April 19, 2005

MEMORANDUM TO: Samson S. Lee, Chief
Safety Section
License Renewal and Environmental Impacts Program
Division of Reactor Improvement Programs

FROM: Matthew A. Mitchell, Chief ***(/RA by MAMitchell)***
Vessels and Internals Integrity and Welding Section
Materials and Chemical Engineering Branch
Division of Engineering

SUBJECT: BROWNS FERRY NUCLEAR LICENSE RENEWAL APPLICATION - FINAL SER SECTIONS FOR TIME-LIMITED AGING ANALYSES (TAC NOS. MC1704, MC1705 AND MC1706)

Section 4.2, Neutron Embrittlement of Reactor Vessel and Internals, of the Browns Ferry Nuclear license renewal application (BFN LRA) includes the following time-limited aging analyses (TLAAs) on neutron irradiation embrittlement of reactor vessel materials: (1) Section 4.2.1, Upper Shelf Energy Reduction due to Neutron Embrittlement, (2) Section 4.2.2, Adjusted Reference Temperature for Reactor Vessel Materials due to Embrittlement, (3) Section 4.2.3, Reflood Thermal Shock Analysis of the Reactor Vessel, (4) Section 4.2.4, Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud, (5) Section 4.2.5, Pressure-Temperature Limits, (6) Section 4.2.6, Reactor Vessel Circumferential Weld Examination Relief, (7) Section 4.2.7 Reactor Vessel Axial Weld Failure Probability. Section 4.7, Other Plant-Specific TLAA of the BFN LRA includes Section 4.7.6, Irradiation Assisted Stress Corrosion Cracking Reactor Vessel Internals.

The staff has completed its review of the TLAAs provided in LRA Section 4.2 and its subsections and Section 4.7.6, as well as Tennessee Valley Authority (TVA's) responses to requests for additional information (RAIs) that were issued on these TLAAs dated January 31, 2005. The staff has determined that TVA has sufficiently addressed the issues raised by the staff in the RAIs, with the exceptions of the open items, and therefore concludes that the TLAAs in Section 4.2 and Section 4.7.6 of the BFN LRA are acceptable pending the resolution of the open items. Attachment provides the staff's Final SER (FSER) section on LRA Sections 4.2 and 4.7.6. This completes the staff review relative to the TLAAs in LRA Sections 4.2 and 4.7.6.

Docket Nos. 50-259, 50-260, 50-296

Attachment: As stated

CONTACT: Ganesh Cheruvenki, DE/EMCB
(301) 415-2501

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FINAL SAFETY EVALUATION REPORT
FOR THE BROWNS FERRY NUCLEAR UNITS 1, 2 AND 3 LICENSE RENEWAL APPLICATION
EVALUATION OF TLAAs FOR REACTOR VESSEL NEUTRON EMBRITTLEMENT

4.1 Identification of Time-Limited Aging Analyses

This section addresses the identification of time-limited aging analyses (TLAAs). The applicant discusses the TLAAs in Sections 4.2.1 through 4.2.8 of the License Renewal Application (LRA). The staff's review of the TLAAs can be found in Sections 4.2 through 4.7 of this Safety Evaluation Report (SER).

The TLAAs are plant-specific safety analyses that are based on an explicitly assumed 40-year plant life. Pursuant to Part 54.21(c)(1) of Title 10 of the *Code of Federal Regulations*, the applicant for license renewal must provide a list of TLAAs, as defined in 10 CFR 54.3.

In addition, pursuant to 10 CFR 54.21(c)(2), an applicant must provide a list of plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs. For any such exemptions, the applicant must provide an evaluation that justifies the continuation of the exemptions for the period of extended operation.

4.2 Reactor Vessel and Internals Neutron Embrittlement

During plant service, neutron irradiation reduces the fracture toughness of ferritic steel in the reactor vessel (RV) beltline region of light-water nuclear power reactors. Areas of review to ensure that the RV has adequate fracture toughness to prevent brittle failure during normal and off-normal operating conditions are (1) upper-shelf energy (USE), (2) adjusted reference temperature (ART), (3) a low-pressure coolant injection (LPCI) reflood thermal shock analysis, (4) heatup and cooldown (pressure-temperature limits) curves, (5) Boiling Water Reactor (BWR) Vessel and Internals Project (VIP) VIP-05 analysis for elimination of circumferential weld inspection, and (6) analysis of the axial welds. The adequacy of the analyses for these six areas is reviewed for the period of extended operation.

ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (delta RT_{NDT}), and a margin (m) term. The delta RT_{NDT} is the product of a chemistry factor (CF) and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," or from surveillance data. The fluence factor is dependent upon the neutron fluence. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculation methods. Revision 2 of RG 1.99 describes the methodology to be used in calculating the margin term. The mean RT_{NDT} is the sum of the initial RT_{NDT} and the delta RT_{NDT}, without the margin term. The delta RT_{NDT} and ART calculations meet the criteria of 10 CFR 54.3(a), therefore they are considered as TLAAs.

The ART values are used in the analysis for the adjusted reference temperature for the RV material due to neutron embrittlement, the pressure-temperature limits analysis, and the reflood thermal shock analysis. The mean RT_{NDT} values are used in the analysis of the circumferential weld examination relief and the axial weld failure probability.

10 CFR Part 50, Appendix G provides the staff's criteria for maintaining acceptable levels of USE for the RV beltline materials of operating reactors throughout the licensed lives of the facilities. The rule 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 life of the facility, unless it can be demonstrated through analysis that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI of the American Society of Mechanical Engineers (ASME) Code. 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 10 CFR Part 50, Appendix H RV material surveillance program.

RG 1.99, Revision. 2, provides an expanded discussion regarding the calculation of Charpy USE values and describes two methods for determining Charpy USE values for RV beltline materials, depending on whether or not a given RV beltline material is represented in the plant's reactor vessel material surveillance program (i.e., 10 CFR Part 50, Appendix H program). If surveillance data is not available, the Charpy USE is determined in accordance with position 1.2 in RG 1.99, Revision 2. If surveillance data is available, the Charpy USE should be determined in accordance with position 2.2 in RG 1.99, Revision 2. These methods refer to Figure 2 in RG 1.99, Revision 2, which indicates the percentage drop in Charpy USE is dependent upon the amount of copper in the material and the neutron fluence. Since the analyses performed in accordance with Appendix G to 10 CFR Part 50 are based on a flaw with a depth equal to one-quarter of the vessel wall thickness (1/4T), the neutron fluence used in the Charpy USE analysis is the neutron fluence at the 1/4T depth location.

The applicant described its evaluation of this TLAA in LRA Section 4.2, "Neutron Embrittlement of the Reactor Vessel and Internals." In order to demonstrate that neutron embrittlement does not significantly impact BWR reactor pressure vessel (RPV) and vessel internals integrity during the license renewal term, the applicant included discussion of the following topics related to neutron embrittlement in LRA Section 4.2:

- reactor vessel materials upper-shelf energy reduction due to neutron embrittlement (LRA Section 4.2.1)
- adjusted reference temperature for reactor vessel materials due to neutron embrittlement (LRA Section 4.2.2)
- reflood thermal shock analysis of the reactor vessel (LRA Section 4.2.3)
- reflood thermal shock analysis of the reactor vessel core shroud (LRA Section 4.2.4)
- reactor vessel thermal limit analyses—operating pressure-temperature limits (LRA Section 4.2.5)
- reactor vessel circumferential weld examination relief (LRA Section 4.2.6)
- reactor vessel axial weld failure probability (LRA Section 4.2.7)
- Irradiation Assisted Stress Corrosion Cracking (IASCC) of the recator vessel and its internals (LRA Section 4.7.6)
- stress relaxation of the core plate hold down bolts (LRA Section 4.7.7)

4.2.1. Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement

4.2.1.1 Summary of Description

In Section 4.2.1 of the LRA, the applicant provided USE values for the limiting beltline materials of the BFN Units 1, 2 and 3. USE is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. 10 CFR 50, Appendix G requires the predicted end-of-life license (EOL) Charpy impact test USE value for RV materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. The applicant stated that the initial unirradiated test data are not available for the BFN RV to demonstrate a minimum 50 ft-lb USE by standard methods. Therefore, EOL fracture energy was evaluated by using the equivalent margin analysis (EMA) methodology described in General Electric NEDO-32205-A, “10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper-Shelf Energy in BWR-2 through BWR-6 Vessels,” which was approved by the staff. According to the applicant, this analysis confirmed that an adequate margin of safety against fracture, equivalent to 10 CFR 50, Appendix G requirements, does exist. The EOL USE calculations satisfy the criteria of 10 CFR 54.3(a). As such, these calculations are a TLAA.

The RVs were originally licensed for a 40 years with an assumed neutron exposure of less than 10^{19} n/cm^2 (E > 1.0 MeV). The current licensing basis calculations use calculated fluences that are lower than this limiting value. The applicant stated that the design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40 year license term for each unit. The tests performed on RV materials provided limited Charpy impact data. It was not possible to develop original Charpy impact test USE values using the methods of 10 CFR 50, Appendix H and ASTM E23, “Methods for Notched Bar Impact Testing of Metallic Materials,” invoked by 10 CFR 50, Appendix G. Therefore, alternative methods approved by the NRC in NEDO-32205-A were used to demonstrate compliance with the 10 CFR 50, Appendix G USE requirement.

