SECTION 4

TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

This section discusses the identification of time-limited aging analysis (TLAAs). The applicant discusses the TLAAs in license renewal application (LRA) Sections 4.2 through 4.7. Safety evaluation report (SER) Sections 4.2 through 4.8 document the review of the TLAAs conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC or the staff).

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

In its letters dated June 9, 2005, and June 15, 2005, the applicant determined that LRA Sections 4.7.2, 4.7.3, and 4.7.5 should not be considered TLAAs; therefore, they were deleted from the application (See SER Sections 4.7.2, 4.7.3, 4.7.5).

In addition, pursuant to 10 CFR 54.21(c)(2), an applicant may 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.

4.1.1 Summary of Technical Information in the Application

To identify the TLAAs, the applicant evaluated calculations for the Browns Ferry Nuclear Plant (BFN) against the six criteria specified in 10 CFR 54.3. The applicant indicated that it had identified the calculations that met the six criteria by searching the current licensing basis (CLB). The CLB includes the updated final safety analysis report (UFSAR), engineering calculations, technical reports, engineering work requests, licensing correspondence, and applicable vendor reports. The applicant listed the following applicable TLAAs in LRA Table 4.1.1, “List of Time-Limited Aging Analyses”:

- neutron embrittlement of the reactor vessel and internals
- metal fatigue
- environmental qualification of electrical equipment
- loss of prestress in concrete containment tendons
- primary containment fatigue
- reactor building crane load cycles
- corrosion – flow reduction
- dose to seal rings for the high-pressure coolant injection and reactor core isolation cooling containment isolation check valves
- radiation degradation of drywell expansion gap foam

- corrosion – minimum wall thickness
- irradiation assisted stress corrosion cracking of reactor vessel internals
- stress relaxation of core plate hold-down bolts
- emergency equipment cooling water weld flaw evaluation

Pursuant to 10 CFR 54.21(c)(2), the applicant stated that it had not identified any exemptions granted under 10 CFR 50.12 that were based on a time-limited aging analysis (TLAA), as defined in 10 CFR 54.3.

4.1.2 Staff Evaluation

In LRA Section 4.1, the applicant identified the TLAAs applicable to BFN. The staff reviewed the information to determine if the applicant had provided adequate information to meet the requirements of 10 CFR 54.21(c)(1) and 10 CFR 54.21(c)(2).

As defined in 10 CFR 54.3, TLAAs are analyses that meet the following six criteria:

1. Involve systems, structures, and components that are within the scope of license renewal, as delineated in 10 CFR 54.4(a);
2. Consider the effects of aging;
3. Involve time-limited assumptions defined by the current operating term (40 years);
4. Are determined to be relevant by the applicant in making a safety determination;
5. Involve conclusions, or provide the basis for conclusions, related to the capability of the system, structure, and component to perform its intended functions, as delineated in 10 CFR 54.4(b); and
6. Are contained or incorporated by reference in the CLB.

The applicant provided a list of common TLAAs from U.S., Nuclear Regulatory Commission Regulatory Guide (NUREG)-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” (SRP-LR) dated July 2001. The applicant listed those TLAAs that are applicable to BFN, in LRA Table 4.1.1, “List of Time-Limited Aging Analyses.”

As required by 10 CFR 54.21(c)(2), an applicant must provide a list of all the exemptions granted under 10 CFR 50.12 that are based on a TLAA and evaluated and justified for continuation through the period of extended operation. In its LRA, the applicant stated that each active exemption was reviewed to determine whether the exemption was based on a TLAA. The applicant did not identify any TLAA-based exemptions. On the basis of the information provided by the applicant with regard to the process used to identify TLAA-based exemptions, as well as the results of the applicant’s search, the staff concluded that the applicant identified no TLAA-based exemptions that are justified for continuation through the period of extended operation, in accordance with 10 CFR 54.21(c)(2).

4.1.3 Conclusion

On the basis of its review, the staff concluded that the applicant provided an acceptable list of TLAAs, as required by 10 CFR 54.21(c)(1). The staff also confirmed that no exemptions to 10 CFR 50.12 have been granted on the basis of a TLAA, as required by 10 CFR 54.21(c)(2).

4.2 Neutron Embrittlement of Reactor Vessel and Internals

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 Vessel and Internals Project (BWRVIP)-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.

The 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 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.

Appendix G of 10 CFR Part 50 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 the American Society of Mechanical Engineers (ASME) Code Section XI, Appendix G. The Rule also mandates that the methods used to calculate USE values account for the effects of neutron irradiation on the USE values for the materials and 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 a given RV beltline material is represented in the plant's reactor vessel material surveillance program. If surveillance data are not available, the Charpy USE is determined in accordance with position 1.2 in RG 1.99, Revision 2. If surveillance data are 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 10 CFR Part 50 Appendix G are based on a flaw with a depth equal to one-quarter of the vessel wall thickness (1/4*t*), the neutron fluence used in the Charpy USE analysis is the neutron fluence at the 1/4*t* 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 boiling water reactor (BWR) RV 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 Technical Information in the Application

In LRA Section 4.2.1, the applicant provided USE values for the limiting beltline materials. USE is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. Appendix G of 10 CFR Part 50 requires the predicted end of life (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 RVs 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 (GE) NEDO-32205-A, "10 CFR 50 Appendix G Equivalent Margin Analysis for Low Upper-Shelf Energy in BWR-2 through BWR-6 Vessels," which has been 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 40 years with an assumed neutron exposure of less than 10^{19} n/cm^2 (E > 1.0 MeV). The CLB 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 Part 50, Appendix H and ASTM E23, "Methods for Notched Bar Impact Testing of Metallic Materials," invoked by 10 CFR Part 50 Appendix G. Therefore, alternative methods approved by the staff in NEDO-32205-A were used to demonstrate compliance with the 10 CFR Part 50, Appendix G USE requirement.

Fluences were calculated for the RVs 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 an SER dated September 14, 2001. The applicant used bounding fluence calculation, for each unit which included an extended power uprate[2] (EPU). The applicant provided the results for one bounding calculation for each RV and determined the peak surface fluence of 1.95 x 10^{18} n/cm^2 and peak 1/4*t* fluence of 1.35 x 10^{18} n/cm^2 for Unit 1 vessel, and peak surface fluence of 2.3 x 10^{18} n/cm^2 and peak 1/4*t* fluence of 1.59 x 10^{18} n/cm^2 for Units 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 1/4*t* location into the vessel wall measured radially from the inside diameter using Equation 3 from Paragraph 1.1 of RG 1.99, Revision 2. This 1/4*t* 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. 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 that percentage of reductions in USE of equal to or less than 23.5 percent and 39 percent, respectively, would meet the requirements of 10 CFR Part 50, Appendix G. The applicant provided results of the EMA for limiting welds and plates on the three RVs, which are summarized in LRA Tables 4.2.1.1 through 4.2.1.6. The applicant stated that the results are acceptable because the limiting USE percentage drop is less than the BWRVIP-74-A percentage drop acceptance criterion in all cases.

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[2]TVA by letter dated January 7, 2005, agreed to decouple the power uprate licensing request from License Renewal Application. The safety review of this item will be further evaluated as part of the EPU review.

4.2.1.2 Staff Evaluation

Appendix G to 10 CFR Part 50, Section IV.A.1 requires, in part, that the reactor pressure vessel (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 ASME Code Section XI, Appendix G.

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 ASME Code Section XI, Appendix G in accordance with 10 CFR Part 50, Appendix G. 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.

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 dated October 18, 2001, from Mr. C.I. Grimes to Mr. C. Terry. 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 percentage 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 percentage reduction in USE value is 23.5 percent for the plates, and 39 percent for welds except for Linde 80 weld. Linde 80 welds are discussed later in this SER.

The staff’s review of LRA Section 4.2.1 identified an area in which additional information was necessary to complete the review of the reactor vessel materials USE reduction due to neutron embrittlement evaluation. The applicant responded to the staff’s request for additional information (RAI) as discussed below.

In RAI 4.2.1-1, dated December 1, 2004, the staff requested that the applicant provide the initial USE values, percentage reduction in USE values, percentage of copper, and 1/4*t* fluence at the end of the period of extended operation (including power uprate conditions) for all the plates and non-Linde 80 weld metals in the beltline region of the RVs. 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 to demonstrate that the beltline plates and non-Linde 80 weld metals of the RVs meet the criteria in the BWRVIP-74 report at the end of the license renewal period. In its response, by letter January 31, 2005, the applicant stated that the initial USE values are not available for BFN; however, BFN has used the EMA method to demonstrate that the BFN vessels will maintain adequate fracture toughness throughout the period of extended operation. The LRA bounding value for EFPY is 54 EFPY for Unit 1 and 52 EFPY for Units 2 and 3. The values for all beltline materials for BFN are listed in Tables 4.2.1-1 through 4.2.1-3 of the applicant's response. The staff has verified the copper contents given in Tables 4.2.1-1 to 4.2.1-3 and concluded that applicant's response for all the beltline materials with the corresponding data in Reactor Vessel Integrity Data Base (RVID) is acceptable.

The applicant stated that the percentage reduction in the USE value for the limiting beltline plate base materials and non-Linde 80 beltline welds for all the units is less than the aforementioned acceptance criteria specified in BWRVIP-74. The staff also verified the reduction in the unirradiated USE values due to neutron radiation for the beltline base metals and non-Linde 80 beltline welds for all the units using the methodology in RG 1.99, Revision 2, and found that all the beltline materials meet the acceptance criteria specified in the staff-approved report BWRVIP-74, and 10 CFR Part 50, Appendix G. Therefore, the staff's concern described in RAI 4.2.1-1 is resolved.

The BWRVIP-74 establishes criteria for Linde 80 welds and other types of welds and base metals in the BWR RPVs. The criteria for Linde 80 welds require that the fracture toughness of the Linde 80 weld shall be established by using J-R curve based on copper and neutron fluence values. By letter dated November 21, 2005, the applicant revised LRA Table 4.2.1.1 to indicate that the limiting beltline circumferential weld for the BFN Unit 1 was made with Linde 80 flux. The applicant in its letter dated November 21, 2005, also provided the fracture toughness data (J-R curve based on the limiting copper and the neutron fluence at the end of the period of extended operation, which includes power uprate) and the $J_{applied}$ values for the Linde 80 weld, and concluded that the subject weld will maintain adequate fracture toughness during the extended period of operation. The staff verified the applicant's data and concluded that the BFN Unit 1 limiting circumferential Linde 80 weld would meet the acceptance criteria specified in the staff-approved BWRVIP-74 report and 10 CFR Part 50, Appendix G for the period of extended operation.

4.2.1.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of RV materials USE reduction due to neutron embrittlement in LRA Section A.3.1.1. On the basis of its review and the RAI response above, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on RV materials USE reduction due to neutron embrittlement and is, therefore, acceptable.

4.2.1.4 Conclusion

The staff reviewed the applicant's RAI response and TLAA on USE, as summarized in LRA Section 4.2.1, and determined that the RV beltline materials at BFN will continue to comply with the staff's USE requirements of 10 CFR Part 50, Appendix G throughout the periods of extended operation for the BFN units. The staff therefore concluded that the applicant's TLAA for USE is in compliance with the requirements of 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 concluded that the UFSAR 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 of Technical Information in the Application

In LRA Section 4.2.2, the applicant summarized the ART determination for the RV materials due to neutron embrittlement. The 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 margin term is defined in RG 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 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 CLB 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 RG 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 an SER dated September 14, 2001. One bounding calculation was performed for each 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 1/4*t* location into the vessel wall measured radially from the inside diameter using equation 3 from Paragraph 1.1 of RG 1.99, Revision 2. This 1/4*t* 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 RG 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 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 x 10^{18} n/cm^2 and peak 1/4*t* fluence of 1.35 x 10^{18} n/cm^2 for, the Unit 1 vessel, and peak surface fluence of 2.3 x 10^{18} n/cm^2 and peak 1/4*t* fluence of 1.59 x 10^{18} n/cm^2 for, the Units 2 and 3 vessels. LRA Table 4.2.2.1 shows bounding fluence values for BFN for 54, 52 and 52 EFPYs of the operation, respectively.

The staff's review of LRA Section 4.2.2 identified areas in which additional information was necessary to complete the review of the ART values for RPV materials due to neutron embrittlement evaluation. The applicant responded to the staff's RAI as discussed below.

In RAIs 4.2.2(A), and 4.2.2(B), dated December 1, 2004, the staff requested that the applicant provide an explanation addressing the following issues:

a. The staff requested that the applicant explain why 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 x 10^{18} n/cm^2 (E > 1.0 MeV) for Unit 1, while Units 2 and 3 achieve 2.3 x 10^{18} n/cm^2 (E > 1.0 MeV) at the end of 60 years.

   After reviewing the applicant's response, dated January 31, 2005, the staff determined that the applicant performed fluence calculations for Unit 1 assuming 54 EFPY of operation and for 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 Unit 1 is lower than the fluence values for Units 2 and 3 is that the maximum delta RT_{NDT} and ART occurs in the circumferential weld material for Unit 1, which is located away from the peak vessel fluence location, whereas for both 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 Unit 1 has an applied axial correction factor of 0.81 and 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[1] evaluation. For consistency with the EPU evaluation, the 54 EFPY value was incorporated into the LRA. The ART values are listed in Tables 4.2.2-1 through 4.2.2-6 of the applicant's response.

   The staff reviewed the applicant's response and found the explanation for using the fluence values cited for Units 1, 2, and 3 acceptable because it accounts for differences in weld location and neutron flux for each unit. The staff found that this approach is acceptable as it identifies the maximum ART values for all three units. Therefore, the staff's concern described in RAI 4.2.2 (A) is resolved.

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[1]TVA by letter dated January 7, 2005, agreed to decouple the power uprate licensing request from License Renewal Application. The safety review of this item will be further evaluated as part of the EPU review.

b. The staff requested that the applicant provide the initial RT_{NDT} and ART values at 1/4t and vessel ID surface at the end of the period of the extended operation for all the materials in the beltline region of the BFN RVs.

The applicant provided information on the above items in Tables 4.2.2-1 to 4.2.2-6 of its response dated January 31, 2005. The staff verified the percentages of copper and nickel and the initial RT_{NDT} given in the applicant's response for all the beltline materials with the corresponding data in RVID and found 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 found them acceptable. Therefore, the staff's concern described in RAI 4.2.2 (B) is resolved.

4.2.2.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of ART for RV materials due to neutron embrittlement in LRA Section A.3.1.2. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on ART for RV materials due to neutron embrittlement and is, therefore, acceptable.

4.2.2.4 Conclusion

The staff reviewed the applicant's TLAA on the calculation of ART values, as summarized in LRA Section 4.2.2 and the RAI response dated January 31, 2005, and 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 concluded that the applicant's TLAA for calculation of the ART values meet the requirements of 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 concluded that the UFSAR 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 of Technical Information in the Application

The applicant stated that UFSAR Section 3.3.5 includes an EOL thermal shock analysis performed on the RVs for a design basis loss of coolant accident (LOCA) followed by a LPCI system initiation. The effects of embrittlement assumed in 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 LPCI system initiation and accounted for the full effects of neutron embrittlement at the end of the current license term

of 40 years. The current analysis assumes EOL material toughness, which in turn depends on EOL ART values. The critical location for fracture mechanics analysis is at one quarter of the vessel thickness (from the inside, 1/4*t*). For this event, the peak stress intensity occurs approximately 300 seconds after the LOCA. The applicant stated that the analysis shows that 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (which is 1/4*t*) is approximately 400 °F. The ART values, described in LRA Section 4.2.2 and tabulated in Table 4.2.2.1, list the ART values for the limiting weld metal of the RVs. The highest calculated RV beltline material ART value is 167.7 °F (Unit 1). Using the equation for K_{IC} presented in ASME Section XI Appendix A and the maximum ART value, the material reaches upper shelf (a K_{IC} value of 200 ksi $\sqrt{in}$) at 272 °F, which is well below the 400 °F, 1/4*t* 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 EOL material toughness, which in turn depends on EOL ART. The critical location for fracture mechanics analysis is at the 1/4*t* 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 1/4*t* is approximately 400 °F, which is much higher than the 54 EFPY ART value167.7 °F for the limiting material of all the three BFN vessels. Therefore, the staff concurred 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 167.7 °F for Unit 1, which is below the 400 °F at 1/4*t* temperature predicted for 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 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the “programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation.” The applicant provided UFSAR supplement summary description of reflood thermal shock analysis of the RV in LRA Section A.3.1.3. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on reflood thermal shock analysis of the RV and is, therefore, acceptable.

4.2.3.4 Conclusion

The staff reviewed the applicant’s TLAA on reflood thermal shock analysis of the RV for a design basis LOCA and concluded that the applicant has 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 determined that this revised analysis for the period of extended operation meets the requirements of 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 of Technical Information in the Application

LRA Section 4.2.4 states that the radiation embrittlement may affect the ability of RV internals, particularly the core shroud, to withstand a LPCI 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 RV core shrouds, the applicant considered the location on the inside surface of the core shroud opposite the midpoint of the fuel centerline as the location most susceptible to damage during a 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 LRA 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 a LPCI transient in this region will be 155,700 psi equivalent to 0.57 percent strain. This strain range of 0.57 percent 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 percentage 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 percent for a temperature of 297 °C (567 °F) with a neutron flux of 8×10^{21} n/cm^2 (E > 1 MeV), while the average value for base metal at 290 °C (554 °F) is 20 percent. The applicant concluded that the measured value of elongation bounds the calculated thermal shock strain amplitude of 0.57 percent and that the calculated thermal shock strain at the most irradiated location is acceptable considering the embrittlement effects for the period of extended operation.

4.2.4.2 Staff Evaluation

In the thermal shock analysis of RV core shrouds, the applicant considered the location on the inside surface of the core shroud opposite the midpoint of the fuel centerline as a location most susceptible to damage during a 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 has been approved by the staff.

