to include reference temperature shift concerns, i.e., pressure-temperature operational limits and pressurized thermal shock.

When Owners of Westinghouse-designed, B&W-fabricated reactor vessels joined the RVIP, additional capsules were fabricated and plans were made to irradiate these capsules to an extended neutron fluence range. This extended range covers the neutron fluence to which the B&W 177-FA reactor vessels are expected to see for the period of extended operation associated with the renewal of an operating license as described in Section 4.0 of BAW-2251A (Reference 6). By incorporating these extended range capsules and the Westinghouse plant-specific capsules into the existing integrated RVIP, an enlarged integrated program, known as the Master Integrated Reactor Vessel Material Surveillance Program (MIRVP), was formed.

The current status of MIRVP capsules and testing of Linde 80 weld material is summarized in BAW-1543(NP), Revision 4, Supplement 6-A (Reference 7). Reactor vessel materials and surveillance data are provided in BAW-2313, Revision 5 (Reference 8). The MIRVP includes irradiation and testing of eight heats of Linde 80 weld material. For B&W fabricated vessels, Linde 80 weld metal is limiting relative to the plates and forgings and the current status of capsule withdrawals for CR3-specific plate surveillance material is provided in BAW-1543, Revision 4, Supplement 6-A.

The limiting beltline circumferential weld for CR3 at 54 EFPY is WF-70, heat number 72105. From Table 2-1, the fluence at 54 EFPY for weld WF-70, is 1.56E+19 n/cm^2. In accordance with BAW-2313,

Revision 5 (Reference 8), two capsules with weld wire heat number 72105 have been irradiated to fluence values equal to or greater than 1.56E+19 n/cm^2 and tested. Therefore, the MIRVP program covers the fluence at 54 EFPY for CR3 weld WF-70, and no additional surveillance material or testing is required for 60-years of operation for CR3.

The limiting beltline axial welds for CR3 at 54 EFPY are WF-8 and WF-18, heat number 8T1762. This heat of material is not in the MIRVP and there is no need to add this material since the CR3 Linde 80 beltline weld materials are adequately represented by the eight heats of material in the MIRVP program. The equivalent margins analysis reported in Section 2.2 evaluate USE reduction for axial welds WF-8 and WF-18 for the period of extended operation.

Upper shell plate material C4344-1 was included in CR3-specific capsules, and all specimens have been removed and tested. From Table 2-1, the 54 EFPY fluence at plate C4344-1 is predicted to be 1.60E+19 n/cm^2. Capsule CR3-F (Reference 7), which contains C4344-1 material, received a fluence of 1.08E+19 n/cm^2 was removed and tested. The MIRVP has determined that no further testing is required for material C4344-1 since the plate material is not the limiting material for the CR3 vessel and the MIRVP meets the requirements of 10 CFR 50 Appendix H (Reference 7). In addition, surveillance data for C4344-1 is credible and use of 10 CFR 50.61 shift prediction is conservative relative to use of surveillance data (Reference BAW-2325, Revision 1, Page B-5, Reference 9).

The MIRVP meets the requirements of 10 CFR 50, Appendix H, which requires that the CR3 beltline materials be monitored by a surveillance program complying with ASTM E 185, as modified by Appendix H. The MIRVP is an integrated program that is reviewed and approved by the NRC. Detailed discussion of the MIRVP should be provided in the CR3 reactor vessel integrity program bases document for license renewal. No additional materials or testing by the MIRVP are required to cover the period of extended operation for CR3 for the 54 EFPY fluence reported in Reference 1.

3.0 CONCLUSIONS

CR3 plans to submit a license renewal application to the NRC in the 1st quarter of 2009 and has recently updated 60-year fluence projections at 54 Effective Full Power Years (EFPY). These fluence projections will be used as inputs for new calculations of RT_{PTS}, ART, and P-T limits that will be completed in 2008. The purpose of this document is to reconcile the 60-year fluence projections with the 48 EFPY fluence projections used in previous 60-year reactor vessel beltline equivalent margins analysis (Reference 3) and underclad cracking analysis (Reference 4)—also reported in BAW-2251A, Appendices B and C--and demonstrate that each of these analyses are valid for 60 years.

Section 2.1 of this report compares the fluence values at 54 EFPY from Reference 1 to the 48 EFPY fluence values used in the 60-year/48 EFPY equivalent margins and underclad cracking analyses (References 3 and 4). The 54 EFPY/60-year fluence exceeds the 48 EFPY fluence for all RV locations with the exception of the RV nozzles and RV closure flange. The nozzles, plates, forgings, and welds identified as beltline materials for the CR3 reactor vessel at 54 EFPY are the same as those identified at 32 EFPY. The CR3 beltline materials include the lower nozzle belt forging, circumferential weld that connects the lower nozzle belt forging to the upper shell, upper shell consisting of two upper shell plates and two upper shell axial welds, the circumferential weld that connects the upper shell to the lower shell, and the lower shell consisting of two lower shell plates and two lower shell axial welds. All RV nozzles will receive fluence less than 1.0E17 n/cm^2 at 54 EFPY and are not considered beltline material.

Section 2.2 of this report shows that the USE values for plates and forgings remain above 50 foot-pounds at 54 EFPY. Section 2.2 also shows that the equivalent margins analyses for the reactor vessel welds reported in Reference 3 remains acceptable for CR3 at 54 EFPY. Calculation of material J-integrals at 54 EFPY shows that TMI-1 weld SA-1526 continues to be limiting, and CR3 is still bounded by the Reference 3 analysis at 54 EFPY.

Section 2.3 reports pressurized thermal shock and the limiting longitudinal welds are WF-8 and WF-18 with an RT_{PTS} of 231.3°F, which is below the screening criterion of 270°F. The limiting circumferential weld is WF-70 with an RT_{PTS} of 253.8°F, which is below the screening criterion of 300°F.

Section 2.4 reports that the current P-T limits are valid through 32 EFPY and will be updated to incorporate the Reference 1 fluence analysis.

Section 2.5 of this report shows that the underclad cracking evaluation reported in Reference 4 (approved by the NRC in BAW-2251A, Appendix C) remains acceptable at 54 EFPY for the closure flange and beltline regions, but CR3 is not bounded for evaluation of underclad cracks for the nozzle belt regions. Additional evaluation is required.

4.0 REFERENCES

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3. Lower Upper Shelf Fracture Analysis - Levels A, B, C, and D, 32-1245770-00 and -01, January 9, 1998.
4. FM Analysis of Postulated Underclad Cracks in B&W Designed RV for 48 EFPY, 32-1245893-00, July 15, 1996.
5. Fluence Tracking System, 77-2108, Revision 1, September 28, 1992.
6. Demonstration of the Management of Aging Effects for the Reactor Vessel, BAW-2251A (43-2251A), August 1999.
7. Master Integrated Reactor Vessel Surveillance Program, BAW-1543, Revision 4, February 1993; and Supplement to the Master Integrated Reactor Vessel Surveillance Program, BAW-1543(NP), Revision 4, Supplement 6-A, June 2007.
8. Reactor Vessel Materials and Surveillance Data Information, BAW-2313, Revision 5, December 2005.
9. Response to Request for Additional Information Regarding Reactor Pressure Vessel Integrity, BAW-2325, Revision 1, January 1999.
10. ART Values for Crystal River Unit 3 at 60 Calendar Years, 32-9058829-000, November 2007.
11. RT_{PTS} Values for Crystal River Unit 3 at 60 Calendar Years, 32-9058826-000, November 2007.