Fluences were calculated for the reactor vessels for the extended 60-year [54 EFPY (Effective Full-Power Year), for Unit 1; 52 EFPY for Units 2 and 3] licensed operating periods, using the methodology of NEDC-32983P, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,” which was approved by the staff in a safety evaluation report (SER) dated September 14, 2001. The applicant used one bounding fluence calculations which includes extended power uprate (EPU) for each unit. The applicant provided the results for one bounding calculation for each RV and determined the peak surface fluence of 1.95×10^{18} n/cm^2 and peak 1/4T fluence of 1.35×10^{18} n/cm^2 for the BFN, Unit 1 vessel, and peak surface fluence of 2.3×10^{18} n/cm^2 and peak 1/4T fluence of 1.59×10^{18} n/cm^2 for the BFN, Unit 2 and 3 vessels. Peak fluences were calculated at the vessel inner surface (inner diameter), for purposes of evaluating USE. The value of neutron fluence was also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter using Equation 3 from Paragraph 1.1 of Regulatory Guide (RG) 1.99, Revision 2. This 1/4T depth is recommended in the ASME Section XI, Appendix G, subarticle G-2120 as the maximum postulated defect depth. The applicant evaluated the EOL USE by an EMA using the 54 EFPY calculated fluence for Unit 1 and the 52 EFPY calculated fluence for Units 2 and 3 respectively. As documented in the staff’s SER, BWRVIP-74-A provided a generic EMA which demonstrated that BWR/3-6 plates and BWR/2-6 welds showing percent of reductions in USE of equal to or less than 23.5% and 39%, respectively, would meet the the requirements of 10 CFR Part 50, Appendix G. The applicant provided results of the EMA for limiting welds and plates on the three BFN RVs, and they are summarized in Tables 4.2.1.1 through 4.2.1.6 of the LRA. The applicant stated that the results are acceptable because the limiting USE percent drop is less than the BWRVIP-74-A percent drop acceptance criterion in all cases.

4.2.1.2 Staff Evaluation

Section IV.A.1.a of Appendix G to 10 CFR Part 50 requires, in part, that the RPV beltline materials have Charpy USE values in the transverse direction for base metal and along the weld for weld material of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will ensure margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

By letter dated April 30, 1993, the Boiling Water Reactor Owners Group (BWROG) submitted NEDO-32205-A to demonstrate that BWR RPVs could meet margins of safety against fracture equivalent to those required by Appendix G of the ASME Code Section XI for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrated that the evaluated materials have the margins of safety against fracture equivalent to Appendix G of ASME Code Section XI, in accordance with Appendix G of 10 CFR Part 50. In that report, the BWROG derived through statistical analysis the unirradiated USE values for materials that originally did not have documented unirradiated Charpy USE values. Using these statistically-derived Charpy USE values, the BWROG predicted the USE values through 40 years of operation in accordance with RG 1.99, Revision 2. According to this RG, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE value in the transverse direction for base metal and along the weld for weld material was 35 ft-lb.

General Electric (GE) performed an update to the USE EMA, which is documented in Electric Power Research Institute (EPRI) TR-113596, "BWR Vessel and Internals Project (VIP) BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, September 1999. The staff review and approval of EPRI TR-113596 was documented in a letter from Mr. C.I. Grimes to Mr. C. Terry dated October 18, 2001. The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron irradiation using the methodology in RG 1.99, Revision 2. Using this methodology and a correction factor of 65 percent for conversion of the longitudinal properties to transverse properties, the lowest Charpy USE at 54 EFPY for all BWR/3-6 plates was projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on NRC Branch Technical Position MTEB 5-2. The EMA acceptance criteria specified in the staff approved report BWRVIP-74, "BWR Vessel and Internals Project (BWRVIP), BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines" are based on the percent reduction in the unirradiated charpy USE values resulting from neutron radiation using the methodology in RG 1.99, Revision 2. The acceptance criteria that are specified in the BWRVIP-74 report indicate that the maximum allowable percent reduction in USE value for the plates is 23.5, and for the welds is 39.

Since the analysis in the BWRVIP-74 is a generic analysis, the applicant submitted plant-specific information in LRA Tables 4.2.1.1 through 4.2.1.6 for BFN, Units 1, 2 and 3, to demonstrate that the limiting beltline materials of the BFN RVs meet the criteria in the BWRVIP-74 report at the end of the license renewal period. In RAI-4.2.1, the staff requested that the applicant provide the initial USE values, percent reduction in USE values, percentage of copper, and 1/4 T fluence at the end of the extended period of operation for all the plates and weld metals in the beltline region of BFN, Units 1, 2 and 3. The applicant in response to the RAI-4.2.1 stated that the initial USE values are not available for the BFN units; however, BFN has used the EMA method to demonstrate that the BFN vessels will maintain adequate fracture toughness throughout the extended period of operation. The LRA bounding value for EFPY for BFN, Unit 1 is 54 EFPY and for BFN, Units 2 and 3 is 52 EFPY. The values for all beltline materials for BFN, Units 1, 2, and 3 are listed in the Tables 4.2.1-1 through 4.2.1-3 of the applicant's response. The applicant stated that the percent reduction in the USE value for the limiting beltline material for all the BFN units is less than the aforementioned acceptance criteria specified in BWRVIP-74. The staff has verified the copper contents given in Tables 4.2.1-1 to 4.2.1-3 of the applicant's response for all the beltline materials with the corresponding data in Reactor Vessel Integrity Data Base (RVID) and finds them acceptable. The staff also verified the reduction in the unirradiated USE values resulting from neutron radiation using the methodology in RG 1.99, Revision 2, and finds that all the beltline materials meet the acceptance criteria specified in the staff approved report BWRVIP-74, and Appendix G of 10 CFR Part 50.

4.2.1.3 FSAR Supplement

Section A.3.1.1 of the LRA includes the following FSAR Supplement summary description for the TLAA on USE:

> 10 CFR 50 Appendix G requires the predicted end-of-life Charpy impact test upper shelf energy for reactor vessel materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. The upper shelf energy is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. The 60 year end-of-life upper shelf energy was evaluated for the BFN reactor vessels by using an EMA methodology approved by the NRC in NEDO-32205-A, "10 CFR 50 Appendix G EMA for Low Upper-Shelf Energy in BWR-2 Through BWR-6 Vessels." The results show that the limiting upper shelf energy EMA percent is less than the EPRI Report TR-113596, BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Inspection and Flaw Evaluation Guidelines (BWRVIP-74) EMA percent acceptance criterion in all cases, and is therefore acceptable.

The applicant's FSAR Supplement Summary description is consistent with the staff analysis for the TLAA on USE in Section 4.2.1.2 of this SER. The FSAR Supplement summary description summarizes the applicable USE requirements that must be met to ensure continued compliance with 10 CFR Part 50, Appendix G. It also states that the RV beltline materials at BFN, Units 1, 2 and 3 will be in compliance with the applicable requirements in 10 CFR Part 50, Appendix G, as projected through the expiration of the extended periods of operation for the units. The staff therefore concludes that FSAR Supplement summary description for the TLAA on USE is acceptable.

4.2.1.4 Conclusion

The staff has reviewed the applicant's TLAA on USE, as summarized in Section 4.2.1 of the LRA, and has determined that the RV beltline materials at BFN, Units 1, 2 and 3 will continue to comply with the staff's USE requirements of 10 CFR Part 50, Appendix G throughout the extended periods of operation for the BFN units. The staff therefore concludes that the applicant's TLAA for USE is in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(ii) and that the safety margins established and maintained during the current operating term will be maintained during the periods of extended operation as required by 10 CFR 54.21(c)(1). The staff also concludes that the FSAR Supplement contains an appropriate summary description of the TLAA on USE for the period of extended operation, as required by 10 CFR 54.21(d).

4.2.2 Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement

4.2.2.1 Summary Description

In Section 4.2.2 of the LRA, the applicant summarized the ART determination for the vessel materials due to neutron embrittlement. The ART is defined as the sum of the initial RT_{NDT}, delta RT_{NDT}, and a margin term. The margin is defined in Regulatory Guide 1.99, Revision 2. As addressed in RG 1.99, Revision 2, delta RT_{NDT} is a function of neutron fluence. Since neutron fluence changes with time, the determination of delta RT_{NDT} (and, therefore, ART) meets the criteria of 10 CFR Part 54.3(a) for being a TLAA.

As described in UFSAR Section 4.2, the RVs were licensed for 40 years with an assumed neutron exposure of less than 10^{19} n/cm^2 (E > 1.0 MeV). The applicant stated that the current licensing basis calculations use calculated fluences that are lower than this limiting value. The design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40-year license term for all three units. The ART values were determined using the embrittlement correlations defined in Regulatory Guide 1.99, Revision 2.

The applicant calculated fluences for the RVs for the extended 60-year [54 EFPY for Unit 1; 52 EFPY for Units 2 and 3] licensed operating periods using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," which was approved by the staff in a SER dated September 14, 2001. One bounding calculation was performed for each BFN reactor vessel. Peak fluences which included consideration of EPU conditions were calculated at the vessel inner surface (inner diameter) for purposes of evaluating USE and ART. The neutron fluence values were also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter using equation 3 from Paragraph 1.1 of Regulatory Guide 1.99, Revision 2. This 1/4T depth is recommended in the ASME Code Section XI, Appendix G, Subarticle G-2120 as the maximum postulated defect depth. The applicant calculated ART values for beltline materials 54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3) based on the embrittlement correlation found in Regulatory Guide 1.99, Revision 2. The peak fluence, and ART values for the 60 year (54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3)) license operating period are presented in LRA Table 4.2.2-1. The applicant claimed that the limiting ARTs allow P-T limits that will provide reasonable operational flexibility.