The staff’s review of LRA Section 4.2.4 identified areas in which additional information was necessary to complete the review of the reflood thermal shock analysis of the reactor vessel core shroud evaluation. The applicant responded to the staff’s RAI as discussed below.

In RAI 4.2.4-1(A), dated December 1, 2004, the staff stated that in LRA Section 4.2.4, “Reflood Thermal Shock Analysis of the RV Core Shroud and Repair Hardware,” the applicant stated 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). Therefore, the staff requested that the applicant provide an explanation of whether this value is bounding at the inside shroud surface for all

three units. If so, submit information whether the neutron fluence values are estimated based on the implementation of EPU[1].

In its response, by letter January 31, 2005, the applicant 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 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. After the review, the staff concurred with the applicant, and accepted 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) and the applicant's response are addressed in SER Section 4.7.6.2 under core shroud subsection.

In RAI 4.2.4-1(C), dated December 1, 2004, the staff stated that the applicant calculated thermal strain resulting from the LPCI reflood thermal shock transient in the core shroud region. The applicant compared the calculated thermal strain with the measured values of percentage of 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 percentage reduction in area as a criterion to evaluate the thermal strain. Therefore, the staff requested that the applicant provide information on the measured percentage 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 percentage of 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.

In its response, by letter January 31, 2005, 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 (°F)	Reduction in Area (%)
1×10^{21}	550	40
6.9×10^{21}	750	52.5

[1]TVA by letter dated January 7, 2005, agreed to decouple the power uprate licensing request from License Renewal Application. The safety review of this item will be further evaluated as part of the EPU review.

Elongation

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

The applicant stated that the bounding shroud fluence (Unit 1) is 5.34 x 10^{21} n/cm^2 (E >1 MeV) for BFN, 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 percent. 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 percentage 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 RV core shroud. This staff position was previously approved for Dresden and Quad Cities LRA SER (NUREG-1796). The staff concluded that the thermal shock strain associated with the 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 concluded that the core shroud will have sufficient ductility during the reflood thermal shock transient during the period of extended operation. The staff accepts the applicant's analysis for the BFN units.

4.2.4.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of reflood thermal shock analysis of the RV core shroud in LRA Section A. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on reflood thermal shock analysis of the RV core shroud and is, therefore, acceptable.

4.2.4.4 Conclusion

The staff reviewed the applicant's TLAA on reflood thermal shock analysis of the RV core shroud and the applicant's responses to the RAIs and concluded that the applicant has demonstrated that the calculated thermal shock strain at the most irradiated portion of the core shroud is acceptable. The staff also accepted the applicant's conservative methodology in establishing the integrity of the most irradiated location of the core shroud during a low-pressure coolant injection thermal shock event. The staff determined that the revised analysis for the period of extended operation meets the requirements of 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 of Technical Information in the Application

In LRA Section 4.2.5, the applicant addressed the RV thermal limit analysis. The ART value is the sum of initial RT_{NDT} + delta RT_{NDT} + margins for uncertainties at a specific location. Neutron embrittlement increases the ART value. Thus, the minimum metal temperature at which an RV is allowed to be pressurized increases. The ART value of the limiting beltline material is used to correct the beltline P-T limits to account for irradiation effects. Appendix G of 10 CFR Part 50 requires RV thermal limit analyses to determine operating P-T limits for three categories of operation: (1) hydrostatic pressure tests and leak tests, referred to as Curve A; (2) non-nuclear heatup/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 heatup, 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 staff for approval before the start of the period of extended operation using an approved fluence methodology. By letter dated December 6, 2004, the applicant submitted updated P-T curves for Unit 1, which are currently being reviewed by the staff. The applicant stated that the P-T curves for Units 2 and 3 were approved by the staff as documented in safety evaluations dated March 10, 2004. The applicant's CLB allows the development of P-T limit curves consistent with the 2000 Edition, 2001 Addenda of Section XI of the ASME 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 found 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 10 CFR Part 50, Appendix G.

4.2.5.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of RV thermal limit analyses: operating temperature and pressure limits in LRA Section A.3.1.5. On the basis of its review, the staff concluded that

the UFSAR supplement summary adequately describes the TLAA on reactor vessel thermal limit analyses: operating P-T limits and is, therefore, acceptable.

4.2.5.4 Conclusion

The staff reviewed the applicant's TLAA on P-T limits, as summarized in LRA Section 4.2.5 and determined that the applicant will generate the P-T limits for the periods of extended operation for BFN. The staff therefore concluded that the applicant's TLAA for the BFN P-T limits will meet the requirements of 10 CFR 54.21(c)(1)(ii) 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 concluded that the UFSAR 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 of Technical Information in the Application

LRA Sections 4.2.6 and A.3.1.6 discuss inspection of the 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 GL 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 Units 2 and 3 have received this relief for the remainder of their current 40-year licensed operating periods. Unit 1 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 staff to extend this relief request.

The staff 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 (DBEs) was considered. LRA Table 4.2.6.1 provides a comparison of Units 2 and 3 RV limiting circumferential weld parameters to those used in the staff evaluation of BWRVIP-05 for the first two key assumptions. Data provided in LRA Table 4.2.6.1 were supplied from Tables 2.6.4 and 2.6.5 of the final safety evaluation of the BWRVIP-05 report.

For Units 2 and 3, the fluence is equivalent to that used in the staff analysis. However, 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 Units 2 and 3 are lower than the 64 EFPY shift from the staff 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 Units 2 and 3 that are considerably lower than the staff mean analysis values. Based on this analysis, the applicant concluded that the RV conditional failure probability is bounded by the staff 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 staff when the applicant requested the relief for the current license term for 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 (BWRVIP Chairman). In this letter, the staff concluded that since the failure frequency for RV 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 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 RV 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 RV 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 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 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 beltline materials meet the criteria specified in the report. To demonstrate that the 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 verified the data for the copper and nickel contents and the initial RT_{NDT} values for Units 2 and 3 beltline materials by comparing them with the corresponding data in the RVID maintained by the staff. The 52 EFPY mean RT_{NDT} value for Units 2 and 3 is 25 °F. The staff checked the applicant's calculations for the 52 EFPY mean RT_{NDT} values for the 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 Units 2 and 3 are less than the 64 EFPY mean RT_{NDT} value of 129.4 °F used by the staff 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 Babcock and Wilcox (B&W) weld, because B&W welded the circumferential welds in the 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 concluded that the BFN RV conditional failure probabilities are bounded by the staff 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 staff when the applicant requested 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 applicant's 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 staff 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 staff letter to the applicant, "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 LRA Section A.3.1.6, reference the safety evaluation request letters identified above. The staff found the response acceptable because the applicant identified the requested references and commits to include them in LRA Sections 4.2.6 and A.3.1.6.

By letter dated May 12, 2004, the applicant submitted a relief request concerning the examination of the Unit 1 RV circumferential welds for the current license period.

In RAI 4.2.6-1, dated December 1, 2004, the staff requested that the applicant provide the RV circumferential weld examination relief analyses for Unit 1. In its response, by letter January 31, 2005, the applicant submitted the following relief analyses related to the 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.

Group	B & W 64 EFPY	BFN Unit 1 54 EFPY
Cu %	0.31	0.27
Ni %	0.59	0.6
CF	196.7	184
Fluence at clad/weld interface 10^{19} n/cm^2	0.19	0.2
Delta RT_{NDT} without margin (°F)	109.4	104
Initial RT_{NDT} (°F)	20	20
Mean RT_{NDT} (°F)	129.4	124
P (F/E) NRC	4.83×10^{-4}	----------
P (F/E) BWRVIP	----------	----------

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.

The staff verified the accuracy of the of the mean RT_{NDT} for the limiting beltline circumferential weld at Unit 1 and found it acceptable. In the staff's evaluation of the BWRVIP-05 report, a fluence of 0.19×10^{19} n/cm^2 for B&W RVs was used for 64 EFPY and the corresponding delta RT_{NDT} value is 109.4 °F. The delta RT_{NDT} value for the limiting beltline weld metal of 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 applicant'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 4.83×10^{-4} per reactor-year for a limiting plant-specific mean RT_{NDT} of 129.4 °F for B&W fabricated RVs. The low temperature over-pressure (LTOP) transient 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 Unit 1 is projected to be 124 °F at the end of the period of extended operation (54 EFPY). In this evaluation, the chemistry factor, delta RT_{NDT}, and mean RT_{NDT} were calculated consistent with the guidelines of RG 1.99, Revision 2. Since the calculated value of mean RT_{NDT} for the circumferential welds at Unit 1 is lower than that for the limiting plant-specific case for B&W fabricated RVs, the vessel failure frequencies of the Unit 1 circumferential welds is less than 4.83×10^{-7} per reactor-year.

The staff found 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 staff 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 the period of extended operation in accordance with 10 CFR 50.55a.

4.2.6.3 UFSAR Supplement

The applicant's UFSAR 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 staff 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 Units 2 and 3. Since the UFSAR supplement summary description adequately describes the TLAA for Units 2 and 3, the staff concluded that the UFSAR supplement summary description for the TLAA on RV circumferential weld examination relief for Units 2 and 3 is acceptable. In addition, in a letter dated May 25, 2005, the applicant stated that the UFSAR supplement summary description also includes Unit 1 as shown in the revised supplement A.3.1.6.

4.2.6.4 Conclusion

The staff reviewed the applicant's TLAA on RV circumferential weld examination relief, as summarized in LRA Section 4.2.6, and determined that the applicant appropriately explained that the conditional failure probabilities for the RV circumferential welds are bounded by the staff analysis in the SER on the BWRVIP-05 report, dated July 28, 1998, and that the applicant will be using procedures and training to limit cold over-pressure events during the period of extended operation for BFN. However, the staff concluded that the LRA Section A.3.1.6 should include circumferential weld examination analysis for Unit 1. The staff, therefore, concluded that the applicant's LRA Section 4.2.6 on TLAA, and LRA Section A.3.1.6 for the BFN RV circumferential weld examination relief will meet the requirements of 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 of Technical Information in the Application

LRA Section 4.2.7 discusses the BWRVIP recommendations for inspection of RV shell welds and contains generic analyses supporting a staff SER conclusion that the axial weld failure rate is no more than 5 x 10^{-6} per reactor year. The applicant stated that the supporting evaluations described in the LRA only apply to 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 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 Units 2 and 3 are all electroslag welds with similar chemistry. The 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 staff in the BWRVIP-05 SER supplement. The applicant concluded that the probability of failure for the axial welds is bounded by the staff 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, in letters dated December 15, 1998, and November 12, 1999, the BWRVIP supplied evaluations of axial weld failure frequency. 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 Pilgrim Nuclear Station SER as a model for BWR RVs that were fabricated with electroslag welds, and demonstrated that a mean RT_{NDT} of 114°F resulted in a failure frequency of 5 x 10^{-6} per reactor-year of operation. The applicant calculated, and the staff confirmed, that the limiting axial weld mean RT_{NDT} value for Units 2 and 3 at 52 EFPY is 108°F, which supports the conclusion that the failure frequencies for Units 2 and 3 will be less than 5 x 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, dated December 1, 2004, the staff requested that the applicant provide an evaluation for the RV axial weld failure probability analyses for Unit 1 for the current license period, and the period of extended 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 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
$RT_{NDT(U)}$ (°F)	-2	23
Mean RT_{NDT} (°F)	114	109
P (F/E) NRC	5.02×10^{-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 RG 1.99, Revision 2. The applicant calculated, and the staff confirmed, that the limiting axial weld mean RT_{NDT} value for Unit 1 at 54 EFPY is 109 °F. This value is lower than that for the limiting mean RT_{NDT} value of 114 °F in the staff's evaluation of BWRVIP-05. Therefore, the staff concluded that the failure frequencies for Unit 1 axial welds will be less than 5×10^{-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

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of RV axial weld failure probability in LRA Section A.3.1.7. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on RV axial weld failure probability and is, therefore, acceptable. In addition, in a letter dated May 25, 2005, the applicant stated that the UFSAR supplement summary description also includes Unit 1 as shown in the revised supplement A.3.1.7.

4.2.7.4 Conclusion

The staff reviewed the applicant's TLAA on the evaluation of RV axial weld failure probabilities, as summarized in LRA Section 4.2.7, and determined that the applicant appropriately describes that the analyses 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, the UFSAR supplement summary description in LRA Section A.3.1.7 should include the analysis on the conditional failure probabilities for the Unit 1 RV axial welds. The staff therefore concluded that the applicant's LRA Sections 4.2.7, and A.3.1.7 related to the analysis of the conditional failure probabilities for the BFN units RV axial welds are acceptable. The staff concluded that the analysis of the RV axial weld failure probability for the BFN units will meet the requirements of 10 CFR 54.21(c)(1)(ii), except as noted above.

4.3 Metal Fatigue

A metal component subjected to cyclic loading at loads less than the static design load may fail due to fatigue. Metal fatigue of components may have been evaluated based on an assumed number of transients or cycles for the current operating term. The validity of such metal fatigue analysis is reviewed for the period of extended operation. The GALL Report identifies fatigue aging related effects that require evaluation as possible TLAAs, pursuant to 10 CFR 54.21(c). Each of these is summarized in the SRP-LR and presented in LRA Section 4.

4.3.1 Reactor Vessel Fatigue Analysis

4.3.1.1 Summary of Technical Information in the Application

In LRA Section 4.3.1, "Reactor Vessel Fatigue Analyses," the applicant stated that the original pressure vessel stress report included ASME Code Section III fatigue analyses of the RV components based on a set of design basis transients and corresponding cycles, which are listed in UFSAR Section 4.2.5. The analyzed components consisted of the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, and closure studs. The original 40-year analysis demonstrated that the cumulative usage factors (CUFs) for these components are below the ASME Code Section III limiting value of 1.0. A re-analysis was performed for BFN to determine the CUFs of these components under EPU and Maximum extended load line limit analysis conditions, for 60 years of operation. LRA Table 4.3.1.1 lists the results of this re-analysis for seven bounding reactor vessel components. These components are the recirculation outlet nozzle, recirculation inlet nozzle, feedwater nozzle, core spray nozzle, the support skirt, the closure stud bolts, and the vessel shell. This table shows that for Units 2 and 3, the recirculation outlet nozzles, the feedwater nozzles, the support skirts and the closure stud bolts, all have 60-year projected CUFs that exceed the ASME Code Section III Class 1 limiting value of 1.0. These results also bound the projected CUFs for Unit 1.

The applicant stated that fatigue aging of the seven components listed in LRA Table 4.3.1-1 will be managed by the Fatigue Monitoring Program (LRA Section B.3.2) for the period of extended operation.

The applicant also stated that the original ASME Code analysis of the reactor vessel also included fatigue analyses of the feedwater nozzles and the control rod drive (CRD) hydraulic system return line nozzles. After several years of operation, these nozzles were found to be susceptible to cracking caused by a number of factors, including rapid thermal cycling. The CRD hydraulic system return line nozzles were therefore capped and removed from service. As such, they are no longer susceptible to rapid thermal cycling. A re-analysis was performed on the feedwater nozzles and modifications were implemented to reduce or eliminate the effects of the high thermal cycling, based on generic BWROG guidance.

Based on its evaluation, the applicant concluded that, for some components, the fatigue analyses of the reactor vessel will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i), or that for the remaining vessel components, the effects of aging will be adequately managed for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

4.3.1.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.3.2, pertaining to the fatigue analysis of reactor vessel components. The CLB fatigue analyses of components associated with the reactor vessels were identified as TLAAs, in accordance with the provisions of 10 CFR 54.3(a) and the components listed in the appropriate tables in the GALL Report. The applicant listed the bounding CUFs associated with these TLAAs and indicated that the CUFs for four vessel components would exceed the ASME Code Section III Class 1 limiting value of 1.0 during the period of extended operation. The applicant, therefore, committed to monitor the fatigue of these vessel components as part of the Fatigue Monitoring Program, which provides for monitoring fatigue stress cycles to ensure that the CUF limit of 1.0 is not exceeded. The staff found this acceptable and concurred with the applicant that the effects of aging of the reactor vessel components for BFN will be adequately managed with the Fatigue Monitoring Program for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii) The staff also found acceptable that, for those components where the CUF did not exceed 1.0, the fatigue analyses were projected to remain valid to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

In accordance with 10 CFR 54.21(d), the applicant's supplement for the UFSAR regarding the reactor vessel fatigue TLAAs is provided in LRA Section A.3.2.1. The staff reviewed this supplement and found it acceptable because it provides a reasonable summary of the information presented in LRA Section 4.3.1.

4.3.1.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided a UFSAR supplement summary description of reactor vessel fatigue analyses in LRA Section A.3.2.1. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the reactor vessel fatigue TLAAs and is, therefore, acceptable.

4.3.1.4 Conclusion

The staff reviewed the applicant's TLAA on the reactor vessel fatigue analyses, as summarized in LRA Section 4.3.1, and determined that the metal fatigue assessments at Units 1, 2, and 3 will continue to comply with the staff's requirements throughout the period of extended operation. The staff, therefore, concluded that the applicant's TLAA for reactor vessel fatigue analyses meets the requirements of 10 CFR 54.21(c)(1)(ii), (iii), and that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1). The staff also concluded that the UFSAR supplement contains an appropriate summary description of the TLAA on reactor vessel fatigue analyses for the period of extended operation, as required by 10 CFR 54.21(d).

4.3.2 Fatigue Analysis of Reactor Vessel Internals

4.3.2.1 Summary of Technical Information in the Application

In LRA Section 4.3.2, "Fatigue Analysis of Reactor Vessel Internals," the applicant stated that the original fatigue evaluation of the reactor vessel internals was performed using ASME Code Section III as a guide. The evaluation determined that the most significant fatigue loading occurs at the jet pump diffuser-to-baffle-plate weld location. The fatigue analysis of this location was the only fatigue analysis actually performed. Since this analysis was based on a number of cycles for a 40-year life, it is considered a TLAA. The calculated CUF was 0.35, less than the ASME Code Section III Class 1 allowable CUF of 1.0. Since the original fatigue analysis was based on a 40-year design life, the calculation for the jet pump diffuser-to-baffle-plate weld was projected for a 60-year life by multiplying the CUF by 1.5, which resulted in a CUF less than the ASME Code allowable of 1.0.

The applicant also stated that at Unit 3, a lower section of the core spray line was replaced, and a repair was installed to address cracking found at the location of the core spray-to-T-box weld. Fatigue calculations were performed for several components of the core spray line using ASME Code Section III as a guide, since the core spray line is not classified as an ASME Code Section III component. However, these analyses are considered as TLAAs since they were based on a 40-year life. A fatigue evaluation of the lower core spray line sectional replacement was performed, resulting in a maximum calculated CUF of 0.45, based on a 40-year design life. An explicit fatigue calculation was also performed for the T-box repair, based on a 40-year design life. The CUF was calculated to be 0.022. The fatigue calculation for the core spray-to-T-box weld repair was evaluated for a lifetime of 60 years by multiplying the 40-year CUF by 1.5, which resulted in a 60-year CUF that is less than the ASME Section III Class 1 limit of 1.0. The fatigue analysis is, therefore, acceptable for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The applicant also concluded that these results are applicable for BFN.

The applicant stated that the core spray-to-T-box weld location is also included for inspection as part of the Boiling Water Reactor Vessel Internals Program (LRA Section B.2.1.12). These inspections will be used to manage the effects of potential cracking of these welds.

For the lower core spray sectional replacement, the design life was specified as 40 years. However, since this modification was installed more than 20 years into the current licensing

period, the applicant concluded that these fatigue calculations will remain valid for the period of extended operation.

Based on the revised fatigue analyses, the applicant concluded that, in accordance with 10 CFR 54.21(c)(1)(i), the fatigue analyses for the reactor internals remain valid for the period of extended operation or, in accordance with10 CFR 54.21(c)(1)(ii), the fatigue analyses have been projected to the end of the period of extended operation. The applicant also stated that, in accordance with 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function(s) of the reactor vessel internals for the BFN units will be adequately managed for the period of extended operation.

4.3.2.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.3.2 pertaining to the fatigue analysis of reactor vessel internals. Based on the reported CUFs corresponding to the reported fatigue analyses, the staff concurred with the applicant that the fatigue analyses for the reactor vessel internals remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i), or that the fatigue analyses have been projected to the end of the period of extended operation, in accordance with10 CFR 54.21(c)(1)(ii). The staff also found acceptable that the effects of aging on the intended function(s) of the reactor internals for BFN will be adequately managed with the Fatigue Monitoring Program for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

In accordance with 10 CFR 54.21(d), the applicant's supplement for the UFSAR regarding the fatigue analyses of reactor vessel internals is provided in LRA Section A.3.2.2. The staff reviewed this supplement and found it acceptable. It provides a reasonable summary of the information presented in LRA Section 4.3.2.

4.3.2.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided a UFSAR supplement summary description of the fatigue analysis of reactor vessel internals in LRA Section A.3.2.2. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the reactor vessel internals fatigue TLAAs and is, therefore, acceptable.

4.3.2.4 Conclusion

The staff reviewed the applicant's reactor vessel internals fatigue TLAAs, as summarized in LRA Section 4.3.2, and determined that the metal fatigue assessments at BFN Units 1, 2, and 3 will continue to comply with the staff's requirements throughout the period of extended operation. The staff, therefore, concluded that the applicant's evaluation of reactor vessel internals fatigue TLAAs meets the requirements of 10 CFR 54.21(c)(1)(i) - (iii), and that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1). The staff also concluded that the UFSAR supplement contains an appropriate summary description of the

TLAA on fatigue analysis of reactor vessel internals for the period of extended operation, as required by 10 CFR 54.21(d).

4.3.3 Piping and Component Fatigue Analysis

4.3.3.1 Summary of Technical Information in the Application

In LRA Section 4.3.3, “Piping and Components Fatigue Analysis,” the applicant stated that the reactor coolant pressure boundary (RCPB) and non-RCPB piping was designed to USA Standard (USAS) B31.1. This code does not require an explicit fatigue analysis. However, the RCPB and non-RCPB piping within the scope of license renewal that is designed to USAS B31.1 requires the application of a stress reduction factor to the allowable thermal stress range if the number of full range cycles exceeds 7000.

The applicant indicated that the assumed thermal cycle count for the analyses can be approximated by the thermal cycles used in the reactor vessel fatigue analysis. These thermal cycles are listed in UFSAR Section 4.2.5. The total count of all these listed thermal cycles is fewer than 1100 over the 40-year plant life. For the 60-year extended operating period, the number of assumed operating cycles would be increased to 1650, considerably fewer than the 7000 cycle threshold in USAS B31.1. In accordance with 10 CFR 54.21(c)(1)(i), the applicant concluded that the existing piping analyses within the scope of licence renewal will remain valid for the period of extended operation.

4.3.3.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.3.3, pertaining to the fatigue analysis of piping and components. The applicant indicated that the RCPB and non-RCPB piping and components at BFN, within the scope of license renewal, were designed to USAS B31.1-1967. Although this Code does not require explicit fatigue analysis, it considers fatigue implicitly in the design calculations by applying a stress range reduction factor to the allowable thermal stress range, which depends on the number of design thermal expansion cycles. The staff, therefore, concurred with the applicant that qualifications of piping to this code are considered TLAAs, in accordance with the provisions of 10 CFR 54.21(c)(1).

In the application of USAS B31.1-1967, the applicant approximated the number of thermal expansion cycles over a 40-year plant life by the thermal cycles used in the reactor vessel fatigue analysis. These thermal cycles are listed in UFSAR Section 4.2.5. For a 60-year plant life, the total count of all significant full thermal cycles was determined as fewer than 1650, which is substantially less than the 7000-cycle full thermal stress range limit in USAS B31.1. The staff concurred with the applicant that an adequate margin of safety for the RCPB and non-RCPB systems will be maintained for the period of extended operation, because the projected number of thermal operating cycles to the end of the period of extended operation is fewer than the design cycle limit of 7000 cycles, and the stress range limits in the current piping calculations therefore remain valid. The staff, therefore, concurred with the applicant that the existing piping analyses, within the scope of license renewal, will remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

In accordance with 10 CFR 54.21(d), the applicant's supplement for the UFSAR regarding the piping and component fatigue analyses is provided in LRA Section A.3.2.3. The staff reviewed this supplement and found it acceptable because it provides a reasonable summary of the information presented in LRA Section 4.3.3.

4.3.3.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of piping and component fatigue analysis in LRA Section A.3.2.3. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the piping and component fatigue TLAA and is, therefore, acceptable.

4.3.3.4 Conclusion

The staff reviewed the applicant's piping and component fatigue TLAA, as summarized in LRA Section 4.3.3, and determined that the metal fatigue assessments at BFN Units 1, 2, and 3 will continue to comply with the staff's requirements throughout the period of extended operation. The staff therefore concluded that the applicant's piping and component fatigue TLAA meets the requirements of 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 period of extended operation, as required by 10 CFR 54.21(c)(1). The staff also concluded that the UFSAR supplement contains an appropriate summary description of the piping and component fatigue TLAA for the period of extended operation, as required by 10 CFR 54.21(d).

4.3.4 Effects of Reactor Coolant Environment On Fatigue Life of Components and Piping (Generic Safety Issue 190)

4.3.4.1 Summary of Technical Information in the Application

In LRA Section 4.3.4, "Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)," the applicant described the actions taken to address the issue of environmentally assisted fatigue. Generic Safety Issue (GSI) 190 addresses the effects of reactor coolant environment on the fatigue life of components and piping. Although GSI 190 is resolved, SRP-LR Section 4.3.1.2 states that for licence renewal, the applicant's consideration of the effects of coolant environment on component fatigue life is an area of review.

The applicant stated that plant-specific calculations were performed for the following fatigue sensitive component locations, identified in NUREG/CR 6260 for older-vintage BWRs:

- reactor vessel shell and lower head
- reactor vessel feedwater nozzle
- reactor recirculation piping (outlet and inlet nozzles)
- core spray system (nozzle and safe end)
- residual heat removal (RHR) line Class 1 piping

- feedwater line Class 1 piping

The applicant stated that for each location listed above, detailed environmental fatigue calculations for 60 years were performed using the appropriate environmental fatigue life correction factor (F_{en}) relationships from NUREG/CR 6583 “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” for carbon and alloy steels, and the appropriate F_{en} relationships from NUREG/CR 5704 “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels,” for stainless steel, as appropriate for the material. These evaluations are consistent with the recommendations in SRP-LR Section 4.3.2.2 for addressing the effects of the reactor coolant environment by assessing the effects on a sample of critical components. The 60-year CUF for the reactor recirculation piping was determined as 4.181, and the 60-year CUF for the feedwater line Class 1 piping was calculated as 1.489. In accordance with 10 CFR 54.21(c)(1)(iii), the applicant stated that all necessary plant transients will be tracked using the Fatigue Monitoring Program, to ensure that CUF values will remain below 1.0 for the period of extended operation. For the locations where the CUF is expected to exceed 1.0 for the 60-year period, the applicant stated that additional fatigue analyses will be performed prior to the period of extended operation, and appropriate action will be taken if the EOL CUF values above 1.0 are projected.

4.3.4.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.3.4 pertaining to the effects of reactor coolant environment on the fatigue analysis of components and piping.

GSI-166, “Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address license renewal. The NRC closed GSI-190 in December, 1999, concluding that:

> The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (Nuclear Energy Institute (NEI) and EPRI), and the different approaches available to the licensees to manage the effects of aging, lead to the conclusion that no generic regulatory action is required, and that GSI-190 is closed.This conclusion is based primarily on the negligible calculated increases in core damage frequency in going from 40 to 60 year lives. However, the calculations supporting resolution of this issue, which included consideration of environmental effects, and the nature of age-related degradation indicate the potential for an increase in the frequency of pipe leaks as plants continue to operate. Thus, the staff concluded that, consistent with existing requirements in 10 CFR 54.21, licensees should address the effects of coolant environment on component fatigue life as aging management programs are formulated in support of license renewal.

The applicant evaluated the component locations listed in NUREG/CR-6260 that are applicable to an older-vintage BWR plant for effect of the environment on the fatigue life of the components. For each location, detailed environmental fatigue calculations were performed using the appropriate F_{en} relationships from NUREG/CR 6583, “Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels,” for carbon and alloy

steels, and those from NUREG/CR 5704, “Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels,” for stainless steel, as appropriate for the material. These calculations showed that two locations were projected to exceed the CUF limiting value of 1.0 prior to the end of the period of extended operation. In accordance with 10 CFR 54.21(c)(1)(iii), the applicant committed to track all necessary plant transients, using the BFN Fatigue Monitoring Program, to ensure that the CUF values will remain below 1.0 for the period of extended operation. For those locations where the CUF is expected to exceed 1.0 for the 60-year period, the applicant stated that additional analyses will be performed prior to the period of extended operation, and appropriate action will be taken if the end-of-life CUF values are projected to be above 1.0.

The staff found the environmental fatigue effects assessment acceptable, since this evaluation is consistent with the recommendations in SRP-LR Section 4.3.2.2 for addressing the effects of the reactor coolant environment by assessing the effects on a sample of critical components. The staff also found acceptable the applicant’s commitment to use the Fatigue Monitoring Program to assure that the CUFs at the critical locations will not exceed the limiting CUF value of 1.0 during the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(iii).

In accordance with 10 CFR 54.21(d), the applicant included a section addressing the effects of reactor coolant environment on fatigue life of components and piping (Issue 190) in LRA Section A.3.2.4. The applicant committed to include the locations that have projected CUF values greater than 1.0 in the Fatigue Monitoring Program. The staff found this supplement acceptable because it provides a reasonable summary of the information presented in LRA Section 4.3.4.

4.3.4.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the “programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation.” The applicant provided a UFSAR supplement summary description of GSI 190 in LRA Section A.3.2.4. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on GSI 190 and is, therefore, acceptable.

4.3.4.4 Conclusion

The staff reviewed the applicant’s TLAA on GSI 190, as summarized in LRA Section 4.3.4, and determined that the metal fatigue assessments at BFN Units 1, 2, and 3 will continue to comply with the staff’s requirements throughout the period of extended operation. The staff therefore concluded that the applicant’s TLAA for GSI 190 meets the requirements of 10 CFR 54.21(c)(1)(iiI), and that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21(c)(1). The staff also concluded that the UFSAR supplement contains an appropriate summary description of the TLAA on GSI 190 for the period of extended operation, as required by 10 CFR 54.21(d).

4.4 Environmental Qualification

The 10 CFR 50.49 Environmental Qualification (EQ) Program has been identified as a TLAA for the purposes of license renewal. The TLAA of EQ electrical components includes all long-lived, passive and active electrical and instrumentation and controls (I&C) components that are important to safety and located in a harsh environment. The harsh environments of the plant are those areas that are subjected to the environmental effects of a LOCA or a high-energy line break (HELB). The EQ equipment comprises SR and Q-list equipment; nonsafety-related (NSR) equipment, the failure of which could prevent satisfactory accomplishment of any SR function; and necessary post-accident monitoring equipment.

As required by 10 CFR54.21(c)(1), the applicant must provide a list of EQ TLAAs in the LRA. The applicant shall demonstrate that one of the following is true for each type of EQ equipment: (1) the analyses remain valid for the period of extended operation; (2) the analyses have been projected to the end of the period of extended operation; or (3) the effect of aging on the intended function(s) will be adequately managed for the period of extended operation.

4.4.1 Summary of Technical Information in the Application

The EQ Program for Units 2 and 3 was established to verify that all plant equipment within the scope of 10 CFR 50.49 is qualified for its application and meets its specified performance requirements when subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life. The EQ Program for Unit 1 will be established to ensure compliance with 10 CFR 50.49. The EQ Program complies with the requirements of 10 CFR 50.49(e)(5) for aging considerations that affect functionality and make provisions to replace the components or establish ongoing qualification when the demonstrated qualified life has expired. The EQ-related equipment is identified in a controlled equipment data base with a qualification binder that is maintained with records on performance specifications, electrical characteristics, and environmental conditions.

The EQ Program manages thermal, radiation and cyclic aging as applicable for all electrical components within the scope of 10 CFR 50.49. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during and after a DBE after experiencing the effects of in-service aging.

The applicant chose Option (iii) of 10 CFR 54.21(c)(1) in its TLAA evaluation to demonstrate that aging effects of the EQ equipment identified in this TLAA will be managed during the period of extended operation by the EQ Program activities. Maintaining qualification through the extended license renewal period requires that existing EQ evaluations be reanalyzed. A summary of the applicant's application of these 10 CFR 50.49(f) methodologies to the EQ evaluations for the period of extended operations follows:

> Analytical Methods - The analytical models used in the re-analysis of an aging evaluation are the same as those applied during the initial qualification. The Arrhenius methodology is an acceptable thermal model for performing an aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (i.e., normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of

establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (i.e., 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. Cyclical aging will be reevaluated for those components subject to this effect.

Data Collection and Reduction Methods - Reducing excess conservatism in the service conditions used in the aging evaluation is one method that can be used in a re-analysis. Evaluations based on actual plant temperature data will, in certain cases, yield desired results for extended service life. Should the applicant opt to use this approach, plant temperature data can be obtained in several ways, including plant monitors, measurements taken by plant personnel, and temperature sensors on various plant equipment. Similar methods of reducing excess conservatism in the component service conditions may be also be used for radiation and cyclical aging.

Underlying Assumptions - Environmental excursions identified during plant operation or maintenance activities that could affect the qualification of an EQ component will be evaluated. Should unexpected adverse conditions be identified, the affected EQ component is evaluated and appropriate corrective actions taken, which may include changes to the qualification basis and conclusions reached, or restructuring of the affected component's EQ requirements.

Acceptance Criteria and Corrective Actions - If the qualification cannot be extended by re-analysis using the above methodologies, the component will be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid.

The applicant stated that the 10 CFR 50.49 EQ Program is consistent with the guidance provided for resolution in the NRC Regulatory Issue Summary 2003-09, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables." The regulatory issue summary states:

> For license renewal, a re-analysis (based on the Arrhenius methodology) to extend the life of the cables by using the available margin based on a knowledge of the actual operating environment compared to the qualification environment, coupled with observations of the condition of the cables during walk-downs, was found to be an acceptable approach. Monitoring I&C cable condition could provide the basis for extending cable life.

The EQ Program allows re-analysis for maintaining qualification using the methods described above. In addition, the EQ Program has the following procedural requirements in place to monitor and track aging effects.

- Detecting degradation of materials or equipment performance by requiring preventive maintenance and periodic surveillance.
- Failure trend evaluations related to equipment and environments.
- Notification of environmental excursions and subsequent evaluation of components.
- Review of licensing, industry, and other generic industry operating experience.

4.4.2 Staff Evaluation

A site-wide EQ Program required by 10 CFR 50.49 has been developed for BFN, and implemented on Units 2 and 3, and it is expected to be implemented on Unit 1 to ensure compliance with 10 CFR 50.49. This item is discussed in SER Section 2.6.1.4.

The staff's review of LRA Section 4.4 identified areas in which additional information was necessary to complete the review of the EQ evaluation. The applicant responded to the staff's RAI as discussed below.