4.2.2.2 Staff Evaluation

The applicant calculated the 54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3) fluences for the BFN RVs using the methodology of NEDC-32983P. Since this methodology is approved by the NRC, the calculated fluences provided in the LRA are acceptable. The applicant provided the results for one bounding calculation for each RV and determined the peak surface fluence of 1.95×10^{18} n/cm^2 and peak 1/4T fluence of 1.35×10^{18} n/cm^2 for BFN, Unit 1 vessel, and peak surface fluence of 2.3×10^{18} n/cm^2 and peak 1/4T fluence of 1.59×10^{18} n/cm^2 for BFN, Unit 2 and 3 vessels. Table 4.2.2.1 of the LRA shows bounding fluence values for BFN, Units 1, 2 and 3 for 54, 52 and 52 EFPYs of the operation, respectively.

In RAIs 4.2.2 (A), and 4.2.2 (B), the staff requested that the applicant provide an explanation addressing the following issues:

(A) The staff requested that the applicant explain why BFN Unit 1 was assumed to achieve 54 EFPYs of operation in a 60 year span given its operating history. Additionally, the staff requested that the applicant provide an explanation for having a peak surface fluence value of 1.95×10^{18} n/cm^2 (E > 1.0 MeV) for BFN, Unit 1, while, the BFN, Units 2 and 3 achieve 2.3×10^{18} n/cm^2 (E > 1.0 MeV) at the end of 60 years.

After reviewing the applicant's response to RAI 4.2.2 (A), the staff determined that the applicant performed fluence calculations for BFN, Unit 1 assuming 54 EFPY of operation and for BFN, Units 2 and 3 assuming 52 EFPY of operation. Based on the peak surface and 1/4 T fluence values, the applicant calculated USE and ART values for the limiting beltline material for each unit. The applicant stated that the reason the reported peak fluence for BFN, Unit 1 is lower than the fluence values for BFN, Units 2 and 3 is because the maximum delta RT_{NDT} and ART occurs in the circumferential weld material for BFN, Unit 1, which is located away from the peak vessel fluence location.

Whereas both BFN, Units 2 and 3 maximum delta RT_{NDT} and ART occurs in the axial weld materials, which corresponds to the peak fluence. Therefore, the reported peak fluence for BFN, Unit 1 has an axial correction factor of 0.81 applied and BFN, Units 2 and 3 do not have the axial correction factor. The applicant also indicated that 54 EFPY was selected for BFN units as a bounding value as part of the EPU evaluation. For consistency with the EPU evaluation, the 54 EFPY value was incorporated into the LRA. The ART values are listed Tables 4.2.2-1 through 4.2.2-6 of the applicant's response.

The staff reviewed the applicant's response and finds that the applicant's explanation for using the fluence values cited for BFN, Units 1,2 and 3 to be acceptable because it accounts for differences in weld location and neutron flux for each unit. The staff finds that this approach is acceptable as it identifies the maximum ART values for all three units.

(B) The staff requested that the applicant provide the initial RT_{NDT} and ART values at 1/4 T and vessel ID surface, at the end of the extended period of the operation for all the materials in the beltline region of the BFN reactor vessels.

The applicant provided information on the aforementioned items in Tables 4.2.2-1 to 4.2.2-6 of its response. The staff has verified the percentages of copper and nickel and the initial RT_{NDT} given in of the applicant's response for all the beltline materials with the corresponding data in RVID and finds them acceptable. The staff also verified the accuracy of the ART values for all the beltline materials using the methodology in RG 1.99, Revision 2 and finds them acceptable.

4.2.2.3 FSAR Supplement

Section A.3.1.2 of the LRA includes the following FSAR Supplement summary description for the TLAA on ART for RPV materials due to neutron embrittlement:

> 10 CFR 50 Appendix G defines the fracture toughness requirements for the life of the reactor pressure vessel. The initial nil-ductility reference temperature (RT_{NDT}) is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics going from ductile to brittle behavior. An increase RT_{NDT} means that higher temperatures are required for the material to continue to act in a ductile manner. The adjusted reference temperature (ART) is defined as RT_{NDT} +delta RT_{NDT} + margin. The 60 year end-of-life delta RT_{NDT} for each BFN reactor pressure vessel beltline materials was calculated based on the embrittlement correlation found in Regulatory Guide 1.99. The calculation results show that the limiting 60 year end-of-life ARTs allow pressure-temperature limits that will provide reasonable operational flexibility.

The staff concludes that the applicant used the staff approved methods for calculating the ART values for the BFN units. The staff concludes the proposed summary description provides an acceptable basis on how the 54 EFPY (Unit 1) and 52 EFPY (Units 2 and 3) ART values will be applied to the P-T limit calculations. Based on this assessment, the staff concludes that the FSAR Supplement Summary description for the TLAA on the adjusted reference temperature calculations is acceptable.

4.2.2.4 Conclusion

The staff has reviewed the applicant's TLAA on the calculation of ART values, as summarized in Section 4.2.2 of the LRA and the RAI response dated January 31, 2005, and has determined that the applicant's calculation of the ART values for the RV beltline materials, as projected through the periods of extended operation for BFN, Units 1, 2 and 3, is in conformance with the recommended guidelines of RG 1.99, Revision 2. The staff therefore concludes that the applicant's TLAA for calculation of the ART values is in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(ii) and that the safety margins established and maintained during the current operating term will be maintained during the periods of extended operation as required by 10 CFR 54.21(c)(1). The staff also concludes that the FSAR Supplement contains an appropriate summary description of the TLAA on ART calculations for the period of extended operation, as required by 10 CFR 54.21(d).

4.2.3 Reflood Thermal Shock Analysis of the Reactor Vessel

4.2.3.1 Summary Description

The applicant stated that UFSAR Section 3.3.5 includes an end-of-life thermal shock analysis performed on the RVs for a design basis loss of coolant accident (LOCA) followed by a low pressure coolant injection. The effects of embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. The applicant stated that this analysis satisfies the criteria of 10 CFR 54.3(a). As such, this analysis is a TLAA.

For the current operating period, a thermal shock analysis was originally performed on the RV components. The analysis assumed a design basis LOCA followed by a low pressure coolant injection and accounted for the full effects of neutron embrittlement at the end of the current license term of 40 years. The current analysis assumes end-of-life material toughness, which in turn depends on end-of-life ART. The critical location for fracture mechanics analysis is at 1/4 of the vessel thickness (from the inside, 1/4T). For this event, the peak stress intensity occurs at approximately 300 seconds after the LOCA. The applicant stated that the analysis shows that at 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (which is 1/4T) is approximately 400EF. The ART values described in Section 4.2.2 and tabulated in Table 4.2.2.1 list the ARTs for the limiting weld metal of the RVs. The highest calculated RV beltline material ART value is 167.7EF (Unit 1). Using the equation for K_{IC} presented in Appendix A of ASME Section XI and the maximum ART value, the material reaches upper shelf (a K_{IC} value of 200 ksi%in) at 272°F, which is well below the 400°F 1/4T temperature predicted for the thermal shock event at the time of peak stress intensity. Therefore, the applicant claimed that the projected analysis is valid for the period of extended operation.

4.2.3.2 Staff Evaluation

The analysis assumes end-of-life material toughness, which in turn depends on end-of-life ART. The critical location for fracture mechanics analysis is at the 1/4T location. For the reflood thermal shock analysis of the RV, the peak stress intensity occurs at approximately 300 seconds after the LOCA. At that time, the temperature at 1/4T is approximately 400 EF, which is much higher than the 54-EFPY ART 167.7EF for the limiting material of all the three BFN vessels. Therefore, the staff concurs with the applicant that the revised thermal shock analysis of the BFN vessels is valid for the period of extended operation because, the ART for the limiting beltline plate material is 167.7EF for BFN Unit 1, which is below the 400 EF at 1/4 T temperature predicted from the thermal shock event at the time of peak stress intensity. The reflood thermal shock analysis is, therefore, bounding and valid for the period of extended operation.

4.2.3.3 FSAR Supplement

Section A.3.1.3 of the LRA includes the following FSAR Supplement summary description for the TLAA on reflood thermal shock analysis of the RV:

> The UFSAR section 3.3.5 includes an end-of-life thermal shock analysis performed on the RVs for a design basis LOCA followed by a low pressure coolant injection. The effects of embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. A revised analysis shows that the minimum temperature predicted for the thermal shock event at the time and location of peak

stress intensity remains well above the limiting adjusted reference temperature (ART) during the period of extended operation.

The staff concludes that the applicant adequately addresses in the UFSAR supplement the effect of reflood thermal shock due to LOCA on the fracture toughness of the most limiting beltline material. The applicant in its analysis demonstrates that the most limiting beltline material would maintain its fracture toughness during the extended period of operation when exposed to stresses due to reflood thermal shock as a result of LOCA.

4.2.3.4 Conclusion

The staff has reviewed the applicant's TLAA on reflood thermal shock analysis of the RV for a design basis LOCA and concludes that the applicant demonstrated that the limiting beltline material will have adequate fracture toughness when exposed to stresses due to reflood thermal shock due to LOCA. The staff determines that this revised analysis for the extended period of operation is in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(ii) and that the safety margins established and maintained during the current operating term will be maintained during the periods of extended operation as required by 10 CFR 54.21(c)(1).