In RAI 4.4-2, dated November 4, 2004, the staff stated that the provisions of 10 CFR 50, Appendix A, General Design Criteria (GDC) 4 require that all equipment (electrical and mechanical) related to safety be designed to accommodate the environmental effects of postulated accidents. Similarly, Standard Review Plan (SRP) 3.11 (NUREG-0800) applies equally to mechanical and electrical equipment. Therefore, the staff requested the applicant to provide a discussion of the materials for mechanical equipment in the LRA that are required to be evaluated as an EQ TLAA that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, diaphragms, and wear cycle aging from lubricant deterioration) and the aging analyses that will be, or have been, conducted to satisfy the requirements of 10 CFR 54.21(c)(1) for the period of extended operation.

In its response, by letter December 20, 2004, the applicant stated that BFN was licensed before the establishment of NRC GDC-4, "Environmental and Dynamic Effects of Design Basis," and NUREG-0800. Consequently, neither GDC-4 nor SRP 3.11 are part of BFN's CLB. Therefore, the applicant does not have a formal mechanical equipment qualification program. As part of the application review process, the applicant performed searches of Industry Guidance (SRP-LR and NEI 95-10), the UFSAR, the Operating Licenses and License Conditions, Technical Specifications, Technical Requirements Manuals, and Licensing Basis Program Documents such as In-Service Inspection and EQ for possible TLAA's. For the type of mechanical equipment described above, the only TLAA found was "Dose to Seal Rings for the High Pressure Coolant Injection and Reactor Core Isolation Cooling Containment Isolation Check Valves," SER Section 4.7.3. On the basis of its review, the staff found that the applicant had adequately addressed the concern and the issue is resolved.

In RAI 4.4-1, dated November 4, 2004, the staff requested the applicant to provide a list of components covered under EQ TLAA. In its response, by letter December, 9, 2004, the applicant provided the list of components covered under the EQ TLAA. On the basis of its review, the staff found that the applicant had adequately addressed the concern and the issue is closed.

The staff reviewed the information in LRA Section 4.4 to determine whether the applicant demonstrated that the effects of aging on the intended function(s) of electrical components will be adequately managed through its existing EQ Program, together with other plant programs/processes, during the period of extended operation as required by 10 CFR 54.21(c)(1)(iii).

The applicant's program activities establish, demonstrate, and document the level of qualification, qualified configuration, maintenance, surveillance, and replacement requirements

necessary to meet 10 CFR 50.49. Qualified life is determined for equipment within the scope of the EQ Program and appropriate actions, replacement or refurbishment are taken prior to or at the end of qualified life of the equipment so that aging limits or acceptable margins are not exceeded.

On the basis of its review, the staff concluded that the applicant had addressed the issues associated with GSI-168. The applicant will continue to manage the effects of aging through the EQ Program for the period of extended operation. The staff found that the applicant had satisfactorily addressed GSI-168 for license renewal, as required by 10 CFR 54.21(c)(1)(iii). The staff issued Regulatory Issue Summary (RIS) 2003-09 on May 2, 2003, to inform addressees of the results of the technical assessment of GSI-168. This RIS requires no action on the part of the addressees. Therefore, the staff considers GSI-168 issue to be resolved.

4.4.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of the TLAA for the period of extended operation." The applicant provided a UFSAR supplement summary description of EQ in LRA A.3.3. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on EQ and is, therefore, acceptable.

4.4.4 Conclusion

On the basis of its review, the staff concluded that the applicant demonstrated that the effect of aging on the intended function(s) of electrical and I&C components will be adequately managed for the period of extended operation by the existing EQ Program as required by 10 CFR 54.21(c)(1)(iii).

4.5 Loss of Prestress in Concrete Containment Tendons

The BFN containments do not have prestressed tendons. As such, this topic is not a TLAA applicable for BFN.

4.6 Primary Containment Fatigue

Cyclic loads acting on the primary containment and the attached piping and components include reactor building interior temperature variation during the heatup and cooldown of the RCS, a postulated LOCA, annual outdoor temperature variations, thermal loads on containment penetrations due to high-energy piping lines (such as steam and feedwater lines), seismic loads, and pressurization due to periodic Type A integrated leak-rate tests.

Metal containment penetration sleeves (including dissimilar metal welds) and penetration bellows may be designed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code. If a plant's code of record requires a fatigue analysis, then this analysis may be a TLAA and must be evaluated in accordance with 10 CFR 54.21(c)(1) to ensure that the effects of aging on the intended functions of the containment sleeves and bellows will be adequately managed for the period of extended operation.

In LRA Section 4.6, the applicant referenced UFSAR Section C.5.1, which states that the primary containment vessels for Units 1 and 2 were designed in accordance with the ASME Code Section III 1965 Edition with Addenda up through Winter 1966. The primary containment vessel for Unit 3 was designed in accordance with the ASME Code Section III1965 Edition with Addenda up through Summer 1967. Subsequently, while performing large-scale testing for the Mark III containment system and in-plant testing for the Mark I containment system, new hydrodynamic loads were identified for the suppression chamber (also referred to as the torus), that were not included in the original structural analyses. These additional loads result from blowdown into the suppression chamber during a postulated LOCA, and from main steam relief valve operation during plant transients. The results of structural analyses for BFN under these effects were reported in the BFN Torus Integrity Long-Term Program Plant Unique Analysis Report (PUAR). This program is described in UFSAR Section C.5.3. The applicant indicated that modifications of the suppression chamber and the suppression chamber vents, including the vent headers and downcomers, were required in order to re-establish the original design safety margins. The safety margins for these components were determined based on the allowable stresses stated in Subsection NE of the 1977 ASME Boiler and Pressure Vessel Code, Section III, including Summer 1977 Addenda.

As part of the review of the Torus Integrity Long-term Program PUAR, the applicant identified the following fatigue analyses as TLAAs:

- fatigue of the torus, vents, and downcomers
- fatigue of torus-attached piping and safety relief valve discharge lines
- fatigue of vent line and process penetration bellows

In analyzing and determining the disposition of these TLAAs for the period of extended operation, the applicant applied the following criteria:

1. The applicant stated that locations with a 40-year CUF of 0.666 are not considered as having adequate analytical or event margin when linearly extrapolated to 60 years. A CUF limit of 0.4 was chosen as providing this margin. Disposition option 10 CFR 54.21(c)(1)(i) was therefore applied to locations with a calculated 40-year CUF less than 0.4.
2. For locations where the 40-year CUF is greater than 0.4, the applicant stated that fatigue will be managed by the Fatigue Monitoring Program described in LRA Section B.3.2. Disposition option 10 CFR 54.21(c)(1)(iii) will, therefore, be applied to these locations.

4.6.1 Fatigue of Suppression Chamber, Vents, and Downcomers

4.6.1.1 Summary of Technical Information in the Application

The applicant stated that the BFN Torus Integrity PUAR includes fatigue analyses of the torus and torus vents, including the vent headers and downcomers. These analyses assumed a limited number of main steam safety relief valve (SRV) actuations and are, therefore, TLAAs.

Based on recorded plant data extrapolated to 40 years, the BFN Torus Integrity PUAR assumed 500 SRV actuations during 40 years of normal operations and the contribution from

the postulated worst-case LOCA. The worst-location and the corresponding fatigue CUFs were determined as follows:

- 0.681, at the intersection of the vent headers with the downcomers
- 0.373, at the downcomer/tiebar intersection
- 0.37, for the torus restraint snubbers

Since only the SRV loads contribute to fatigue during normal operation, normal operation may continue so long as the CUF contribution from SRV actuations has not exceeded 1.0 minus the CUF contribution expected from the postulated worst-case LOCA phenomena.

The applicant indicated that, based on operating experience, the total number of SRV actuations is not expected to exceed 500 actuations for any unit during the period of extended operation. This expectation is based on an estimate of the total number of SRV actuations expected for each unit until the end of the period of extended operation. The applicant described the methodology used for estimating the total number of SRV actuations. It was based on estimating the number of SRV actuations from the start up of each unit through August 2003, an estimate of the number of valve actuations expected for the remainder of the current licensing term and for the requested period of extended operation.

The applicant stated that, based on this methodology, the number of SRV actuations from the startup of each unit through August 2003 was estimated to be 146 actuations for Unit 1, 254 actuations for Unit 2 (worst case), and 188 actuations for Unit 3. (These estimates included both planned and unplanned SRV actuations.) The estimated total number of SRV actuations from August 2003 until the end of the period of extended operation was projected to be 239 for Unit 2. Thus, the estimated total number of SRV actuations at the end of the period of extended operation for Unit 2 is 493. This is the worst-case estimate of the total number of SRV actuations expected at the end of the period of extended operation. Thus, the assumed number of 500 SRV actuations for the three units is considered to be conservative.

To ensure that corrective actions are taken before any CUF approaches 1.0, the applicant indicated that, in accordance with 10 CFR 54.3(c)(1)(iii), the applicant will manage the high CUF locations for the period of extended operation by monitoring the SRV actuations using the Fatigue Monitoring Program.

4.6.1.2 Staff Evaluation

The staff reviewed the LRA regarding the fatigue TLAAs of the torus, vents and downcomers. The staff also reviewed the applicant's disposition of these TLAAs and found it acceptable because it specified the threshold limit of CUF equals 0.4 for 40 years of operation as a criterion for determining if the fatigue analyses performed under the PUAR will remain valid for the period of extended operation. The staff concurred with the applicant that this criterion will provide additional analytical or event margin over the minimum CUF value of 0.666 for the period of extended operation. Those locations, by not exceeding the threshold criterion, will therefore remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). In accordance with 10 CFR 54.21(c)(1)(iii), for locations where the CUF exceeds the criterion above, the staff found the applicant's commitment to manage the effects of fatigue for the period of extended operation with the Fatigue Monitoring Program acceptable

because it will provide assurance that the monitored CUF at a location will not exceed the ASME Code Section III CUF limiting value of 1.0; or, if the CUF is projected to exceed this limit, the applicant committed to take appropriate corrective action to assure that this limit will not be exceeded, as stated in LRA Section 4.6, in accordance with the Fatigue Monitoring Program. As described in LRA Section B.3.2, the Fatigue Monitoring Program will include an enhancement to monitor the fatigue of the torus and torus vents, and the vent headers and downcomers, using an EPRI-licensed cycle counting and fatigue usage tracking computer program. The applicant also committed to implement this enhancement prior to the period of extended operation.

In accordance with 10 CFR 54.21(d), the applicant's supplement for the UFSARs regarding the suppression chamber, vents, and downcomers fatigue TLAAs is provided in LRA Section A.3.4 "Containment Fatigue." The staff reviewed this supplement and found it acceptable because it provides a reasonable summary of the information presented in LRA Section 4.6.1.

4.6.1.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided a UFSAR supplement summary description of metal fatigue analyses of suppression chamber, vents, and downcomers in LRA Section A.3.4.

4.6.1.4 Conclusion

On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the metal fatigue TLAAs of suppression chamber, vents, and downcomers and is, therefore, acceptable.

4.6.2 Fatigue of Torus Attached Pipe and Safety Relief Valve Discharge Lines

4.6.2.1 Summary of Technical Information in the Application

In LRA Section 4.6.2, the applicant stated that there are thirteen Target Rock dual-mode MSRVs to allow blowdown from the main steam piping in the drywell to the suppression pool via individual discharge lines passing through the main vents. These lines enter the suppression chamber through penetrations in the suppression chamber vent header and the steam is discharged to the suppression pool water through T-quenchers attached to the ends of the lines. There are, in addition, a number of other external piping systems attached to the suppression chamber shell.

The torus integrity PUAR indicates that an evaluation of the fatigue effects of Mark I containment cyclic "new loads" on main steam relief valve discharge lines internal to the suppression chamber and on torus-attached piping external to the suppression chamber was performed using a program developed by the Mark I Owners Group.

The fatigue analyses assumed 500 SRV actuations for a 40-year plant lifetime, and included the effects of both mechanical and thermal expansion load cycling. These analyses are,

therefore TLAAs. The analyses concluded that the worst location on the main steam safety relief valve (MSRV) discharge lines would have a fatigue CUF of less than 0.35 at the end of 40 years of operation. The analyses also concluded that the worst location on the torus attached piping would have a fatigue CUF of less than 0.103 at the end of 40 years of operation. The applicant concluded that, for the MSRV discharge lines and T-quenchers, the MSRV discharge line penetrations, the torus attached piping systems, and the associated penetration locations, the predicted 60-year CUF will, therefore be less than 0.666 (worst-case CUF is 0.35 x 60/40 = 0.53). The applicant thus concluded that the MSRV discharge lines and the torus-attached piping fatigue analyses will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.6.2.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.6.2 regarding the fatigue TLAAs of the torus attached piping and the SRV discharge lines. The staff reviewed the applicant's disposition of these TLAAs and found it acceptable because the applicant selected a threshold limit of CUF equals 0.4 for 40 years of operation as a criterion for determining whether the fatigue analyses performed under the PUARs will remain valid for the period of extended operation. Based on this criterion, the staff concurred with the applicant's disposition of these TLAAs, since it demonstrated that the highest 40-year CUFs will not exceed the threshold limit of 0.40. These locations will therefore remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

In accordance with 10 CFR 54.21(d), the applicant's supplement for the UFSARs regarding the fatigue TLAAs of the torus attached piping and the SRV discharge lines is provided as part of LRA Section A.3.4. The staff reviewed this supplement and found it acceptable because it provides a reasonable summary of the information presented in LRA Section 4.6.2.

4.6.2.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation." The applicant provided a UFSAR supplement summary description of fatigue of torus attached pipe and SRV discharge lines in LRA Section A.3.4.

4.6.2.4 Conclusion

On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the metal fatigue TLAA of torus attached pipe and SRV discharge lines and is, therefore, acceptable.

4.6.3 Fatigue of Vent Line and Process Penetration Bellows

4.6.3.1 Summary of Technical Information in the Application

The applicant stated in LRA Section 4.6.3 that the torus vent line bellows are flexible expansion joints allowing movement of the main vent pipes through the torus wall without developing

significant interaction loads, and maintaining the required pressure boundary. The analysis of the suppression chamber bellows is described in the PUAR and was performed in accordance with Standards of the Expansion Joint Manufacturers Association, Inc. The design life of the bellows is stated in UFSAR Section C.5.2 as 7000 thermal cycles over the 40-year life for the plant and the fatigue analyses are, therefore, TLAAs.

Containment pipe penetrations that must accommodate pipe thermal movement also have expansion bellows. Containment process piping expansion joints between the drywell shell penetrations and process piping are the only ones subject to significant thermal expansion and contraction. The design life of these bellows is also stated as 7000 operating thermal cycles over the design life at containment normal, test, and limiting design pressures throughout the 40-year life for the plant and are, therefore, TLAAs.

For the suppression chamber vent line bellows and the containment penetration bellows, thermal cycles are imposed by the thermal expansion cycles experienced by the attached piping. The assumed thermal cycle count for the analyses used in the codes associated with the piping and components can be conservatively approximated by the full thermal cycles (not including power reductions) used in the reactor vessel fatigue analysis listed in UFSAR Section 4.2.5. The applicant stated that the total count of all full thermal cycles (not including power reductions) is less than 1100 for a 40-year plant life. For the 60-year plant life, the number of thermal cycles for piping analyses would be proportionally increased to less than 1650, which is less than 25 percent of the 7000-cycle design life.

Since the suppression chamber bellows and the containment penetration bellows metal fatigue analyses have a large design fatigue life margin, the applicant concluded that the analyses will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.6.3.2 Staff Evaluation

The staff reviewed the technical information in LRA Section 4.6.3 regarding the metal fatigue TLAAs of the vent line bellows and the containment process piping penetration bellows. The staff concurred with the applicant's disposition of this TLAA and found it acceptable because it demonstrated, in accordance with 10 CFR 54.21(c)(1)(i), that the number of full thermal cycles expected by the end of the period of extended operation will not exceed the 7000-cycle design-life of these bellows.

In accordance with 10 CFR 54.21(d), the applicant's supplement for the UFSAR regarding the metal fatigue TLAAs of the vent line and process penetration bellows is provided in LRA Section A.3.4. The staff reviewed this supplement and found it acceptable. It provides a reasonable summary of the information presented in LRA Section 4.6.3.

4.6.3.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided a UFSAR supplement summary description of metal fatigue analyses of vent line and process penetration bellows in LRA Section A.3.4.

4.6.3.4 Conclusion

On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the metal fatigue TLAAs of vent line and process penetration bellows and is, therefore, acceptable.

4.7 Other Plant-Specific Analyses

In LRA Section 4.7, the applicant provided its evaluation of plant-specific TLAAs. The TLAAs evaluated include the following:

- reactor building crane load cycles
- corrosion – flow reduction
- dose to seal rings for the high pressure coolant injection and reactor core isolation cooling containment isolation check valves
- radiation degradation of drywell expansion gap foam
- corrosion – minimum wall thickness
- irradiation assisted stress corrosion cracking of reactor vessel internals
- stress relaxation of core plate hold-down bolts
- emergency equipment cooling water weld flaw evaluation

4.7.1 Reactor Building Crane Load Cycles

4.7.1.1 Summary of Technical Information in the Application

The applicant stated in Section 4.7.1 that the 125-ton reactor building overhead crane serves three reactor units and includes a 5-ton auxiliary load hoist. The crane is designed to meet the design loading requirements of the Crane Manufacturers Association of America (CMAA) Specification 70. For cyclic loading, CMAA 70 specifies that a crane classified as Service Class A1 is limited to 100,000 loading cycles over the design life. The applicant's analysis identifies that the total number of expected cycles for this crane over the entire life including construction, the 60-years of operation for all three units, and the decommissioning, has been conservatively estimated at less than 21,00 loading cycles. Of these, less than 1000 lifts are expected to be more than 90 percent of the rated capacity. The applicant concluded that the analysis of the 125-ton reactor building crane qualifies the passive structural components for extended life in accordance with CMAA 70 Service Class A1 requirements.

4.7.1.2 Staff Evaluation

During its review of the applicant's analysis the staff determined that additional information was needed to complete its review. The staff identified that TVA letter dated September 28, 1982, in response to NUREG-0612, stated that the structural and rotating parts of the crane were designed for infinite life. In RAI 4.7.1-1, the applicant was requested to clarify if infinite life is still valid or to explain the derivation of the total number of loading cycles estimated. In this RAI, the applicant was also requested to explain the difference between the 21,000 cycles estimated in

LRA Section 4.7.1 and the 7,500 cycles estimated in LRA Section B.2.1.20. Further, the applicant was requested to clarify if additional loading cycles caused by vibration during crane operation are considered in the analysis or are the basis for not including loading cycles induced by vibration. By letter dated January 12, 2005, the applicant explained that its letter dated September 28, 1982, is based on an endurance limit of 40 percent of the tensile strength which, although reasonable, is not in accordance with CMMA 70; therefore, the results of the evaluation for license renewal supercede the September 28, 1982, results provided to the NRC. The applicant also clarified that the 7,500 lifts are full-load equivalent cycles, and that the estimated load lifts are less than 1,000 near-rated lifts, less than 10,000 moderate-load lifts, and less than 10,000 light-load lifts. In regard to vibration, the applicant's response clarified that a review of operating experience indicates that vibration in the structural components has not been noticed or reported for the reactor building crane. The applicant identified that non-structural vibration and wear issues have been reported. For example, motor generator vibration has been reported, measured, and promptly corrected. The staff determined that the applicant's response satisfactorily answers the staff's technical concerns, and all items related to RAI 4.7.1-1 are resolved.

Based on its review of the applicant's analysis included in the LRA and additional clarifications provided by the applicant in response to RAI 4.7.1-1, the staff concurred with the applicant that the reactor building crane has been evaluated and is qualified for the period of extended operation. The crane is qualified for a 100,000-cycle design life, which exceeds the estimated load cycles for the life of the crane including life extension. Hence counting actual load cycle is is not required for the reactor building crane because estimated load cycles are well below the limits for the crane established by CMAA 70. Therefore, fatigue life is not significant to the operation of this equipment, and the analysis is valid for the period of extended operation. The applicant provided a satisfactory validation of 10 CFR 54.21(c)(1)(i). The staff also reviewed the UFSAR Supplement A.3.5.1 and determined that the UFSAR Supplement includes an appropriate summary description of the reactor building crane load cycles TLAA evaluation for the period of extended operation, as required by 10 CFR 54.21(d).

4.7.1.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided UFSAR supplement summary description of reactor building crane load cycles in LRA Section A.3.5.1. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on reactor building crane load cycles and is, therefore, acceptable.

4.7.1.4 Conclusion

On the basis of its review, the staff concluded that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i) that the analyses remain valid for the reactor building crane load cycles TLAA. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the reactor building crane load cycles TLAA evaluation for the period of extended operation, as required by 10 CFR 54.21(d). Therefore, the staff has reasonable assurance that the safety margins established and maintained during the current

operating term will be maintained during the period of extended operation, as required by 10 CFR 54.21 (c)(1).

4.7.2 Corrosion – Flow Reduction

LRA Section 4.7.2 originally considered a design calculation that addresses concerns whether the flow reduction due to corrosion in carbon steel piping used in raw water systems is a TLAA. In a letter dated June 15, 2005, the applicant provided additional information. The functional basis for determining the acceptability is based on periodic flow testing as described in the Technical Instruction 0-TI-171 RHRSW Sump Pump Flow Test, Surveillance Instruction 0-SI-4.5.C.1(4) EECW System Annual Flow Rate Test, Surveillance Instruction 1/2/3-SI-4.5.C.1(3) RHRSW Pump and Header Operability and Flow Test, and Surveillance Instructions 0-SI-4.11.B.1.g for Fire Protection Piping. Based on its further review, the applicant determined that the calculation should not be considered to be a TLAA; therefore, this section is deleted from the application.

4.7.3 Dose to Seal Rings for the High Pressure Coolant Injection and Reactor Core Isolation Cooling Containment Isolation Check Valves

Although this TLAA was included in the initial LRA, the applicant by its letter dated June 9, 2005, made a review of the safety determination per 10 CFR 54.3, and stated as follows:

> LRA Section 4.7.3 originally considered a design calculation that determines the dose to seal rings on the high-pressure coolant injection system and reactor core isolation cooling system testable check valves to be a TLAA. After further review, the applicant determined that the calculation is used to validate the seal design, but is not relied on to make a safety determination. The ability of the valve to perform its safety function is verified by Type C leak testing performed per BFN Technical Instruction 0-TI-360, "Containment Leak Rate Programs." Based on this further review, the applicant determined that the calculation should not be considered to be a TLAA, and that Section 4.7.3, “Dose To Seal Rings For The High Pressure Coolant Injection And Reactor Core Isolation Cooling Containment Isolation Check Valves,” should be deleted from the LRA.

The staff concurred with the applicant’s assessment that this is not a TLAA and its determination not to include it in the safety evaluation.

4.7.4 Radiation Degradation of Drywell Expansion Gap Foam

4.7.4.1 Summary of Technical Information in the Application

In LRA Section 4.7.4, the applicant stated that the steel drywell shell is enclosed in reinforced concrete for shielding purposes and to provide additional resistance to deformation and buckling of the drywell over areas where the concrete backs up the steel shell. The drywell is separated from the reinforced concrete by a gap of approximately 2 inches and filled with polyurethane foam.

4.7.4.2 Staff Evaluation

In RAI 4.7.4-1, dated December 10, 2004, the staff stated that LRA Table 3.5.2.2 lists the aging management review (AMR) results of expansion joint (elastomer, polyurethane foam) as a TLAA and refers the TLAA to LRA Section 4.7. LRA Section 4.7.4, "Radiation Degradation of Drywell Expansion Gap Foam," states that an analysis of the effect of dose on the foam showed that the material properties will remain within the limits assumed by the original design analysis for the additional 20 years of extended operation. Therefore, the staff requested the applicant to provide a more detailed discussion of the analysis,[1] including a discussion of the method and assumptions adopted in the analysis, the type of data extrapolation applied, and the quantitative results obtained to justify the applicant's assertion that the requirements of 10 CFR 54.21(c)(1)(i) are fully met.

In its response, by letter dated January 31, 2005, the applicant stated that:

> The TLAA analysis determines that the total dose to the polyurethane foam located between the drywell steel and the reactor building concrete will result in a total dose of less than 1.0E8 rads. The material properties of the polyurethane foam will remain within the limits assumed by the original analysis for a total dose of less than 1.0 E08 rads.
>
> The analysis model consists of the standard geometry sphere with a steel clad of 0.825 inches (drywell steel thickness). The radius of the sphere is 33.5 feet. Computer code QAD-P5Z, which is a point kernel variation of QAD-P5F, was used to determine dose and/or exposure rates. The computer code PARINT integrated the dose rates over time. The principle gamma source from normal operation is N-16; therefore the photon spectrum for normal operation is for N-16 with an arbitrary 1 Ci activity as input. The resultant dose rate was then scaled to the appropriate power level. The STP computer code determined the time dependent photon spectra. STP is the standard TVAN computer code for source term development. Gamma and neutron attenuation are considered.
>
> Actual power conditions are utilized in the TLAA analysis. This applies for roughly the first 25% of plant life during which time each unit was down for a significant amount of time. For conservatism, it is assumed that EPU starts October 24, 2003, even though Unit 1 has yet to be restarted. Prior to October 24, 2003, Units 2 and 3 are at 105% (uprate) conditions. For an additional conservatism, Permali neutron shielding has not been included in the TLAA analysis.
>
> The foam will only receive the significant dose from the drywell. The drywell is surrounded by a minimum of 5 feet of concrete. It is clear that the drywell sources will have a greater impact than any sources in the reactor building. The reactor building source impact will be negligible compared to the drywell.

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[1]TVA by letter dated January 7, 2005, agreed to decouple the power uprate licensing request from License Renewal Application. The safety review of this item will be further evaluated as part of the EPU review.

> The maximum dose for 60 year operation at EPU conditions without Permali neutron shielding occurs for Unit 2 and is 9.92E+07 which is less than a total dose of 1.0E08 rads used in the original analysis. Therefore, the material properties of the polyurethane foam will remain within the limits assumed by the original analysis.

In addition, the staff requested the applicant to provide tests or other research publication based justification for making the following assertion that: "The material properties of the polyurethane foam will remain within the limits assumed by the original analysis for a total dose of less than 1.0 E08 rads."

In its letter dated May 24, 2005, the applicant responded with the following:

> The basis for asserting that the polyurethane foam will maintain its material properties when exposed to radiation dosage is BFN UFSAR Section 5.2.3.2 which states in part "… Irradiation tests have shown that no change in the resilient characteristics will take place for exposures up to 10^8R." This is in accordance with BFN's current licensing basis. Additionally, this same information is presented in Section 4.7.4, "Summary Description," of the LRA.

The staff found that the applicant provided adequate engineering analysis results and related material test data to fully resolve the RAI. Therefore, the staff's concern described in RAI 4.7.4-1 is resolved.

4.7.4.3 UFSAR Supplement

UFSAR Section 5.2.3.2 states that irradiation tests have shown that no change in the resilient characteristics will take place for exposures up to 1.0×10^8 rads. This test-based material performance data, in conjunction with the above-discussed TLAA analysis results, form the basis for the staff's determination that the effects of aging due to radiation degradation of drywell expansion gap foam will be adequately managed. The applicant provided UFSAR supplement summary description of drywell expansion gap foam in LRA Section A.3.5.3. On the basis of its review, the staff concluded that the UFSAR supplement summary adequately describes the TLAA in LRA Section 4.7.4, "Radiation Degradation of Drywell Expansion Gap Foam."

4.7.4.4 Conclusion

The staff reviewed the applicant's TLAA on radiation degradation of drywell expansion gap foam, as summarized in LRA Section 4.7.4, including information submitted in response to the staff's RAI and determined that the effects of aging due to radiation degradation of drywell expansion gap foam will be adequately managed. Therefore, the staff concluded that the applicant has demonstrated that the effects of aging due to radiation degradation of drywell expansion gap foam will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).

4.7.5 Corrosion – Minimum Wall Thickness

Although this TLAA was included in the initial LRA, the applicant by its letter dated June 15, 2005, made a review of the safety determination per 10 CFR 54.3, and stated as follows:

LRA Section 4.7.5 originally considered a design calculation that shows corrosion/erosion resulting in decreasing pipe wall thickness to be a TLAA. The functional basis for ensuring the wall thickness acceptability is accomplished by inspection, testing, and monitoring activities performed by plant procedures implementing SPP-9.7, Corrosion Control Program. Based on its further review, the applicant determined that the calculation should not be considered a TLAA; therefore, this section is deleted from the application.

The staff concurred with the applicant's assessment that this is not a TLAA and with its determination not to include in the safety evaluation.

4.7.6 Irradiation Assisted Stress Corrosion Cracking of Reactor Vessel Internals

4.7.6.1 Summary of Technical Information in the Application

The applicant in LRA Section 4.7.6 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 5×10^{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.

4.7.6.2 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 x 10^{20} n/cm^2 (E > 1 MeV) and 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 are managed by two aging management programs (AMPs): (1) Chemistry Control Program, and (2) Boiling Water Reactor Vessel Internals Program. The Boiling Water Reactor Vessel Internals Program in turn addresses several BWRVIP inspection programs that are designed for various RV internal components. In addition, the Boiling Water Reactor Vessel Internals Program invokes the ASME 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 CLB, 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.

<u>Top Guide</u> - In addition to the implementation of the Chemistry Control Program, and the Boiling Water Reactor Vessel Internals Program, 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 has been approved by the staff. The implementation of these additional guidelines and the AMPs is consistent with the GALL AMP XI.M9. The staff found that, by implementing a proper chemistry program as dictated by the Chemistry Control Program, the oxidizing nature of the RCS water can be controlled and, thereby, the corrosion of the top guide can be controlled.

In RAI B.2.1.12-1(A), dated December 1, 2004, the staff indicated that the BWRVIP-26 report lists 5 x 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 x 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, GE assumed a lower allowable stress value, acknowledging the high fluence value at this location. The conclusion of GE’s analysis, and the fact that a single failure at this location has no safety consequence, was that no inspection was necessary to manage IASCC in top guide grid beams.

The staff was 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 Nuclear Power station. 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 cannot be demonstrated, then an inspection program to manage this aging effect to preclude loss of component intended function is required.

In its response, 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 BWRVIP. The BWRVIP implements the requirements of NRC-accepted BWRVIP-26. The 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, as part of the BWRVIP, to work 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 determined that the applicant was required to submit, for NRC review and approval, a site-specific AMP that addresses potential multiple failures of the top guide grid beams. The applicant, in its response dated May 25, 2005, indicated that it will perform inspections of the guide beams similar (in inspection methods, scope and frequency of inspection) to the inspections specified in the BWRVIP-47, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines," for the control rod guide tube components. The applicant stated that the extent of examination and its frequency will be based on a ten percent sample of the total population, which includes all grid beam and beam-to-beam crevice slots, within 12 years and five percent of the population is to be completed within six years. The applicant stated that the program to inspect the top guide grid beams will be implemented prior to the end of the current license period. The sample locations selected for examination will be in areas that are exposed to highest neutron fluence. The staff found this response acceptable because it defines a representative population of IASCC susceptible locations, and selects locations in the top guide that are exposed to the highest neutron fluences. In addition, the proposed inspection requirements were previously accepted by the staff in the SE related to the license renewal of Peach Bottom Atomic Power Station, Units 2 and 3. The staff considered this RAI resolved.

Core Shroud - In addition to the implementation of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, Chemistry Control Program, and BWR Vessel Internals Program, 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 requested 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 found that, by implementing a proper chemistry program as dictated by the Chemistry Control Program, the oxidizing nature of 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 ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, 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 AMPs is consistent with the GALL AMP XI.M9, and Table IV.B1.1-a through IV.B1.1-g, the staff found that the applicant had demonstrated that the effects of aging will be adequately managed so

that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

In RAI 4.2.4-1(B), dated December 1, 2004, the staff stated that 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 (E > 1.0 MeV). Therefore, the staff requested that the applicant address the aging effect due to IASCC in the core shroud component.

In its response, by letter January 31, 2005, the applicant stated that the core shrouds are classified as "Category C," based on the core shroud classification criteria contained in Appendix B of the BWR Vessel Internals Program. The BWR Vessel Internals Program 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 accepted it (pending the approval of the BWRVIP-76 report) because implementation of the BWR Vessel Internals Program and the Chemistry Control Program would adequately manage the aging effect due to IASCC in the core shroud components and is consistent with GALL AMP XI.M9 and XI.M2.

Core Plate - The applicant proposed to implement the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, Chemistry Control Program, and BWR Vessel Internals Program. The BWR Vessel Internals Program in turn invokes the inspection guidelines of the BWRVIP-25, "Boiling Water Reactor Core Plate Inspection and Flaw Evaluation Guidelines," which has been approved by the staff. The staff found that by implementing a proper chemistry program as dictated by the Chemistry Control Program, 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 the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, 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 AMPs is consistent with the GALL AMP XI.M9, and Table IV.B1.1-a through IV.B1.1-g, the staff found that the applicant had demonstrated that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the CLB 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 ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, the Chemistry Control Program, and BWR Vessel Internals Program, the applicant committed to invoke the inspection guidelines specified in BWRVIP-47, "Boiling Water Reactor Lower Plenum Inspection and Flaw Evaluation Guidelines," which has been approved by the staff. The staff found that by implementing a proper chemistry program as dictated by the Chemistry Control Program, 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 the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, 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 AMPs is consistent with the GALL Report, the staff found 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 CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3).

4.7.6.3 UFSAR Supplement

Section LRA A.3.5.5 includes the following UFSAR 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. The applicant, in its response dated May 25, 2005, revised LRA Section A 3.5.5 of the UFSAR supplement summary description which describes that the inspection guidelines that are specified in the BWRVIP-25 report will be implemented for managing the aging effect on core plate. The staff considered this acceptable.

4.7.6.4 Conclusion

The staff reviewed the applicant's TLAA on IASCC of the RV internals, as summarized in LRA Section 4.7.6, and determined that, except for the top guide grid beams, the applicant appropriately describes that by implementing the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, the Chemistry Control Program and BWR Vessel Internals Program, and relevant additional BWRVIP guidelines related to RV internal components, the aging effect due to IASCC will be adequately managed for the period of extended operation. The license renewal action items related to the implementation of the BWRVIP-25, BWRVIP-26, and BWRVIP-47 guidelines are discussed in SER Section 3.1 on AMR. In addition, the staff believes that the implementation of these additional guidelines and AMPs is consistent with the GALL AMP XI.M9, and Table IV.B1. Therefore, the staff concluded that the applicant had 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 CLB for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).

4.7.7 Stress Relaxation of the Core Plate Hold-Down Bolts

4.7.7.1 Summary of Technical Information in the Application

The core plate hold-down bolts connecting the core plate to core shroud are initially preloaded during installation. These bolts are subject to stress relaxation due to thermal and irradiation effects. The loss of preload over time due to stress relaxation is considered a TLAA and evaluated accordingly. In the LRA, the applicant stated that it evaluated the loss of preload of the core plate hold-down bolts for the 40-year lifetime and concluded that all core plate hold-down bolts will maintain some preload throughout the life of the plant. This conclusion was based on an analysis of loss of preload for core plate hold-down bolts, referenced in BWRVIP-25, Appendix B, "BWR Core Plate Inspection and Flaw Evaluation Guidelines." (Reference 5). For the 60-year lifetime, the applicant estimated the expected loss of preload to be less than 20 percent. With this loss of preload, the applicant stated that the core plate will maintain sufficiently high preload at the end of the period of extended operation to prevent sliding under both normal and accident conditions. Based on this assumption, the applicant concluded that the loss of preload is acceptable for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.7.7.2 Staff Evaluation

To complete its review, the staff requested additional information regarding the data and analyses that were used to determine that the loss of preload due to stress relaxation at the end of the period of extended operation would be less than 20 percent. The staff also requested that the applicant show that the hold-down bolts would meet the required ASME Code Section III stress acceptance criteria at the end of the period of extended operation.