4.2.4. Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud

4.2.4.1 Summary Description:

In Section 4.2.4 of the LRA, the applicant stated that the radiation embrittlement may affect the ability of RV internals, particularly the core shroud, to withstand a low-pressure coolant injection thermal shock transient. The applicant stated that the analysis of core shroud strain due to reflood thermal shock is based on the calculated lifetime neutron fluence. In the thermal shock analysis of the BFN RV core shrouds, the applicant considered the location on the inside surface of the core shroud opposite to the midpoint of the fuel centerline as a location most susceptible to damage during an LPCI thermal shock transient because it receives the maximum irradiation. This analysis satisfies the criteria of 10 CFR 54.3(a). As such, this analysis is a TLAA.

The applicant stated that it used the approved fluence methodology discussed in Section 4.2.2, and the 54 EFPY fluence at the most irradiated point on the core shroud was calculated to be 5.34×10^{21} n/cm^2 (E>1 MeV) for BFN Units. The maximum thermal shock stress due to LPCI transient in this region will be 155,700 psi equivalent to 0.57% strain. This strain range of 0.57% was calculated at the midpoint of the shroud when it is exposed to 54 EFPY fluence. The applicant compared the calculated strain range with the measured values of percent of elongation for annealed Type 304 stainless steel irradiated to 8×10^{21} n/cm^2 (E>1 MeV). The measured value of percent elongation for stainless steel weld metal is 4% for a temperature of 297EC (567°F) with a neutron flux of 8×10^{21} n/cm^2 (E>1 MeV), while the average value for base metal at 290EC (554°F) is 20%. The applicant concluded that the measured value of elongation bounds the calculated thermal shock strain amplitude of 0.57%, the calculated thermal shock strain at the most irradiated location is acceptable considering the embrittlement effects for the extended period of operation.

4.2.4.2 Staff Evaluation

In the thermal shock analysis of BFN RV core shrouds, the applicant considered the location on the inside surface of the core shroud opposite to the midpoint of the fuel centerline as a location most susceptible to damage during an LPCI thermal shock transient because it receives the maximum irradiation. This fluence is calculated using the methodology of NEDC-32983P, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation," which is approved by the NRC. The staff issued the following RAIs (4.2.4-1) requesting the applicant to address the basis for its analysis related to the reflood thermal shock of the RPV core shroud:

RAI 4.2.4-1(A): In LRA Section 4.2.4, “Reflood Thermal Shock Analysis of the RV Core Shroud and Repair Hardware,” the applicant states that the total integrated neutron flux at the end of 54 EFPY at the shroud inside surface is expected to be 5.34×10^{21} n/cm^2 (E > 1 MeV). The staff requests that the applicant provide an explanation whether this value is bounding at the inside shroud surface for all the three Units. If so, submit information whether the neutron fluence values are estimated based on the implementation of EPU.

The applicant in its response stated that the calculation of shroud fluence, 5.34×10^{21} n/cm^2 (E > 1 MeV) is based on the inner diameter peak flux of 3.14×10^{12} n/cm^2-sec (E > 1 MeV) for 54 EFPY, which is the lifetime used for BFN Unit 1. Since lifetime used for BFN Units 2 and 3 is 52 EFPY, 5.34×10^{21} n/cm^2 (E > 1 MeV) fluence from Unit 1 is bounding for all the BFN units. The fluence value for the shroud inner diameter was based on the implementation of EPU conditions. The staff after the review concurs with the applicant, and accepts the conservative bounding fluence value of 5.34×10^{21} n/cm^2 (E > 1 MeV) for all the three units.

RAI 4.2.4-1(B): This RAI and the applicant’s response are addressed in Section 4.2.8.2 under core shroud subsection of this SER.

RAI 4.2.4-1(C): The applicant calculated thermal strain resulting from the low-pressure coolant injection reflood thermal shock transient in the core shroud region. The applicant compared the calculated thermal strain with the measured values of per cent elongation of annealed type 304 stainless steel irradiated to 8×10^{21} n/cm^2 (E > 1.0 MeV). In a previous analysis performed by Dresden/Quad Cities, the applicant used the percent reduction in area as a criterion to evaluate the thermal strain. The staff requests that the applicant for BFN units, provide information on the measured percent reduction in area values for the irradiated type 304 stainless steel. The applicant should compare the results of the analysis obtained from using the reduction in area, with the ones using the percent elongation, and justify which of these properties is more appropriate to use in evaluating the local thermal shock strain associated with the reflood thermal shock event at the most irradiated core shroud region.

The applicant submitted the following reduction in area and elongation values for irradiated stainless steel materials:

Reduction in Area

Fluence (n/cm^2, E>1MeV)	Test Temperature (^{0}F)	Reduction in Area (%)
1×10^{21}	550	40
6.9×10^{21}	750	52.5

Elongation

Material	Fluence n/cm^2, (E>1MeV)	Test Temperature (^{0}F)	Elongation (%)
Base	8 X 10^{21}	554	20
Weld	8 X 10^{21}	567	4

The applicant stated that the bounding shroud fluence (BFN Unit 1) is 5.34 x 10^{21} n/cm^2 (E>1 MeV) for all the three BFN units, and the listed ductility values bound all three BFN shrouds. As described in LRA Section 4.2.4, the maximum thermal shock stress results in a calculated thermal shock strain amplitude of 0.57%. Both reduction in area and elongation values which are values at failure are significantly in excess of the calculated thermal shock strain at the most irradiated location. While the analysis indicates that either measure of ductility is acceptable for the period of extended operation, reduction in area is a more appropriate measure of ductility for the reflood thermal shock event. The strain associated with the reflood thermal shock event is very localized and is constrained by the surrounding bulk material. As such, it is similar to the triaxial stress condition present in the neck region (where the area reduction is taking place) during a tensile test. The percent reduction in area is a measure of this triaxial stress state and, as such, is the most appropriate property for evaluating the effect of thermal shock on the shroud. The staff agrees with the applicant's justification for using percent reduction in area as the material property for the evaluation of the thermal shock strain because, this property represents the localized strain condition constrained by the bulk material. This condition is similar to the triaxial stress condition that is present in the area of reduction which is conservative. In addition, this analysis is similar to the one that was previously approved by the staff in the LRA of Dresden/Quad Cities. The staff concludes that the thermal shock strain associated with LOCA is less than the reduction in area or elongation, which would be expected to fail the shroud at the highest fluence point. Therefore, the staff determines that the core shroud will have sufficient ductility during the reflood thermal shock transient during the extended period of operation. The staff accepts the applicant's analysis for the BFN units.

4.2.4.3 UFSAR Supplement

Section A.3.1.4 of the LRA includes the following FSAR Supplement summary description for the TLAA on reflood thermal shock analysis of the RV core shroud:

> The RV core shrouds were evaluated for a low pressure coolant injection reflood thermal shock transient considering the embrittlement effects of 40-year radiation exposure (32 EFPY, Effective Full Power Year). The analysis was revised for the 60-year radiation exposure using the approved fluence methodology described in Section A.3.1.2. The results show that the calculated thermal shock strain at the most irradiated location is acceptable considering the embrittlement effects for a 60-year operating period.

The staff believes that the applicant adequately addresses in the UFSAR supplement the effect of reflood thermal shock due to LOCA on the fracture toughness of the most limiting core shroud material. The applicant in its analysis demonstrates that the most limiting core shroud material would maintain its fracture toughness during the extended period of operation when exposed to stresses due to reflood thermal shock as a result of LOCA.

4.2.4.4 Conclusion

The staff has reviewed the applicant's TLAA on reflood thermal shock analysis of the RV core shroud and concludes that the applicant demonstrated that the calculated thermal shock strain at the most irradiated portion of the core shroud is acceptable. The staff also accepts the applicant's conservative methodology in establishing the integrity of the most irradiated location of the core shroud during a low pressure coolant thermal shock event. The staff determines that the revised analysis for the extended period of operation is in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(ii) and that the safety margins established and maintained during the current operating term will be maintained during the periods of extended operation as required by 10 CFR 54.21(c)(1).

4.2.5 Reactor Vessel Thermal Limit Analyses—Operating Pressure-Temperature Limits

4.2.5.1 Summary Description

In Section 4.2.5 of the LRA, the applicant addressed the RV thermal limit analysis. The ART is the sum of initial RT_{NDT} + delta RT_{NDT} + margins for uncertainties at a specific location. Neutron embrittlement increases the ART. Thus, the minimum metal temperature at which a RV is allowed to be pressurized increases. The ART of the limiting beltline material is used to correct the beltline P-T limits to account for irradiation effects. The 10 CFR Part 50, Appendix G requires RV thermal limit analyses to determine operating pressure-temperature (P-T) limits for three categories of operation: 1) hydrostatic pressure tests and leak tests, referred to as Curve A; 2) non-nuclear heat-up / cooldown and low-level physics tests, referred to as Curve B; and 3) core critical operation, referred to as Curve C. P-T limits are developed for three vessel regions: the upper vessel region, the core beltline region, and the lower vessel bottom head region. The calculations associated with generation of the P-T curves satisfy the criteria of 10 CFR 54.3(a). As such, this topic is a TLAA.

The applicant stated that the BFN Technical Specifications Section 3.4.9 contains P-T limit curves for heat up, cooldown, criticality, and inservice leakage and hydrostatic testing. According to the applicant, limits are also imposed on the maximum rate of change of reactor coolant temperature. The P-T limit curves are currently calculated for 12 EFPY (Unit 1), 17.2 EFPY (Unit 2) and 13.1 EFPY (Unit 3) operating periods. The applicant stated that new P-T limits will be calculated and submitted for approval prior to the start of extended operation.

4.2.5.2 Staff Evaluation

The applicant plans to calculate vessel P-T limit curves for all BFN units and submit them to the NRC for approval before the start of the extended period of operation using an approved fluence methodology. By letter dated December 6, 2004, the applicant submitted updated of P-T curves for BFN, Unit 1 which are currently being reviewed by the staff. The applicant stated that the P-T curves for BFN, Units 2 and 3 were approved by the staff as documented in Safety Evaluations dated March 10, 2004. The applicant's current licensing basis allows the development of P-T limit curves consistent with the 2000 Edition, 2001 Addenda of the ASME Section XI code. The applicant stated that it will manage the P-T limits using approved fluence calculations when there are changes in power of core design in conjunction with surveillance capsule results from the BWRVIP integrated surveillance program. The staff finds the applicant's plan to manage the P-T limits acceptable because the change in P-T curves will be implemented by the license amendment process (i.e., modifications of technical specifications) and will meet the requirements of 10 CFR 50.60 and Appendix G to 10 CFR 50.

4.2.5.3 UFSAR Supplement

Section A.3.1.5 of the LRA includes the following FSAR Supplement summary description for the TLAA on RPV thermal limit analyses-operating temperature and pressure limits:

> 10 CFR Part 50 Appendix G requires RV thermal limit analyses to determine operating pressure-temperature limits for heatup, cooldown, criticality, and inservice leakage and hydrostatic testing. Because of the relationship between the operating pressure temperature limits and the fracture toughness transition of the RV, all three units will require new operating pressure-temperature limit curves to be calculated and approved for the extended period of operation.

The applicant's FSAR supplement summary description for the TLAA on the P-T limits appropriately describes that the applicant will revise the P-T limits for the extended periods of operation for the BFN units. Since the FSAR Supplement summary description adequately describes the TLAA, the staff concludes that the FSAR supplement summary description for the TLAA on the P-T limits is acceptable.

4.2.5.4 Conclusion

The staff has reviewed the applicant's TLAA on P-T limits, as summarized in Section 4.2.5 of the LRA and has determined that the applicant will generate the P-T limits for the extended periods of operation for BFN, Units 1, 2 and 3. The staff therefore concludes that the applicant's TLAA for the BFN P-T limits will be in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(iii) when the P-T limits for the periods of extended operation are generated and incorporated into the BFN technical specifications and that the safety margins established and maintained during the current operating term will be maintained during the periods of extended operation as required by 10 CFR 54.21(c)(1). The staff also concludes that the FSAR Supplement contains an appropriate summary description of the TLAA on P-T limits for the period of extended operation, as required by 10 CFR 54.21(d).

4.2.6 Reactor Vessel Circumferential Weld Examination Relief

4.2.6.1 Summary Description

Section 4.2.6 and Appendix A.3.1.6 of the LRA discuss inspection of the BFN RV circumferential welds. These sections of the LRA indicate that the applicant will use an approved relief from ultrasonic testing of RV circumferential shell welds. The applicant stated that the relief from RV circumferential weld examination requirements under Generic Letter 98-05 is based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on RV metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period. The applicant stated that BFN, Units 2 and 3 have received this relief for the remainder of their current 40 year licensed operating periods. BFN, Unit 1 has submitted a relief request (currently under review by the staff) for the remainder of its 40 year licensed operating period. The circumferential weld examination relief analyses meet the requirements of 10 CFR 54.3(a). As such, they are a TLAA.

The basis for this relief request was an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on topical report BWRVIP-05, "Reactor Vessel Shell Weld Inspection Guidelines," and the extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for the period of extended operation and approval by the NRC to extend this relief request.

The NRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis were: 1) the neutron fluence was the estimated end-of-license mean fluence, 2) the chemistry values were mean values based on vessel types, and 3) the potential for beyond-design-basis events was considered. LRA Table 4.2.6.1 provides a comparison of the BFN, Units 2 and 3 RV limiting circumferential weld parameters to those used in the NRC evaluation of BWRVIP-05 for the first two key assumptions. Data provided in LRA Table 4.2.6.1 was supplied from Tables 2.6.4 and 2.6.5 of the Final Safety Evaluation of the BWRVIP-05 report.

For BFN, Units 2 and 3, the fluence is equivalent to that used in the NRC analysis. However, the BFN, Units 2 and 3 weld materials have significantly lower copper values (0.09 vs. 0.31) than those used in the NRC analysis. As a result, the shifts in reference temperature for BFN, Units 2 and 3 are lower than the 64 EFPY shift from the NRC SER analysis. In addition, the unirradiated reference temperatures for both units are significantly lower. The combination of initial RT_{NDT} and delta RT_{NDT} without margin yields mean RT_{NDT} values for BFN, Units 2 and 3 that are considerably lower than the NRC mean analysis values. Based on this analysis, the applicant concluded that the BFN RPV conditional failure probability is bounded by the NRC analysis. The applicant claimed that the procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when BFN requested the relief for the current license term for BFN, Units 2 and 3.

4.2.6.2 Staff Evaluation

The technical basis for relief is discussed in the staff's final SER concerning the BWRVIP-05 report, which is enclosed in a July 28, 1998, letter from Mr. G.C. Laines, NRC to Mr. C. Terry, the BWRVIP Chairman. In this letter, the staff concluded that since the failure frequency for circumferential welds in BWR plants is significantly below the criterion specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and below the core damage frequency (CDF) of any BWR plant, the continued inspection would result in a negligible decrease in an already acceptably low value of RV failure. Therefore, elimination of the inservice inspection (ISI) for RPV circumferential welds is justified. The staff's letter indicated that BWR applicants may request relief from ISI requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating that (1) at the expiration of the license, the circumferential welds satisfy the limiting conditional failure probability for circumferential welds in the NRC staff's July 28, 1998 evaluation, and (2) the applicants have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the frequency specified in the staff's SER. The letter indicated that the requirements for inspection of circumferential RV welds during an additional 20-year license renewal period would be reassessed, on a plant-specific basis, as part of any BWR LRA. Therefore, the applicant must request relief from inspection of circumferential welds during the license renewal period per 10 CFR 50.55a.

Section A.4.5 of the BWRVIP-74 report indicates that the staff's SER of the BWRVIP-05 report conservatively evaluated the BWR RVs to 64 EFPY, which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. The NRC staff used the mean RT_{NDT} value for materials to evaluate failure probability of BWR circumferential welds at 32 and 64 EFPY in the staff SER dated July 28, 1998. The neutron fluence used in this evaluation was the neutron fluence at the clad-weld (inner) interface.

Since the staff analysis discussed in the BWRVIP-74 report is a generic analysis, the applicant submitted plant-specific information to demonstrate that the BFN beltline materials meet the criteria specified in the report. To demonstrate that the BFN vessels for Units 2 and 3 have not become embrittled beyond the basis for the relief, the applicant, in LRA Table 4.2.6.1, supplied a comparison of 52 EFPY material data for the limiting BFN circumferential welds with that of the 64 EFPY reference case in Appendix E of the staff's SER of the BWRVIP-05 report. The BFN material data included amounts of copper and nickel, chemistry factor, the neutron fluence, delta RT_{NDT}, initial RT_{NDT}, and mean RT_{NDT} of the limiting circumferential weld at the end of the renewal period. The staff has verified the data for the copper and nickel contents and the initial RT_{NDT} values for BFN, Unit 2 and 3 beltline materials by comparing them with the corresponding data in the Reactor Vessel Integrity Data Base (RVID) database maintained by the NRC. The 52 EFPY mean RT_{NDT} value for BFN, Units 2 and 3 is 25EF. The staff has checked the applicant's calculations for the 52 EFPY mean RT_{NDT} values for the BFN circumferential welds using the data presented in LRA Table 4.2.6.1 and found them accurate. These 52 EFPY mean RT_{NDT} values for BFN, Units 2 and 3 are less than the 64 EFPY mean RT_{NDT} value of 129.4EF used by the NRC for determining the conditional failure probability of a circumferential weld. The 64 EFPY mean RT_{NDT} value from the staff SER dated July 28, 1998, is for a B&W weld, because B&W welded the circumferential welds in the BFN vessels. Since the BFN 52 EFPY mean RT_{NDT} values are less than the 64 EFPY value from the staff SER dated July 28, 1998, the staff concludes that the BFN RV conditional failure probabilities are bounded by the NRC analysis.

The applicant stated that the procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when BFN requested the relief for the current license period, but it did not explicitly cite a document that supports this statement. The applicant stated that the procedure and training requirements identified in the BFN request to use the BWRVIP-05 report are provided in the document, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Alternative to Inspection of Reactor Pressure Vessel Circumferential Welds, BFN Power Station, Units 2 and 3 (attached to NRC Letter to TVA "Browns Ferry Nuclear Plant Unit 2, Relief Request 2-ISI-9, Alternatives for Examination of Reactor Pressure Vessel Shell Welds (TAC No. MA8424)," August 14, 2000, and NRC letter to TVA, "Browns Ferry Nuclear Plant Unit 3, Relief Request 3-ISI-1, Revision 1, Alternatives for Examination of Reactor Pressure Vessel Shell Welds (TAC No. MA5953)," November 18, 1999. The applicant further stated that LRA Section 4.2.6, and associated UFSAR Supplement Section A.3.1.6, reference the Safety Evaluation request letters identified above. The staff finds the response acceptable because the applicant identifies the requested references and commits to include them in LRA Section 4.2.6 and associated UFSAR Supplement Section A.3.1.6.