In RAI 4.7.7-1, dated March 3, 2005, the staff requested that the applicant demonstrate the applicability of BWRVIP-25, (Reference 5) Appendix A, core plate hold-down bolt analysis to the BFN units, based on the configuration and the geometry of the BFN core plate hold-down bolts and the reactor environment (temperature and neutron fluence) assumed in the report.

In its response, by letter dated May 31, 2005, the applicant stated that the BFN core plate corresponds to that in BWRVIP-25, Figure 2-4, and that BFN was specifically considered in the original BWRVIP-25 evaluation, incorporating typical values of temperature and fluence. An analysis was initially performed for a 40-year plant life, and subsequently for a 60-year plant life, as discussed in BWRVIP-25, Appendix B, paragraph B.4, which addressed license renewal. This initial BWRVIP-25 based analysis assumed 20 percent hold-down bolt preload relaxation over a 60-year plant life.

To address EPU conditions in conjunction with license renewal, the applicant stated that a plant-specific calculation was performed for the BFN units. This calculation was based on the BFN fluence calculation which was performed considering EPU operating power and time conditions. The applicant stated that the applicable maximum bolt fluence was determined to be 5×10^{19} n/cm^2 (E > 1MeV) at the end of the 60-year plant life. The resulting hold-down bolt load relaxation was determined to be 15 percent, based on General Electric Nuclear Energy (GENE) design documents.

The staff reviewed this response and concluded that additional information was needed to complete its evaluation. The additional information was requested in the follow-up to RAI 4.7.7-1 which is discussed later.

In RAI 4.7.7-2, dated March 3, 2005, the staff requested that the applicant:

(a) Identify the temperature of the bolts during the normal operation and the projected bolt neutron fluence at the end of the period of extended operation.

(b) Explain how it was determined that the effects of temperature and neutron fluence result in a 20 percent loss of preload.

(c) Provide a detailed description of the methodology and data used at BFN to perform the analysis as described in (b), and include the basis for the relaxation curves.

In its response to RAI 4.7.7-2, dated May 31, 2005, the applicant responded as follows:

(a) The normal operating temperature for the core plate bolts is 550°F. For the BFN units, the projected fluence was determined to be 5×10^{19} n/cm^2 (E > 1MeV) for a 60-year lifetime, (assuming a 90 percent capacity factor) for the bolt at the peak radial location. The arrangement of the bolts around the periphery of the core plates assures that many of the bolts experience a significantly lower lifetime fluence.

(b) The plant-specific evaluation used GENE proprietary relaxation curves from a GENE material design document for irradiated stainless steel properties at 550°F, that was developed in the1970s time frame. The document was based on a combination of GENE internal reports and industry data to evaluate bolt stress relaxation.

(c) The BFN calculation was performed based on the BFN-specific core plate geometry, fluence and temperature. The BFN fluence conditions and the expected bolt stress relaxation made use of either GENE methods or GENE design documents. In support of the relaxation value used in the calculations, the applicant provided relaxation vs. fluence data from BWRVIP-99, "Crack Growth Rates in Irradiated Stainless Steels In BWR Internal Components." (Reference 6). This data was developed for type 316 stainless steel material, based on data found in the literature. The applicant justified the application to type 304 stainless steel material on the basis that the two commercial material alloys have the same single-phase austenitic microstructure and crystal structure, with no precipitates present in either alloy, and the mechanical properties are essentially identical at 550°F.

The staff reviewed the information in this response and concluded that additional information was needed to complete its evaluation. The additional information was requested in RAI 4.7.7-3 through 4.7.7-7 by letter dated June 22, 2005.

In RAI 4.7.7-3, dated June 22, 2005, the staff requested that the applicant provide the data that GENE used to develop the stress relaxation curves and explain how this data was utilized to establish the curves.

In its response, by letter dated June 29, 2005, the applicant presented a mean design curve developed by GENE using stress relaxation values of irradiated stainless steel materials. The data was obtained from measurements made on springs and bent-beam specimens.

The staff's review of the applicant's response to RAI-4.7.7-3 is included in the staff's review of RAI 4.7.7-4.

In RAI 4.7.7-4, dated June 22, 3005, the staff stated that the applicant referenced BWRVIP-99 report, Figure 7-13, which shows data and modeling projections for stress relaxation versus fluence values measured in displacements per atom (dpa) for 20 percent cold-worked type 316 stainless steel material. The staff requested that the applicant provide an explanation justifying the applicability of the Type 316 stainless steel data to the Type 304 stainless steel core plate hold-down bolts at the BFN units.

In its response, by letter dated June 29, 2005, the applicant stated that the stress relaxation property of irradiated stainless steel materials does not vary with change in chemical composition. To support this claim, the applicant provided Halden (in-situ tests in the Halden reactor) data which show that there is very small variation in stress relaxation values between Type 304, 316, and 348 stainless steel specimens. The stress relaxation data were obtained from specimens that were exposed to 4.4 to 6 x 10^{20} n/cm^2 (E > 1MeV) in 288 °C water. These neutron fluence values are nearly 10 times higher than that of BFN core plate hold down bolts; therefore, stress relaxation values for the BFN bolts will be less than the values that are presented in the data. The applicant compared the Halden data with GENE data and concluded that for a given neutron fluence value the corresponding stress relaxation value that is obtained from the GENE data is more conservative than that from the Halden data.

The staff reviewed the applicant's responses to RAIs 4.7.7-3 and 4.7.7-4 and concluded that supporting data to the applicant's claim that the variation in chemical composition of stainless steel materials has very little effect on the stress relaxation of the irradiated stainless steel materials. Therefore, the staff concluded that the stress relaxation curves for the irradiated Type 316 stainless steel material can be applicable to irradiated Type 304 stainless steel materials. The staff reviewed the data in the applicant's response dated June 29, 2005, and found that for a given neutron fluence value the corresponding stress relaxation value obtained from GENE data is conservative and is acceptable.

In RAI 4.7.7-5, dated June 22, 2005, the staff requested that the applicant provide the dpa values for Type 304 core plate hold-down bolts that correspond to end-of-life fluence (54 EFPY) using appropriate model for the BFN units.

The staff’s review of the applicant’s response to RAI 4.7.7-5 is included in the staff’s review of the follow-up to RAI 4.7.7-1.

In RAI 4.7.7-6, dated June 22, 2005, the staff requested that the applicant provide justification for the application of relaxation curves obtained based on data from torsion tests to axial relaxation in bolts.

In its response, by letter dated June 29, 2005, the applicant stated that the GENE stress relaxation data is obtained from test samples that include springs that represent torsional loading, and bent-beam specimens that represent tension loading. The applicant presented stress relaxation data that represented tension loading and another set representing shear loading, and they both exhibit similar behavior as GENE stress relaxation curve, but at a lower value. The data also indicated that the stress relaxation curve was not affected by the specimen

or type of loading. Therefore, the applicant concluded that the stress relaxation values that are presented are applicable for torsional and axial loadings.

The staff reviewed the applicant's response and concluded that the stress relaxation curves and the applicant's presented data on the stress relaxation values are applicable for torsional and axial loadings.

In RAI 4.7.7-7, dated June 22, 2005, the staff requested that the applicant provide the calculations referenced in Appendix B of BWRVIP-25 so that it can evaluate the stress relaxation of the core plate hold-down bolts for the end-of-license fluence (54 EFPY) for the BFN units.

In its response to RAIs 4.7.7-5 and 4.7.7-7, dated June 29, 2005, the applicant provided a proprietary response in reply to the staff RAIs (ADAMS Accession No: ML052150189). In the response the applicant stated that a plant-specific calculation was performed for the BFN units using a neutron fluence value of 5×10^{19} n/cm^2 (E > 1MeV) which is equivalent to 0.07 displacement damage (measured as dpa) at the peak fluence location. The dpa value is calculated based on the calculated fast fluence and an effective dpa cross section (E > 1 MeV) of approximately 1380 barns for steel. The GENE stress relaxation value for this neutron fluence and dpa values is 15 percent, which is a conservative value, falls within the bounding value of 20 percent that was specified in the BWRVIP-25 report. The staff's review of the applicant's response to RAI-4.7.7-5 is included in the staff's review of the follow-up to RAI 4.7.7-1.

The staff reviewed the information in the responses to RAI 4.7.7-3 through 4.7.7-7, and concluded that additional information was needed to complete its evaluation. The additional information was requested in follow-up to RAI 4.7.7-1 and 4.7.7-2 by letter dated August 2, 2005.

In the follow-up to RAI 4.7.7-1, dated August 2, 2005, the staff indicated that in the data provided by TVA in its submittal dated June 29, 2005, the applicant compared the stress relaxation for the BFN core plate hold-down bolts to the stress relaxation data derived from springs and stainless steel bent beam specimens. The staff requested that the applicant provide information regarding the values of neutron flux and temperature at which the bent beam and spring test specimens were exposed, and compare them to the neutron flux and temperature values of the BFN core plate hold-down bolts. If these neutron flux and temperature values are different from those for the spring and bent beam specimens, the staff requested that the applicant evaluate the impact of these differences on the predicted stress relaxation values of the BFN core plate hold-down bolts.

In its response to the follow-up to RAI 4.7.7-1, dated September 6, 2005, the applicant addressed the effects of temperature and neutron flux on the stress relaxation values at which the irradiation tests were conducted. In its response, the applicant stated that given the large range of higher flux for which the properties are the same, the impact of the lower flux to which the bolts are exposed is viewed to be negligible. In support, the applicant stated that the temperature and fluxes associated with the design basis data are appropriate for use in predicting stress relaxation in the BFN core plate bolts. The test data was all generated at temperatures from 530 °F to 600 °F and, therefore, is fully representative of BWR operating conditions. The nuclear spectrum is also similar to that for the core plate bolt region. While the

test data was generated at higher fluxes than present in the core plate region, the applicability of the data for use in the core plate bolt assessment is supported by mechanistic understanding as well as component test results.

Since the temperatures at which the majority of the irradiation tests were conducted represent the temperatures of the core plate hold-down bolts at the BFN units, the applicant claimed that the stress relaxation data that was provided by GENE would be representative of the BFN core plate hold-down bolts. The applicant further reiterated that the tests conducted at a neutron flux value higher than that of the core plate hold down bolts can be applicable for evaluating the stress relaxation data for the BFN's core plate hold-down bolts.

The staff reviewed the applicant's responses to the aforementioned RAI and determined that the applicant's justification for using the GENE methodology in the applicant’s response in developing the stress relaxation curves is acceptable for the following reasons. GENE developed the stress relaxation curve for irradiated austenitic stainless steel materials at temperatures equivalent to the BWR normal operating temperatures and at a neutron fluence value equivalent to 54 EFPY for the BFN units. The stress relaxation data demonstrates that the impact of test temperature and neutron flux values for the test samples are not significant. The stress relaxation curve indicates that the relaxation value for the neutron fluence equivalent to 54 EFPY at the BFN units is 15 percent. The staff concluded that the stress relaxation value of 15 percent is a conservative value and falls within the bounding value of 20 percent that was provided in the generic analysis of the staff-approved BWRVIP-25 report.

In the follow-up to RAI 4.7.7-2, dated August 2, 2005, the staff requested that the applicant show that, under design basis accident condition loading stated in Scenario 3 of BWRVIP-25, Appendix A, the axial and bending stresses for the mean and highest loaded hold-down bolts will not exceed the ASME Section III allowable stresses for P_m (primary membrane) and $P_m + P_b$ (primary membrane plus bending) as a result of a 20 percent reduction in the specified bolt pre-load. The staff also requested that the applicant state clearly the assumptions on which the analysis was based.

In its response to the follow-up RAI 4.7.7-2, dated September 6, 2005, the applicant indicated that the BFN current licensing basis states that: "Two considerations important to the core support evaluation are sliding of the core support and buckling of the supporting beams. Evaluations have determined that the core support will not slide under postulated accident conditions with preload on the hold-down bolts. Additional resistance to sliding is provided by aligning pins which further stabilize the core support." The applicant also provided a (proprietary) stress calculation of the hold-down bolts which demonstrated that the axial and bending stresses met the stress criteria in BWRVIP-25, Appendix A.

The staff reviewed the applicant's response and identified the following concerns:

- The analysis does not correspond to the plant-specific core plate/hold-down bolt analysis recommended in Appendix A of BWRVIP-25. The applicant's analysis assumes that the core plate is rigid. The recommended approach is based on an elastic finite element analysis of the core plate/hold-down bolts.
- The applicant selected friction due to hold-down bolt preload as the means to prevent sliding of the core plate under horizontal loading. BWRVIP-25 recommends the

installation of wedges to prevent sliding; it does not recommend high preload to induce sufficient friction to prevent sliding. No basis for this choice was provided.

- The analysis is based on stipulated high preload (including 20 percent relaxation) of the hold-down bolts and a high static coefficient of friction to prevent sliding of the core plate under accident basis horizontal loading. No basis was provided for this high static coefficient of friction. Based on a comparison with values found in the literature, the coefficient of friction used in the analysis is similar to that stipulated as friction between dry metal surfaces. This value is not considered applicable to friction between the core plate and its shroud support, which are immersed in a BWR hot water environment. The staff believes that the static coefficient of friction in this environment is considerably lower, similar to that for friction between lubricated metal surfaces.
- As a result of the assumed rigidity of the core plate and high coefficient of static friction, and leading to the prevention of sliding under horizontal loading, the only stress state in the hold-down bolts is axial, caused by the bolt pre-load and vertical loading on the core plate. The core plate/hold-down bolt analysis in BWRVIP-25, Appendix A is based on relatively low bolt pre-load and no friction. As a result, the core plate is restrained from sliding by the hold-down bolts only, which induces bending stresses in the bolts. A low coefficient of friction may show that core plate sliding under the horizontal loading may not be prevented, thus inducing bending stresses in the hold-down bolts, in addition to the axial stresses.
- BWRVIP-25 indicates that "of special interest is the amount of bending induced in the bolts when the core plate bows upward, or when load from the beams is no longer transferred to the rim." This effect cannot be determined from the applicant's analysis if the core plate is assumed rigid.
- The stipulated hold-down bolt preload in the applicant's analysis is considerably larger that the preload in the analysis in BWRVIP-25, Appendix A. The effect of this preload on the structural integrity of the core plate was not evaluated.
- The finite element analysis of the core plate/hold-down bolts in Appendix A shows that the axial and transverse bolt loads vary around the circumference of the core plate. The axial loads in the highest loaded bolts are about twice the mean of the axial bolt loads. The applicant's analysis, based on a rigid plate analysis, shows that all bolts are uniformly loaded in tension and does not reflect the true distribution of the bolt loads.
- BWRVIP-25 specifies the design basis accident loads that should be considered in a plant-specific analysis. It is not clear that all applicable loads were considered in the applicant's analysis.

Based on these concerns, the staff concluded that the applicant did not provide reasonable assurance that the axial and bending stresses in the hold-down bolts will meet the ASME Section III primary stress limits as stated in BWRVIP-25, Appendix A, under the BFN plant-specific design basis accident loading and with 20 percent relaxation of hold-down bolt preload. This was, therefore, identified by the staff as Open Item 4.7.7.

Follow-up teleconferences with the applicant were held on October 14 and 18, 2005, to address the resolution of Open Item 4.7.7. This open item was included as one of four open items in an interim evaluation by the Advisory Committee on Reactor Safeguards of BFN's license renewal application and in the NRC's draft Safety Evaluation Report. By letter dated October 31, 2005,

the staff provided the applicant a summary and discussion of the teleconferences, in which the staff position on this open item was summarized. The letter summarized the staff's concerns, as follows:

The applicant did not use the staff-approved analysis that was used in BWRVIP-25 report for the BFN units. The methodology used in the BWRVIP-25 report is more conservative. For BFN units, the applicant used a less conservative methodology, such as using a high static coefficient of friction value to ensure prevention of sliding of the core plate which eliminated the bending stresses in the core plate hold-down bolts. The staff determined that the static coefficient of friction used by the applicant is not supported by the available information provided in the literature.

The staff also questioned whether the applicant had considered using wedges to prevent core plate sliding, and if the wedges are installed, the aging management of core plate hold-down bolts will not be considered a TLAA item. The applicant stated that this option was evaluated but it is costly to install wedges in each unit. This option was, therefore, withdrawn.

The staff identified and summarized the following concerns:

(1) The analysis is significantly different from the structural analysis in BWRVIP-25, and is not based on a finite element model of the core plate.

(2) It is not clear that all loads listed in BWRVIP-25, such as fuel lift load, were included in the analysis.

(3) The applicant selected friction due to high bolt preload (significantly larger than that specified in BWRVIP-25) as the means to prevent side motion of the core plate. BWRVIP-25 recommends the use of wedges to prevent side motion; it does not recommend high bolt preload and friction.

(4) The applicant analysis assumes a high static coefficient of dry friction as the mechanism to prevent side motion of the core plate. The staff questions the basis for this assumption for a core plate that is in a BWR water environment.

(5) BWRVIP-25, Appendix A, page 4-6 states that "of special interest is the amount of bending induced in the bolts when the core plate bows upward, or when load from the beams is no longer transferred to the rim." No such bending was evaluated in the applicant's analysis.

(6) The BWRVIP-25 structural analysis shows a variation of the axial forces in the hold-down bolts with location around the plate circumference, and that the axial force in the highest-loaded bolt is about twice the mean axial bolt load. The applicant analysis shows that all bolts are uniformly loaded in tension. This indicates that the highest stresses in the hold-down bolts have not been determined.