By letter dated May 12, 2004, the applicant submitted a relief request concerning the examination of the BFN Unit 1 RV circumferential welds for the current license period. The staff requested in a RAI-4.2.6-1 that the applicant provide the RV circumferential weld examination relief analyses for the BFN, Unit 1. The applicant submitted the following relief analyses related to the BFN, Unit 1 RV circumferential weld examination:

> The following table provides a comparison of the BFN Unit 1 RV limiting circumferential weld parameters to those used in the NRC evaluation of BWRVIP-05 for the first two key assumptions. Data provided in this table was supplied from Tables 2.6.4 and 2.6.5 of the Final Safety Evaluation of the BWRVIP-05 Report (NRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, “Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report,” (TAC No. M93925), July 28, 1998.
>
> The fluence assumed for Unit 1 is very conservative based on an extended shutdown period from 1985 to a scheduled restart in 2007, which will result in less than 32 EFPY of vessel exposure through the end of the extended period of operation. However, TVA conservatively chose to use the higher exposure of 54 EFPY to simplify the basis for the Unit 1 vessel evaluations. As shown in the table, the Unit 1 unirradiated weld RT_{NDT} is identical to the reference B&W plant unirradiated weld RT_{NDT} used in the NRC analysis, and the Unit 1 fluence value is approximately equivalent to that used in the NRC analysis. However, because the Unit 1 chemistry factor is less than the reference B&W plant, the mean RT_{NDT} values for Unit 1 at 54 EFPY are bounded by the 64 EFPY Mean RT_{NDT} assumed by the NRC in its analysis. Accordingly, Unit 1 is bounded by the conditional failure probability calculated by the Staff for the limiting B&W vessel. An extension of this relief for the 60-year period will be submitted to the NRC for approval prior to entering the period of extended operation.

Group	B & W 64 EFPY	BFN Unit 1 54 EFPY
Cu %	0.31	0.27
Ni %	0.59	0.60
CF	196.7	184
Fluence at clad/weld interface 10^{19} n/cm^2	0.19	0.2
Delta RT_{NDT} without margin (^{0}F)	109.4	104
Initial RT_{NDT} (^{0}F)	20	20
Mean RT_{NDT} (^{0}F)	129.4	124
P (F/E) NRC	4.83 X 10^{-4}	-------------------------
P (F/E) BWRVIP	---------------------	-----------------------

The staff verified the accuracy of the of the mean RT_{NDT} for the limiting beltline circumferential weld at the BFN, Unit 1 and finds it acceptable. In the staff’s evaluation of the BWRVIP-05 report a fluence of 0.19 x 10^{19} n/cm^2 for B&W RVs was used for 64 EFPY and the corresponding delta RT_{NDT} value is 109.4^0 F. The delta RT_{NDT} value for the limiting beltline weld metal of BFN, Unit 1 is less than the limiting delta RT_{NDT} value in the staff’s evaluation of BWRVIP-05 report, which is conservative. Therefore, the licensee’s calculated mean RT_{NDT} value for the limiting beltline weld metal is acceptable and meets the requirements specified in staff’s approved SER for the BWRVIP-05 report.

The staff’s SER for the BWRVIP-05 report provides a limiting conditional failure probability of 4.83 X 10^{-4} per-reactor year for a limiting plant-specific mean RT_{NDT} of 129.4EF for B&W

fabricated RVs. The low temperature over pressure transient (LTOP) frequency is the frequency of the transient occurring, determined as 10^{-3} per reactor-year in the evaluation of BWRVIP-05 report. The conditional failure probability is the probability of failure, if the event were to occur. The vessel failure frequency is the product of conditional failure probability and LTOP frequency. Comparing the information in the RVID with that submitted in the analysis, the staff confirmed that the mean RT_{NDT} of the circumferential welds at BFN, Unit 1 is projected to be 124EF at the end of the extended period of operation (54 EFPY). In this evaluation, the chemistry factor, delta RT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of RV Materials." Since the calculated value of mean RT_{NDT} for the circumferential welds at BFN, Unit 1 is lower than that for the limiting plant-specific case for B&W fabricated RVs, the vessel failure frequencies of the BFN, Unit 1 circumferential welds is less than 4.83×10^{-7} per reactor-year.

The staff finds that the applicant's evaluation for this TLAA is acceptable because the BFN 54 EFPY conditional failure probabilities for the RV circumferential welds are bounded by the NRC analysis in the staff SER dated July 28, 1998, and the applicant will be using procedures and training to limit cold over-pressure events during the period of extended operation. This analysis satisfies the evaluation requirements of the staff SER dated July 28,1998; however, the applicant is still required to request relief for the circumferential weld examination for extended period of operation in accordance with 10 CFR 50.55a.

4.2.6.3 UFSAR Supplement

Section A.3.1.6 of the LRA includes the following FSAR Supplement summary description for the TLAA on RV circumferential weld examination relief:

> Units 2 and 3 have received relief from RV circumferential weld examination requirements under Generic Letter 98-05 for the remainder of the 40 year licensed operating period. The circumferential weld examination relief analyses are based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on RV metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period. Although a conditional failure probability has not been recalculated, an analysis that concluded values at the end of a 60 year life are less than the 64 EFPY value provided by the NRC leads to the conclusion that the BFN RV conditional failure probability is bounded by the NRC analysis in its safety evaluation report (SER) for BWRVIP-05. The procedures and training used to limit cold over-pressure events will be the same as that approved by the NRC when BFN requested the BWRVIP-05 technical alternative be used for the current term for Units 2 and 3. An extension of this relief for the 60-year period will be submitted to the NRC for approval prior to entering the period of extended operation.

The applicant's FSAR supplement summary description for the TLAA on RV circumferential weld examination relief appropriately describes that the conditional failure probabilities for the RV circumferential welds are bounded by the NRC analysis in the staff SER dated July 28, 1998, and the applicant will be using procedures and training to limit cold over-pressure events during the period of extended operation for BFN, Units 2 and 3. Since the UFSAR Supplement summary description adequately describes the TLAA for BFN, Units 2 and 3, the staff concludes that the UFSAR supplement summary description for the TLAA on RV circumferential weld examination relief for BFN, Units 2 and 3 is acceptable. However, the UFSAR supplement summary description should include RV circumferential weld examination relief for the BFN Unit 1. **This is Open Item- 4.2.6.3-1**.

4.2.6.4 Conclusion

The staff has reviewed the applicant's TLAA on RV circumferential weld examination relief, as summarized in Section 4.2.6 of the LRA and has determined that the applicant appropriately describes that the conditional failure probabilities for the RV circumferential welds are bounded by the NRC analysis in the staff SER on the BWRVIP-05 report, dated July 28, 1998, and the applicant will be using procedures and training to limit cold over-pressure events during the period of extended operation for BFN Units 1, 2 and 3. However, the staff concludes that the UFSAR supplement A.3.1.6 should include circumferential weld examination analysis for the BFN, Unit 1. The staff therefore concludes that the applicant's TLAA Section 4.2.6, and UFSAR supplement A.3.1.6 (pending revision) for the BFN RV circumferential weld examination relief will be in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(ii), except as noted above.

4.2.7 Reactor Vessel Axial Weld Failure Probability

4.2.7.1 Summary Description

Section 4.2.7 of the LRA discusses the BWRVIP recommendations for inspection of RV shell welds and contains generic analyses supporting an NRC SER conclusion that the axial weld failure rate is no more than 5×10^{-6} per reactor year. The applicant stated that the supporting evaluations described in the LRA only apply to BFN, Units 2 and 3. The axial weld failure probability analysis meets the requirements of 10 CFR 54.3(a). As such, it is a TLAA.

The applicant compared the limiting axial weld properties at 52 EFPY for BFN, Units 2 and 3 with the limiting axial weld properties provided in the supplement to NRC SER for BWRVIP-05. The limiting axial welds at BFN, Units 2 and 3 are all electroslag welds with similar chemistry. The BFN, Units 2 and 3 limiting weld chemistry, chemistry factor, and 52 EFPY mean RT_{NDT} values are within the limits of the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 SER supplement. The applicant concluded that the probability of failure for the axial welds is bounded by the NRC evaluation.

4.2.7.2 Staff Evaluation

In its July 28, 1998 letter to Mr. C. Terry, the BWRVIP Chairman, the staff identified a concern about the failure frequency of axially-oriented welds in BWR RVs. In response to this concern, the BWRVIP supplied evaluations of axial weld failure frequency in letters dated December 15, 1998, and November 12, 1999. The staff's SER on these analyses is enclosed in a March 7, 2000 letter from Mr. J. Strosnider NRC to Mr. C. Terry, BWRVIP Chairman. The staff performed a generic analysis using Piligrim as a model for BWR RVs which were fabricated with electroslag welds, and demonstrated that a mean RT_{NDT} of 114^0 F resulted in a failure frequency of 5×10^{-6} per reactor-year of operation. The applicant calculated, and the staff confirmed, that the limiting axial weld mean RT_{NDT} value for BFN, Units 2 and 3 at 52 EFPY is 108^0 F, which supports the conclusion that the failure frequencies for BFN, Units 2 and 3 will be less than 5×10^{-6} per reactor-year of operation at the end of their period of extended operation. Therefore, this analysis is acceptable.

In RAI 4.2.7-1, the staff requested that the applicant provide an evaluation for the RV axial weld failure probability analyses for BFN, Unit 1 for the current license period, and the extended period of operation. In its response to RAI 4.2.7-1, by letter dated January 31, 2005, the applicant provided the following evaluation on the RV axial weld failure probability analysis for the BFN, Unit 1:

> The table provided below compares the limiting axial weld 54 EFPY properties for Unit 1 against the values taken from Table 2.6.5 found in the NRC SER for BWRVIP-05 and associated supplement to the SER (NRC letter from Jack R. Strosnider, to Carl Terry, BWRVIP Chairman, "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report," (TAC No. MA3395), March 7, 2000). The SER supplement required the limiting axial weld to be compared with data found in Table 3 of the document. For Unit 1 the comparison was made to the 'Mod 2' plant information. The supplemental SER stated that the 'Mod 2' calculations most closely match the 5 x 10^{-6} RV failure frequency.
>
> Effects of Irradiation on RV Axial Weld Properties BFN Unit 1:

Value	NRC BWRVIP-05 SER MOD 2	BFN Unit 1 54 EFPY
Cu %	0.219	0.24
Ni %	0.996	0.37
CF	------	141
Fluence at clad/weld interface 10^{19} n/cm^2	0.148 (Peak Axial Fluence)	0.24
ΔRT_{NDT} without margin (^{0}F)	116	86
$RT_{NDT(U)}$ (^{0}F)	-2	23
Mean RT_{NDT} (^{0}F)	114	109
P (F/E) NRC	5.02 X 10^{-6}	Not Calculated

> The limiting axial weld is an electroslag weld with similar chemistry. The Unit 1 limiting weld chemistry, chemistry factor, and 54 EFPY mean RT_{NDT} values are within the limits of the values assumed in the analysis performed by the NRC staff in the BWRVIP-05 SER supplement and the 64 EFPY limits and values obtained from Table 2.6.5 of the SER. Therefore, the probability of failure for the axial welds is bounded by the NRC evaluation.

In this evaluation, the chemistry factor, delta RT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of RV Materials." The applicant calculated, and the staff confirmed, that the limiting axial weld mean RT_{NDT} value for BFN, Unit 1at 54 EFPY is 109^0 F. This value is lower than that for the limiting mean RT_{NDT} value of 114EF in the staff's evaluation of BWRVIP-05. Therefore, the staff concludes that the failure frequencies for BFN, Unit 1 axial welds will be less than 5 x 10^{-6} per reactor-year of operation. The probability of failure for the axial welds is bounded by the staff evaluation.

4.2.7.3 UFSAR Supplement

Section A.3.1.7 of the LRA includes the following FSAR Supplement summary description for the TLAA on RV axial weld failure probability.

> The BWRVIP-05 recommendations for inspection of RV shell welds contain generic analyses supporting an NRC SER conclusion that the generic plant axial weld failure rate is no more than 5 x 10^{-6} per reactor year. BWRVIP-05 showed that this axial weld failure rate of 5 x 10^{-6} per reactor year is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described in A.3.1.6. The Units 2 and 3 limiting weld chemistry, chemistry factor and 60 year life mean RT_{NDT} values are within the limits of the values assumed in the analysis performed by the NRC staff in its BWRVIP-05 SER supplement. Therefore, the probability of failure for the axial welds is bounded by the NRC evaluation.

The applicant's UFSAR supplement summary description for the TLAA on RV axial weld failure probability appropriately describes that the conditional failure probabilities for the RV axial welds are bounded by the NRC analysis in the staff SER dated July 28, 1998 for the period of extended operation for BFN, Units 1, 2 and 3. Since the UFSAR Supplement summary description adequately describes the TLAA for BFN, Units 2 and 3, the staff concludes that the UFSAR supplement summary description for the TLAA on RV axial weld axial weld failure probability for BFN Units 2 and 3 is acceptable. However, the UFSAR supplement should include the analysis on the RV axial weld failure probability for the BFN, Unit 1 for the period of extended operation. **This is Open Item- 4.2.7.3-1**

4.2.7.4 Conclusions

The staff has reviewed the applicant's TLAA on the evaluation of RV axial weld failure probabilities, as summarized in Section 4.2.7 of the LRA, and has determined that the applicant appropriately describes that the analysis of the conditional failure probabilities for the BFN, Units 2 and 3 RV axial welds is bounded by the NRC analysis in the staff SER on the BWRVIP-05 report, dated July 28, 1998. However, UFSAR supplement summary description in Section A.3.1.7 of the LRA, should include the analysis on the conditional failure probabilities for the BFN, Unit 1 RV axial welds. The staff therefore concludes that the applicant's TLAA Section 4.2.7, and UFSAR supplement A.3.1.7 (pending their revision) related to the analysis of the conditional failure probabilities for the BFN units RV axial welds are acceptable. The staff concludes that the analysis of the RV axial weld failure probability for the BFN units will be in compliance with the staff's acceptance criterion for TLAAs in 10 CFR 54.21(c)(1)(ii), except as noted above.

4.2.8 Irradiation Assisted Stress Corrosion Cracking (IASCC) of RV Internals

4.2.8.1 Summary Description

The applicant in Section 4.7.6 of the LRA, provided the following description for the TLAA on IASCC in austenitic stainless steel RV internal components:

> Austenitic stainless steel reactor internal components exposed to neutron fluence greater than 5 x 10^{20} n/cm^2 (E > 1 MeV) are considered susceptible to Irradiation Assisted Stress Corrosion Cracking (IASCC) in the BWR environment. As described in the SER (ML003776810, 12/07/2000) to BWRVIP-26, "BWR Top Guide Inspection and Flaw Evaluation Guidelines," IASCC of reactor internals is considered a TLAA. Fluence calculations have been performed for the RV and internals. Four components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide; (2) Shroud; (3) Core Plate and (4) In-core Instrumentation Dry Tubes and Guide Tubes.
>
> The top guide, shroud, core plate and in-core instrumentation dry tubes and guide tubes are considered susceptible to IASCC. The aging effect associated with IASCC, crack initiation and growth, will require aging management. Three components, top guide, shroud and incore instrumentation dry tubes and guide tubes, have been evaluated by the BWRVIP, as described in the Inspection and Evaluation Guidelines for

each component: BWRVIP-26 (Top Guide), BWRVIP-76 (Shroud), and BWRVIP-47 (in-core instrumentation dry tubes and guide tubes). BFN implements the BWRVIP recommendations, as described in B.2.1.5 (Chemistry Control Program) and B.2.1.12 (BWR Vessel Internals Program). The core plate has been determined to be susceptible to IASCC and this is considered a plant-specific TLAA. BFN will manage this TLAA with two aging management programs: Chemistry Control Program (B.2.1.5) and BWR Vessel Internals Program (B.2.1.12). For the period of extended operation, the BWR Vessel Internals Program will perform inspections of the core plate in the regions of the highest fluence.

4.2.8.2 Staff Evaluation

The staff reviewed the information provided by the applicant in the LRA and determined that the austenitic stainless steel materials that are located in the following RV internal components are exposed to neutron fluence greater than 5×10^{20} n/cm^2 (E > 1 MeV) are considered susceptible to IASCC in the BWR environment: (1) Top Guide; (2) Shroud; (3) Core Plate; and, (4) In-core Instrumentation Dry Tubes and Guide Tubes. The applicant stated that the aging effects due to IASCC in the aforementioned components is managed by two aging management programs: (1) AMP B. 2.1.5, "Chemistry Control Program," and (2) AMP B.2.1.12, "Boiling Water Reactor Vessel Internals Program." AMP B.2.1.12 in turn addresses several BWRVIP inspection programs that are designed for various RV internal components. In addition, AMP B.2.1.12 invokes AMP B.2.1.4, "ASME B&PV Code Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program." The applicant claimed that implementation of these AMPs provides reasonable assurance that the aging effects due to IASCC will be managed so that the systems and components within the scope of this program will continue to perform their intended functions, consistent with the current licensing basis, for the period of extended operation. The applicant committed to implement the relevant BWRVIP programs to manage aging effects that are associated with each of the aforementioned components. The staff, in the following paragraphs, discusses the effectiveness of these AMPs in managing the aging effect due to IASCC in each of the aforementioned components.

Top Guide:

In addition to the implementation of AMPs B.2.1.5, and B.2.1.12, the applicant committed to invoke the inspection guidelines that are specified in the BWRVIP-26 "Boiling Water Reactor Top Guide Inspection and Flaw Evaluation Guidelines," which was approved by the staff. The implementation of these additional guidelines and the AMPs is consistent with the GALL Report AMP XI.M9. The staff finds that by implementing a proper chemistry program as dictated by AMP B.2.1.5, the oxidizing nature of the RCS water can be controlled and, thereby, the corrosion of the top guide can be controlled.