(7) The effect of the large bolt preloads on the structural integrity of the core plate was not evaluated.

The staff stated its position that, for the BFN units, the applicant should apply the staff-approved methodology that was used in the BWRVIP-25 report.

By letter dated November 16, 2005, the applicant stated in Enclosures 3 and 9 the following commitment for BFN for the core plate hold-down bolts:

The applicant will perform a BFN plant-specific analysis consistent with BWRVIP-25 to demonstrate that the core plate hold-down bolts can withstand normal, upset, emergency, and faulted loads, as applicable, considering the effects of stress relaxation until the end of the period of extended operation. The installed core plate configuration and bolt preload will be used for the plant-specific analysis. The analysis will use the plant-specific design basis loads and load combinations. The analysis will incorporate detailed flux/fluence analyses and improved stress relaxation correlations.

In accordance with BFN's CLB, the ASME Boiler and Pressure Code, Section III will be used as a guide in determining limiting stress intensities for reactor vessel internals. For those components for which stresses exceed the ASME Code allowables, either the elastic stability of the structure or the resulting deformation or displacement will be examined to determine if the safety design basis is satisfied. Appropriate corrective action will be taken if the plant-specific analysis does not satisfy the above criteria. The installation of core plate wedges to eliminate the need for the enhanced inspections of the core plate hold-down bolts as recommended by BWRVIP-25 is considered an acceptable corrective action.

The analysis or the corrective action taken to resolve this issue will be submitted to the staff for review two years prior to the period of extended operation.

The staff reviewed the applicant's commitment and concluded that it provides adequate assurance that the 60-year stress relaxation of the core plate hold-down bolts due to neutron exposure will not compromise the structural integrity and operability of the core plate to the end of the period of extended operation. Open Item 4.7.7 is, therefore, closed.

4.7.7.3 UFSAR Supplement

In a letter dated November 16, 2005, the applicant revised LRA Section A.3.5.6 to include the UFSAR supplement summary description for the TLAA on stress relaxation of the core plate hold-down bolts. On the basis of its review of the UFSAR supplement, the staff concluded that the summary description of the applicant's actions to address stress relaxation of the core plate hold-down bolts is adequate.

4.7.7.4 Conclusion

The staff concluded that the applicant's commitment to provide a revised analysis, two years prior to the start of the period of extended operation, regarding the stress relaxation TLAA of the core plate hold-down bolts, and that the analysis will remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii), is acceptable. The staff also concluded that the UFSAR supplement contains an appropriate summary description of this TLAA evaluation, sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.7.8 Emergency Equipment Cooling Water Weld Flaw Evaluation

4.7.8.1 Summary of Technical Information in the Application

The TLAA of the EECW weld flaw evaluation is discussed in LRA Section 4.7.8. The applicant performed an analysis on 17 selected EECW system piping welds that have flaws. The original analysis included a stress evaluation of the flawed welds and fatigue crack growth calculations. The fatigue crack growth calculations were based on a conservative projection of 125 cycles for the remaining 25 years of the 40-year plant operating life based on five cycles per year. A cycle occurs when piping, including a subject weld, is removed from service then returned to service. This projection was derived from a very conservative estimate that each weld could experience up to five cycles per year. Review of the system function indicated that continuous operation is intended; however, some interruptions have been required for maintenance and other considerations. The applicant considers the fatigue crack growth portion of this analysis to be a TLAA.

As part of the LRA, the applicant found, based on current and recent plant operating experience, that it is unusual for any of these weld locations to experience more than one cycle in any given year. For the TLAA, the applicant assumed two cycles per year for the past and the foreseeable future. The cycle count of two cycles per year was applied to the 25 remaining operating years (projected when the calculations were performed), plus the 20 years of extended operation, resulting in a total cycle count of 90. This is less than the estimated cycle count used for qualification in the original calculation. Therefore, the applicant's position is that in accordance with 10 CFR 54.21(c)(1)(i), the analyses remain valid for the period of extended operation.

4.7.8.2 Staff Evaluation

As required by 10 CFR 54.21, applicants for license renewal must manage time-dependent aging effects by one of three acceptable methods:

1. Demonstrate that the TLAA on the aging effect for the current operation term remains valid for the period of extended operation.
2. Demonstrate that the TLAA on the aging effect for the current operation term and has been projected to the end of the period of extended operation.
3. Demonstrate that the effect of aging on the intended functions will be adequately managed for the period of extended operation.

In RAI 4.7.8-1, dated November 4, 2004, the staff requested that the applicant provide background information, including the code class, flaw inspection history, flaw sizes, and a description of any analysis including the method that was used to determine the flaw evaluation. In its response, by letter dated December 9, 2004, the applicant stated, in part:

> The flawed EECW welds are on BFN Seismic Class I piping that was designed to the B31.1-1967 Power Piping Code. For the BFN ASME Section XI program the welds are classified as ASME Class 3. Design conditions for the EECW system are 200 psig and 200 °F. All of the related piping is qualified by analysis. This analysis satisfies BFN

Design Criteria No. BFN -50-C-7103 which supplements B31.1 analysis requirements by invoking plant condition dependent stress equations from ASME Section III, 1971 Edition, Summer 1973 Addenda. The stress analyses of the piping systems are also considered a Time Limited Aging Analysis (TLAA) which is addressed in the Application TLAA Section 4.3.3.

History of Discovery – A weld inspection program was initiated at BFN to determine the effects of MIC on the stainless steel piping girth butt welds in the EECW system, as a result of MIC discoveries at other plants. The inspection program was implemented by performing radiography on a sample of EECW piping welds. Radiography had not been performed on these welds during installation, as it was not required by the applicable code and specifications. The inspection identified defects in 33 welds. The 33 welds which had identified defects were reviewed by the ISI Level III interpreter and 27 of the welds were rejected because they did not meet ISI flaw acceptance standards. The ISI Level III interpreter determined that the other welds did meet flaw acceptance standards.

Analysis Performed – Two analyses were performed in association with the qualification of the remaining 27 EECW welds with welding defects.

The applicant performed a bounding fracture mechanics analysis for the scope of stainless steel EECW pipe sizes encompassing the 27 welds that had been rejected based on ASME Section XI acceptance standards. Of the 27 welds, 10 were found to be acceptable using the bounding fracture mechanics analysis. The remaining 17 welds are the subject of the TLAA.

For the 17 welds identified in LRA Section 4.7.8, the applicant indicated that a location-specific fracture mechanics analysis was performed. The weld-specific analysis applied essentially the same approach and considerations as the bounding analysis except that location-specific stresses determined for ASME Code Section III, Subsection NC-3652, Equations 9 and 10 in the piping analyses of record were used to calculate both the ASME Code Section XI allowable flaw size and the fatigue crack growth due to cyclic load for the 25 years remaining in the plant life. The applicant found that for the controlling location (i.e., maximum thermal stress) in each pipe size applicable to the 17 welds, fatigue crack growth for the 25-year period was insignificant. Although the staff did not perform a detailed review of the applicant's analysis, the staff found the applicant's approach acceptable. The remaining issue is whether the applicant's demonstration that the TLAA on the aging effect for the current operation term remains valid for the period of extended operation.

The applicant stated in its LRA that, based on current and recent plant experience, it is unusual for any of these weld locations to experience more that one cycle in any given year.

The applicant stated that review of the EECW system indicates that continuous operation is intended; however, some interruptions have been required for maintenance and other considerations. Through an informal request on January 31, 2005, the staff requested the applicant to provide the following information as a follow-up to RAI 4.7.8: (a) Based on the design function of the EECW system, discuss when and at what frequency would the system be shut down; (b) Based on the design function and the total past history, discuss whether the number of cycles in the fatigue evaluation bound the number of cycles projected for the period of extended operation; (c) Describe events, and the frequency that they have occurred, that

resulted in system operational interruptions; and (d) Should the EECW system experience more cycles than is bounded by the applicant's analysis, discuss any plant procedures in place to identify this condition.

The applicant responded by letter on March 2, 2005, and provided the following as a follow up to RAI 4.7.8:

> The EECW system is intended to be in a continuous standby condition (i.e. under pressure-minimum flow) in both shutdown and operating plant modes. As currently designed, sections of this system may be isolated and depressurized for routine maintenance or repair. Based on operating history and future (anticipated operations) a total of 125 full pressure cycles (0 psig to design operating pressure) was selected as a conservative measure to ensure the number of fatigue cycles would not be exceeded. The preventative maintenance work orders scheduled on this system are of a periodicity of no less than 96 weeks (almost 2 years) and unless unexpected repairs are required, the system would not need to be depressurized. Using a conservatism of a little over 2 times in a year makes sense for it would be very unlikely for the same Section of the EECW system to be shutdown [sic] > 2 times in a year. Please review preventative maintenance scheduled items on [the] following page [Not included in this evaluation. See March 2, 2005 letter]. An administrative tracking system will be developed and used to ensure that the 125 fatigue cycles will not be exceeded.

Based on operating history and anticipated future operations coupled with the applicant's commitment to develop an administrative tracking system to ensure that the EECW system does not exceed the applicant's 125 full pressure cycles, the staff concluded that the EECW weld flaw evaluation is valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i) and is, therefore, acceptable.

4.7.8.3 UFSAR Supplement

As required by 10 CFR 54.21(d), applicants for license renewal must include a UFSAR supplement summary description of the "programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation." The applicant provided a UFSAR supplement summary description of EECW weld flaw evaluation in LRA Section A.3.5.7. On the basis of its review, and the responses to the staff's RAIs, the staff concluded that the UFSAR supplement summary adequately describes the TLAA on EECW weld flaw evaluation and is, therefore, acceptable.

4.7.8.4 Conclusion

The staff reviewed the applicant's TLAA on EECW weld flaw evaluation, as summarized in LRA Section 4.7.8, including information submitted in response to the staff's RAIs, and determined that the effects of EECW weld flaw evaluation will be adequately managed. Therefore, the staff concluded that the applicant has demonstrated that the effects of EECW weld flaw evaluation will be adequately managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii).

4.8 Conclusion for Time-Limited Aging Analyses

The staff reviewed the information in LRA Section 4, “Time-Limited Aging Analyses.” On the basis of its review, the staff concluded that the applicant has provided an adequate list of TLAAs, as defined in 10 CFR 54.3. Further, the staff concluded that the applicant demonstrated that (1) the TLAAs will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i); (2) the TLAAs have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii); or (3) that the aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii). The staff also reviewed the UFSAR supplement for the TLAAs and found that the UFSAR supplement contains descriptions of the TLAAs sufficient to satisfy the requirements of 10 CFR 54.21(d). In addition, the staff concluded that no plant-specific exemptions are in effect that are based on TLAAs, pursuant to 10 CFR 54.21(c)(2).

With regard to these matters, the staff concluded that there is reasonable assurance that the activities authorized by the renewed licenses will continue to be conducted in accordance with the CLB, and that any changes made to the CLB, in order to comply with 10 CFR 54.29(a), are in accordance with the Atomic Energy Act of 1954, as amended, and the NRC’s regulations.

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SECTION 5

REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS[1]

In accordance with Title 10, Part 54, of the *Code of Federal Regulations* (10 CFR Part 54), the Advisory Committee on Reactor Safeguards (ACRS) will review the license renewal application (LRA) for the Browns Ferry Nuclear (BFN) Units 1, 2, and 3. The ACRS Subcommittee on Plant License Renewal will continue its detailed review of the LRA after this safety evaluation report (SER) is issued. The applicant and staff from the U.S. Nuclear Regulatory Commission (the staff) will meet with the subcommittee and the full committee to discuss issues associated with the review of the LRAs.

After the ACRS completes its review of the LRAs and the SER, the full committee will issue a report discussing the results of its review. An update to this SER will include the ACRS report. This update will also include the staff's response to any issues and concerns identified in the ACRS report.

[1] This section is revised. See BFN LRA SER Supplement 1.

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SECTION 6

CONCLUSIONS[1]

The staff of the U.S. Nuclear Regulatory Commission (NRC or the Commission) reviewed the license renewal applications for the Browns Ferry Nuclear, Units 1, 2, and 3, in accordance with Commission regulations and NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," dated July 2001. Title 10, Section 54.29, of the *Code of Federal Regulations* (10 CFR 54.29) provides the standards for issuance of a renewed license.

On the basis of its evaluation of the license renewal applications, the NRC staff concluded that the requirements of 10 CFR 54.29(a) have been met and that all open items and confirmatory items have been resolved.

The staff notes that any requirements of Subpart A of 10 CFR Part 51 are documented in Supplement 21 to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants: Regarding Browns Ferry Nuclear, Units 1, 2, and 3, Final Report," dated June 23, 2005.

[1] This section is revised. See BFN LRA SER Supplement 1.

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APPENDIX A
COMMITMENTS FOR LICENSE RENEWALS
OF BFN UNITS 1, 2, AND 3[1]

During the review of the Browns Ferry Nuclear Plant (BFN) license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff, the applicant made commitments related to aging management programs (AMPs) to manage aging effects of structures and components (SCs) before the period of extended operation. The following tables list these commitments, along with the implementation schedules and the sources of the commitments.

- Table 1 lists those commitments that are not for a specific unit.
- Table 2 lists commitments that are specific to Unit 1.

Note that these tables also contain non-AMP commitments.

[1] This commitment table is revised. See BFN LRA SER Supplement 1

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TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
1. Accessible Non-Environmental Qualification Cables and Connections Inspection Program	Develop and implement new program.	A.1.1	Prior to the period of extended operation	• LRA Section B.2.1.1
2. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program	Revise implementing documents for LPRM cable system aging to reference existing Technical Specification requirements and license renewal reference(s).	A.1.2	Prior to the period of extended operation	• LRA Section B.2.1.2 • Response to follow-up to RAI 2.5-2 dated March 2, 2005
	Develop and implement new program to manage IRM cable system aging.		Prior to the period of extended operation	• LRA Section B.2.1.2 • Response to follow-up to RAI 2.5-2 dated March 2, 2005
3. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program	Develop and implement new program to manage the medium-voltage cables to the Residual Heat Removal Service Water pumps.	A.1.3	Prior to the period of extended operation	• LRA Section B.2.1.3 • Response to RAI 3.6-3(a) dated December 9, 2004 • Response to follow-up RAI 3.6-3 dated January 18, 2005
4. ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program	Revise implementing documents to include license renewal reference(s).	A.1.4	Prior to the period of extended operation	• LRA Section B.2.1.4

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
5. Chemistry Control Program	Revise implementing documents to include license renewal reference(s).	A.1.5	Prior to the period of extended operation	• LRA Section B.2.1.5
6. Reactor Head Closure Studs Program	Revise implementing documents to include license renewal reference(s).	A.1.6	Prior to the period of extended operation	• LRA Section B.2.1.6
7. Boiling Water Reactor Vessel Inside Diameter Attachment Welds Program	Revise implementing documents to include license renewal reference(s).	A.1.7	Prior to the period of extended operation	• LRA Section B.2.1.7
8. Boiling Water Reactor Feedwater Nozzle Program	Revise implementing documents to include license renewal reference(s).	A.1.8	Prior to the period of extended operation	• LRA Section B.2.1.8
9. Boiling Water Reactor Control Rod Drive Return Line Nozzle Program	Revise implementing documents to include license renewal reference(s).	A.1.9	Prior to the period of extended operation	• LRA Section B.2.1.9
10 Boiling Water Reactor Stress Corrosion Cracking Program	Revise implementing documents to include license renewal reference(s).	A.1.10	Prior to the period of extended operation	• LRA Section B.2.1.10
11. Boiling Water Reactor Penetrations Program	Revise implementing documents to include license renewal reference(s).	A.1.11	Prior to the period of extended operation	• LRA Section B.2.1.11 • Enclosure 1 of TVA letter dated September 14, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
12. Boiling Water Reactor Vessel Internals Program	Revise implementing documents to include license renewal reference(s).	A.1.12	Prior to the period of extended operation	• LRA Section B.2.1.12
	Inspect the top guide beams		Prior to the period of extended operation	• Response to NRC Question (3) dated May 25, 2005
	Establish an aging management program for the steam dryers.		Two years before the first BFN unit enters the period of extended operation	• Response to RAI 3.1-1 dated January 31, 2005
	Enhance the Reactor Pressure Vessel Internals Inspection (RPVII) Units 1, 2, and 3 procedure to require visual inspection of the Access Hole Covers (AHCs) and inspection of the AHC welds.		Two years before the first BFN unit enters the period of extended operation	• Response to RAI B.2.1.12-1(C) dated January 31, 2005 • Response to NRC Question (7) dated May 25, 2005
	Implement the inspection of weld TS-2 (BWRVIP-41).		When inspection technique for weld TS-2 being developed by the BWRVIP Inspection Committee is available.	• Response to Question (12) dated May 25, 2005
13. Flow-Accelerated Corrosion Program	Revise implementing documents to include license renewal reference(s).	A.1.14	Prior to the period of extended operation	• LRA Section B.2.1.15
14. Bolting Integrity Program	Revise implementing documents to include license renewal reference(s).	A.1.15	Prior to the period of extended operation	• LRA Section B.2.1.16

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
15. Open-Cycle Cooling Water System Program	Revise implementing documents to include license renewal reference(s).	A.1.16	Prior to the period of extended operation	• LRA Section B.2.1.17
16. Closed-Cycle Cooling Water System Program	Revise implementing documents to include license renewal reference(s).	A.1.17	Prior to the period of extended operation	• LRA Section B.2.1.18
17. Inspection of Overhead Heavy Load and Light Load Handling Systems Program	Revise implementing documents to include license renewal reference(s).	A.1.18	Prior to the period of extended operation	• LRA Section B.2.1.20
18. Compressed Air Monitoring Program	Revise implementing documents to: • Include license renewal reference(s). • Incorporate guidelines in ASME OM-S/G-2000, Part 17; ANSI/ISA-S7.0.01-1996; and EPRI TR 108147	A.1.19	Prior to the period of extended operation	• LRA Section B.2.1.21
19. BWR Reactor Water Cleanup System Program	Revise implementing documents to include license renewal reference(s).	A.1.20	Prior to the period of extended operation	• LRA Section B.2.1.22
20. Fire Protection Program	Revise implementing documents to include license renewal reference(s).	A.1.21	Prior to the period of extended operation	• LRA Section B.2.1.23