The staff, in a RAI-B.2.1.12(A), indicated that the BWRVIP-26 report lists 5×10^{20} n/cm^2 (E > 1.0 MeV) as the threshold fluence beyond which components may be susceptible to IASCC. According to the generic analysis in BWRVIP-26, the location on the top guide that will see a fluence equal to or greater than 5×10^{20} n/cm^2 (E > 1.0 MeV) is the grid beams. This is location 1, as identified in BWRVIP-26, Table 3-2, "Matrix of Inspection Options." In its evaluation of the top guide assembly in BWRVIP-26, General Electric (GE) assumed a lower allowable stress value, acknowledging the high fluence value at this location. The conclusion of this analysis, and the fact that a single failure at this location has no safety consequence, was that no inspection was considered necessary to manage IASCC in top guide grid beams.

The staff is concerned that multiple failures of the top guide grid beams are possible when the threshold fluence for IASCC is exceeded. According to BWRVIP-26, multiple cracks have been observed in top guide beams at Oyster Creek. In order to exclude the top guide grid beams from inspection when their fluence exceeds the threshold value, it must be demonstrated that failure of all beams that exceed the threshold fluence will not impact the safe shutdown of the reactor during normal, upset, emergency, and faulted conditions. If this can not be demonstrated, then an inspection program to manage this aging effect to preclude loss of component intended function is required.

In its response to RAI-B.2.1.12(A), by letter dated January 31, 2005, the applicant indicated that LRA section 4.7.6 considered the fluence at the top guide as a TLAA. The applicant manages this TLAA with the Chemistry Control Program and the BWR Vessel Internals Program (BWRVIP). The BWRVIP implements the requirements of NRC-accepted BWRVIP-26. NRC letter to Carl Terry, BWRVIP Chairman, dated June 10, 2003, states the following: "The staff believes that a comprehensive evaluation of the impact of IASCC and multiple failures of the top guide beams is necessary, and that an inspection program for top guide beams for all BWRs should be developed by the BWRVIP to ensure that all BWRs can meet the requirements of 10 CFR Part 54 throughout the period of extended operation." The applicant made a commitment to work as part of the BWRVIP to resolve these issues generically. When resolved, the applicant will follow the BWRVIP recommendations resulting from that resolution. Prior to the period of extended operation, the applicant will develop a site specific inspection program, if necessary to manage the effects of IASCC in the top guide.

The staff determines that the applicant must submit for NRC review and approval, a site-specific AMP that addresses the potential multiple failures of the top guide grid beams. **This is Open Item-4.2.8.2-1.** The staff finds that the implementation of the AMPs is consistent with the GALL Report AMP XI.M9, and Table IV. B1.2-a, IV.B1.2-b of NUREG-1801 with the above exception. Therefore, the staff determines that the applicant has demonstrated that the effects of aging due to IASCC in top guide will be adequately managed, except as noted above, so that its intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

Core Shroud:

In addition to the implementation of the AMPs B.2.1.4, B.2.1.5 and B.2.1.12, the applicant committed to implement the inspection guidelines of BWRVIP-76 "Boiling Water Reactor Core Shroud Inspection and Flaw Evaluation Guidelines." The staff's review of this report is not complete. The applicant proposed to evaluate the staff SER and complete SER action items. The staff requires that the applicant make a commitment to follow all the requirements and limitations that may be specified in the staff SE on the BWRVIP-76 report. The staff finds that by implementing a proper chemistry program as dictated by AMP B.2.1.5, the oxidizing nature of the RCS water can be controlled and, thereby, the corrosion of the core shroud can be controlled. In addition, implementation of the inservice inspection program mandated by AMP B.2.1.4, and additional inspection guidelines required by BWRVIP-76, will adequately identify any cracking in a timely manner, so that proper repair and other mitigation techniques can be implemented to restore the function of the core shroud. Since the implementation of these additional guidelines and the AMPs is consistent with the GALL Report AMP XI.M9, and Table IV.B1.1-a, through IV.B1.1-g of NUREG-1801, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

In LRA Section 4.2.4, the applicant stated that the maximum 54 EFPY fluence at the inside surface of the core shroud is 5.34×10^{21} n/cm^2. The staff, in RAI 4.2.4-1 (B), requested that the applicant address the aging effect due to IASCC in the core shroud component.

The applicant in its response to RAI 4.2.4-1 (B), stated that the BFN core shrouds are classified as “Category C” based on the core shroud classification criteria contained in Appendix B of BWRVIP-76 (currently under review by the staff), which is a part of AMP B.2.1.12. The BFN BWR Vessel Internals AMP requires inspection of core shroud welds in accordance with “Category C” core shroud inspection requirements contained in BWRVIP-76. The staff reviewed this response and accepts it (pending the approval of the BWRVIP-76 report) because implementation of AMPs B.2.1.12, and B.2.1.5 would adequately manage the aging effect due to IASCC in the core shroud components, and is consistent with GALL XI.M9 and XI.M2.

Core Plate:

The applicant proposed to implement AMPs B.2.1.4, B.2.1.5 and B.2.1.12. The AMP B.2.1.12 in turn invokes the inspection guidelines of the BWRVIP-25, “Boiling Water Reactor Core Plate Inspection and Flaw Evaluation Guidelines,” which was approved by the staff. The implementation of these additional guidelines and the AMPs is consistent with the GALL Report AMP XI.M9. The staff finds that by implementing a proper chemistry program as dictated by AMP B.2.1.5, the oxidizing nature of the RCS water can be controlled and, thereby, the corrosion of the core plate can be controlled. In addition, implementation of the inservice inspection program mandated by AMP B.2.1.4, and additional inspection guidelines required by BWRVIP-25, will adequately identify any cracking in a timely manner, so that proper repair and other mitigation techniques can be implemented to restore the function of the core plate. Since the implementation of these additional guidelines and the AMPs is consistent with the GALL Report AMP XI.M9, and Table IV.B1.1-a, through IV.B1.1-g of NUREG-1801, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

In-core Instrumentation Dry Tubes and Guide Tubes

In addition to the implementation of the AMPs B.2.1.4, B.2.1.5 and B.2.1.12, the applicant committed to invoke the inspection guidelines that are specified in BWRVIP--47 “Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines,” which was approved by the staff. The staff finds that by implementing a proper chemistry program as dictated by AMP B.2.1.5, the oxidizing nature of the RCS water can be controlled and, thereby, the corrosion of the in-core instrumentation dry tubes and guide tubes can be controlled. In addition, implementation of the inservice inspection program mandated by AMP B.2.1.4, and additional inspection guidelines required by BWRVIP-47, will adequately identify any cracking in a timely manner, so that proper repair and other mitigation techniques can be implemented to restore the function of the in-core instrumentation dry tubes and guide tubes . Since the implementation of these additional guidelines and the AMPs is consistent with the GALL Report AMP XI.M9, and Table IV. B1.6-a of NUREG-1801, the staff finds that the applicant has demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(a)(3).

4.2.8.3 UFSAR Supplement

Section A.3.5.5 of the LRA includes the following FSAR Supplement summary description for the TLAA on IASCC of the RV internals.

> Austenitic stainless steel RV internal components exposed to a neutron fluence greater than 5×10^{20} n/cm ^{2}(E > 1 MeV) are considered susceptible to irradiation assisted stress corrosion cracking (IASCC) in the BWR environment. Fluence calculations have been performed for the RV and internals. Four components have been identified as being susceptible to IASCC for the period of extended operation: (1) Top Guide; (2) Shroud; (3) Core Plate and (4) In-core Instrumentation Dry Tubes and Guide Tubes. Three components (top guide, shroud and in-core instrumentation dry tubes and guide tubes) have been evaluated by the BWRVIP, as described in the Inspection and Evaluation Guidelines for each component: BWRVIP-26 (Top Guide), BWRVIP-76 (Shroud), and BWRVIP-47 (incore instrumentation dry tubes and guide tubes). BFN implements the BWRVIP recommendations. The Chemistry Program and the BWR Vessel Internals Program will be used to manage the core plate.

The applicant's UFSAR supplement summary description for the TLAA on IASCC of the RV internals appropriately describes the implementation of relevant AMPs that would enable the applicant to effectively manage this aging effect. The staff however, requires that the applicant revise the UFSAR supplement to indicate that the inspection guidelines of the BWRVIP-25 "Boiling Water Reactor Core Plate Inspection and Flaw Evaluation Guidelines," will be implemented to effectively manage the aging effect on core plate. **This is Open Item- 4.2.8.3-1.** The staff determines that this applicant must revise the UFSAR supplement summary description to address the open items associated with core plate.

4.2.8.4 Conclusion

The staff has reviewed the applicant's TLAA on IASCC of the RV internals, as summarized in Section 4.7.6 of the LRA, and has determined that except for the top guide grid beams, the applicant appropriately describes that by implementing the AMPs B.2.1.4, B.2.1.5 and B.2.1.12, and relevant additional BWRVIP guidelines related to RV internal components, the aging effect due to IASCC will be adequately managed for the extended period of operation. The license renewal action items related to the implementation of the BWRVIP-25, BWRVIP-26 and BWRVIP-47 guidelines are discussed in Section 3.1.3.1.6.4 of the staff's SER on Aging Management Review. In addition, the staff believes that the implementation of these additional guidelines and the AMPs is consistent with the GALL Report AMP XI.M9, and Table IV. B1 of NUREG-1801. Therefore, the staff concludes that the applicant has demonstrated that the effects of aging due to IASCC in the RV internals with the exception of the top guide grid beams as stated above, will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).