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
21. Fire Water System Program	Revise implementing documents to: • Include license renewal reference(s). • Perform flow tests or non-intrusive examinations to identify evidence of loss of material due to corrosion.	A.1.22	Prior to the period of extended operation	• LRA Section B.2.1.24
	Perform sprinkler head inspections to ensure signs of degradation, such as corrosion, are detected in a timely manner.		Prior to exceeding the 50-year service life for any sprinkler	• LRA Section B.2.1.24
22. Aboveground Carbon Steel Tanks Program	Revise implementing documents to include license renewal reference(s).	A.1.23	Prior to the period of extended operation	• LRA Section B.2.1.26
23. Fuel Oil Chemistry Program	Revise implementing documents to include license renewal reference(s).	A.1.24	Prior to the period of extended operation	• LRA Section B.2.1.27 • Enclosure 1 of TVA letter dated September 14, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
24. Reactor Vessel Surveillance Program	Revise implementing documents to include license renewal reference(s).	A.1.25	Prior to the period of extended operation	• LRA Section B.2.1.28
	Enhance the Integrated Surveillance Program (ISP) per proposed BWRVIP-116.		Prior to the period of extended operation	• LRA Section B.2.1.28
	If the ISP is not approved two years prior to the commencement of the license renewal period, a plant-specific surveillance program for each BFN unit will be submitted to the NRC.		Two years prior to the commencement of the license renewal period	• Response to RAI B.2.1.28-1(A) dated January 31, 2005 • Response to Question (9) dated May 25, 2005
	Maintain Unit 1 and Unit 3 surveillance capsules (standby capsules) available to the ISP.		Unit 3 is ongoing Unit 1 will commence at restart	• Response to Question (10) dated May 25, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
25. One-Time Inspection Program	Develop and implement new program.	A.1.26	Prior to the period of extended operation	• LRA Section B.2.1.29
	Develop and submit procedure for NRC review.		At least two years prior to the expiration of the current operating license	• Response to Proposed Unresolved Item 3.0-4 LP dated May 27, 2005
	Perform a one-time inspection of the ASME equivalent Class MC supports in a submerged environment of the Units 2 and 3 Torus.		Prior to the period of extended operation	• Response to RAI B.2.1.33-2 dated January 18, 2005
	Perform a one-time inspection of the in-scope submerged concrete in one individual CCW pump bay of the Intake Pumping Station.		Prior to the period of extended operation	• Response to Question 359 dated October 8, 2004 • Response to RAI 3.5-16 dated April 5, 2005
	Perform ultrasonic thickness measurements of tank bottoms for those tanks specified in the Fuel Oil Chemistry Program (B.2.1.27) and the Aboveground Carbon Steel Tanks Program (B.2.1.26).		Prior to the period of extended operation	• Response to RAI 7.1.19-1 dated May 25, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
26. Selective Leaching of Materials Program	Develop and implement program.	A.1.27	Prior to the period of extended operation	• LRA Section B.2.1.30
27. Buried Piping and Tanks Inspection Program	Revise implementing documents to include license renewal reference(s).	A.1.28	Prior to the period of extended operation	• LRA Section B.2.1.31
	Add a trigger to the excavation permit document to require notification of engineering to perform a piping inspection when piping is excavated.		Complete	• NRC Inspection Report dated January 27, 2005
	Determine (via engineering evaluation) if sufficient inspections have been performed to draw conclusion regarding ability of underground coating to protect piping. If required, conduct a focused inspection to draw conclusion concerning the coating.		Within ten years after entering the period of extended operation	• Response to RAI 7.1.22-1 dated May 25, 2005
	Revise implementing documents to inspect buried piping when it is excavated.		Complete	• Response to RAI 7.1.22-1 dated May 25, 2005
28. ASME Section XI Subsection IWE Program	Revise implementing documents to include license renewal reference(s).	A.1.29	Prior to the period of extended operation	• LRA Section B.2.1.32

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
29. ASME Section XI Subsection IWF Program	Revise implementing documents to include license renewal reference(s).	A.1.30	Prior to the period of extended operation	• LRA Section B.2.1.33
	Enhance program to manage the aging effects of ASME equivalent Class MC supports.		Prior to the period of extended operation	• Response to Follow-up RAI B.2.1.33-1 dated May 31, 2005
30. 10 CFR 50 Appendix J Program	Revise implementing documents to include license renewal reference(s).	A.1.31	Prior to the period of extended operation	• LRA Section B.2.1.34
31. Masonry Wall Program	Revise implementing documents to include license renewal reference(s).	A.1.32	Prior to the period of extended operation	• LRA Section B.2.1.35
	Revise implementing procedures to clearly identify structures with masonry walls within scope and to clarify qualification requirements for personnel who perform masonry wall walkdowns.		Prior to the period of extended operation	• LRA Section B.2.1.35

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
32. Structures Monitoring Program	Revise implementing documents to include license renewal reference(s).	A.1.33	Prior to the period of extended operation	• LRA Section B.2.1.36
	Enhance procedures implementing the10 CFR 50.65 Maintenance Rule Program to identify all structures and structural components within scope.		Prior to the period of extended operation	• LRA Section B.2.1.36 • Response to GALL audit Question 173 dated October 8, 2004 • Response to GALL audit Question 357 dated October 8, 2004
	Enhance procedures implementing the 10 CFR 50.65 Maintenance Rule program sampling approach to include examinations of below-grade concrete when excavated.		Prior to the period of extended operation	• LRA Section B.2.1.36 • Response to GALL audit Question 285 dated October 8, 2004
	Enhance procedures implementing the 10 CFR 50.65 Maintenance Rule program to include the guidance provided in ACI 349.3R-96 Chapter 7.		Prior to the period of extended operation	• LRA Section B.2.1.36
	Enhance LCEI-CI-C9, Attachment 1, “Buried Piping Inspection Checklist,” to include “Mechanical Penetration” as an inspection attribute.		Prior to entering the period of extended operation	• Response to GALL audit Question 285 dated October 8, 2004

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
33. Inspection of Water-Control Structures Program	Revise implementing documents to include license renewal reference(s).	A.1.34	Prior to the period of extended operation	• LRA Section B.2.1.37
	Revise implementing documents to identify required structures and structural components within the scope of license renewal.		Prior to the period of extended operation	• LRA Section B.2.1.37
	Revise implementing documents to include special inspections following the occurrence of large floods, earthquakes, tornadoes, and intense rainfall.		Prior to the period of extended operation	• LRA Section B.2.1.37
	Implement periodic monitoring of the raw service water in close proximity to the Intake Pumping Station for the requirements of an aggressive environment.		Prior to the period of extended operation	• Response to RAI 3.5-16 dated April 5, 2005
34. Environmental Qualification Program	Revise implementing documents to include license renewal reference(s).	A.1.35	Prior to the period of extended operation	• LRA Section B.3.1
35. Fatigue Monitoring Program	Implement enhanced Fatigue Monitoring Program using the EPRI-licensed FatiguePro® cycle counting and fatigue usage tracking computer program.	A.1.36	Prior to the period of extended operation	• LRA Section B.3.2
36. Systems Monitoring Program	Revise implementing documents to include license renewal reference(s).	A.2.1	Prior to the period of extended operation	• LRA Section B.2.1.39 • Enclosure 1 of TVA letter dated September 14, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
37. Bus Inspection Program	Develop and implement new program.	A.2.2	Prior to the period of extended operation	• LRA Section B.2.1.40 • Response to RAI 3.6-4 dated December 9, 2004
38. Diesel Starting Air Program	Revise implementing documents to include license renewal reference(s).	A.2.3	Prior to the period of extended operation	• LRA Section B.2.1.41
39. Time-Limited Aging Analysis: Reactor Vessel Thermal Limit Analyses: Operating Pressure-Temperature Limits (P-T)	Develop and submit revised P-T limits to the NRC for approval.	A.3.1.5	Prior to the period of extended operation	• LRA Section A.3.1.5 • LRA Section 4.2.5
40. Time-Limited Aging Analysis: Environmental Qualification of Electrical Equipment	Revise existing EQ program to cover the extended period of operation.	A.3.3	Prior to the period of extended operation	• LRA Section A.3.3 • LRA Section 4.4
41. Time-Limited Aging Analysis: Other Plant Specific Time-Limited Aging Analysis: Emergency Equipment Cooling Water Weld Flaw Evaluation	Implement an administrative tracking system to ensure limiting number of fatigue cycles will not be exceeded at the select EECW locations.	A.3.5.7	Prior to the period of extended operation	• LRA Section A.3.5.7 • Response to RAI 4.7.8 dated March 2, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
42. RAI 2.1-2,A-3	Identify additional piping segments and supports/equivalent anchors to be placed in scope.	N/A	Complete	• Response to RAI 2.1-2,A-3 dated September 3, 2004 • TVA response dated February 28, 2005
43. RAI 2.1-2,B	Implement Unit 1, 2, and 3 DCNs to qualify twelve temperature switches in the Turbine Building.	N/A	Prior to the period of extended operation	• Response to RAI 2.1-2,B dated September 3, 2004
44. RAI 2.1-2,C RHRSW tunnel	Include 24-inch Raw Cooling Water discharge piping located in the RHRSW tunnel in scope of license renewal.	N/A	Complete	• Response to RAI 2.1-2,C RHRSW Tunnel dated September 3, 2004 • TVA response dated January 31, 2005
45. RAI 2.1-2,C Intake Pumping Station	Revise 10 CFR 54.4(a)(2) Scoping Methodology document to address components located in the lower compartments of the Intake Pumping Station.	N/A	Prior to next annual update	• Response to RAI 2.1-2,C Intake Pumping Station dated September 3, 2004
46. Open Item OI 2.4-3	Perform one time confirmatory ultrasonic thickness (UT) measurements on a portion of the cylindrical section of the drywell on Units 2 and 3.	N/A	Prior to the period of extended operation	• Enclosures 1 and 9 of TVA letter dated November 16, 2005

TABLE 1: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (NON-UNIT SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
47. Open Item OI 4.7.7	Perform a BFN plant-specific analysis consistent with BWRVIP-25 to demonstrate that the core plate hold-down bolts can withstand required loads, considering the effects of stress relaxation until the end of the period of extended operation. Take appropriate corrective action if the analysis does not satisfy the specified criteria. Submit the analysis or the corrective action taken to resolve the core plate hold-down bolt issue to the NRC for review.	N/A	Two years prior to the period of extended operation	• Enclosures 3 and 9 of TVA letter dated November 16, 2005
48. Open Item from AMP Inspection on Inspection of RHRSW Piping	Perform a confirmatory inspection of the RHRSW pump pit supply piping. Include instructions in the CCW pump pit Preventive Maintenance Program to periodically inspect the sluice gate valves. Perform a confirmatory inspection of the seismic restraints in the RHRSW pump pit.	N/A	Prior to the period of extended operation	• Enclosures 4 and 9 of TVA letter dated November 16, 2005

TABLE 2: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (UNIT 1 SPECIFIC)

NOTE: This Table does not contain all of the same Item Numbers as contained in Table 1. While there is a one-to-one correlation of items with the same number, the same Item Numbers are not in both tables as explained below:

- For Item Numbers 1. through 49., only those Item Numbers that have a Unit 1 specific commitment are included in this table.
- Item Number 63. applies only to Unit 1.

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
2. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits Program	Include Unit 1 High-Range Radiation Monitoring cables in the Environmental Qualification (EQ) Program.	A.1.2	Prior to Unit 1 restart	• Response to GALL audit Question 169 dated October 8, 2004
5. Chemistry Control Program	Include Unit 1 in the program.	A.1.5	Prior to Unit 1 restart	• LRA Section B.2.1.5
7. Boiling Water Reactor Vessel Inside Diameter Attachment Welds Program	Include Unit 1 in the program.	A.1.7	Prior to Unit 1 restart	• LRA Section B.2.1.7
8. Boiling Water Reactor Feedwater Nozzle Program	Upgrade Unit 1 operating procedures to decrease the magnitude and frequency of feedwater temperature fluctuations.	A.1.8	Prior to Unit 1 restart	• LRA Section B.2.1.8
10. Boiling Water Reactor Stress Corrosion Cracking Program	Include Unit 1 in the program.	A.1.10	Prior to Unit 1 restart	• LRA Section B.2.1.10 • Response to GALL audit Question 181 dated October 8, 2004

TABLE 2: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (UNIT 1 SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
11. Boiling Water Reactor Penetrations Program	Include Unit 1 in the program.	A.1.11	Prior to Unit 1 restart	• LRA Section B.2.1.11 • Response to GALL audit Question 194 dated October 8, 2004
12. Boiling Water Reactor Vessel Internals Program	Include Unit 1 in the program.	A.1.12	Prior to Unit 1 restart	• LRA Section B.2.1.12 • Response to Question (4b) dated May 25, 2005
13. Flow-Accelerated Corrosion Program	Include Unit 1 in the program.	A.1.14	Prior to Unit 1 restart	• LRA Section B.2.1.15 • Response to GALL audit Question 144 dated October 8, 2004
15. Open-Cycle Cooling Water System Program	Include Unit 1 in the program.	A.1.16	Prior to Unit 1 restart	• LRA Section B.2.1.17 • Response to GALL audit Question 144 dated October 8, 2004
16. Closed-Cycle Cooling Water System Program	Include Unit 1 in the program.	A.1.17	Prior to Unit 1 restart	• LRA Section B.2.1.18 • Response to GALL audit Question 144 dated October 8, 2004
18. Compressed Air Monitoring Program	Include Unit 1 in the program.	A.1.19	Prior to Unit 1 restart	• LRA Section B.2.1.21

TABLE 2: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (UNIT 1 SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
19. BWR Reactor Water Cleanup System Program	Include Unit 1 in the program.	A.1.20	Prior to Unit 1 restart	• LRA Section B.2.1.22 • LRA Section F.13
20. Fire Protection Program	Update the Fire Protection Report and to incorporate Unit 1 as an operating unit. Fully implement the program on Unit 1.	A.1.21	Prior to Unit 1 restart	• LRA Section B.2.1.23
21. Fire Water System Program	Update the Fire Protection Report and procedures to incorporate Unit 1 as an operating unit. Fully implement the program on Unit 1.	A.1.22	Prior to Unit 1 restart	• LRA Section B.2.1.24
24. Reactor Vessel Surveillance Program	Either include Unit 1 within the BWRVIP ISP, or submit for NRC approval a plant specific surveillance program that meets the requirements of 10 CFR 50 Appendix H for the period of extended operation.	A.1.25	Prior to the period of extended operation	• LRA Section B.2.1.28
	Ensure BWRVIP-86-A and BWRVIP-116 are revised to incorporate Unit 1, and submit to the NRC a license amendment request to implement the ISP for site-specific use for Unit 1.		Prior to the period of extended operation	• Response to RAI B.2.1.28-1 dated January 31, 2005
25. One-Time Inspection Program	Perform a one-time inspection of the ASME equivalent Class MC supports in a submerged environment of the Unit 1 Torus.	A.1.26	Prior to Unit 1 restart	• Response to RAI B.2.1.33-2(b) dated January 18, 2005
34. Environmental Qualification Program	Include Unit 1 in the program.	A.1.35	Prior to Unit 1 restart	• LRA Section B.3.1

TABLE 2: BFN COMMITMENT LIST ASSOCIATED WITH LRA APPENDIX A AGING MANAGEMENT PROGRAMS AND TLAAs (UNIT 1 SPECIFIC)

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
47. Open Item OI 2.4-3	Perform one time confirmatory UT measurements on the drywell vertical cylindrical area immediately below the drywell flange	N/A	Prior to Unit 1 restart	• Enclosures 1 and 9 or TVA letter dated November 16, 2005
49. Unit 1 Periodic Inspection Program	Develop and implement new program.	A.2.4	Prior to the period of extended operation	• Response to Proposed Unresolved Items 3.0-2 LP (1 & 2) and 3.0-3 LP dated May 27, 2005 • Enclosure 1 of TVA letter dated September 14, 2005
	Develop and submit implementing procedure(s) for NRC review.		At least two years prior to the period of extended operation	• Response to Proposed Unresolved Items 3.0-4 LP dated May 27, 2005
63. Response to NRC Questions Concerning RPV Internals	Replace all BFN Unit 1 dry tubes.	N/A	Prior to Unit 1 restart	• Response to Question (8) dated May 25, 2005
	Perform MSIP for Unit 1 Control Rod Drive Return Line Cap.		Prior to Unit 1 restart	• Response to Question (6) dated May 25, 2005
	Change the Unit 1 AHCs to bolted design.		Prior to Unit 1 restart	• Response to NRC Question (7) dated May 25, 2005

TABLE 3: UNIT 1 RESTART COMMITMENTS THAT ARE DISCUSSED IN LRA APPENDIX F

NOTE: See Note at the beginning of Table 2

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
50. Appendix F.1	Evaluate and modify, as required, main steam leakage path piping to ensure structural integrity.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
51. Appendix F.2	Implement Containment Atmosphere Dilution System modification.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
52. Appendix F.3	Revise Fire Protection Program to ensure compliance with 10 CFR 50 Appendix R. Revise Fire Protection Report per Unit 1 License Condition 2.C.13.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
53. Appendix F.4	Implement Environmental Qualification Program.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
54. Appendix F.5	Address GL 88-01, and make necessary plant modifications.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
55. Appendix F.6	BWRVIP Programs used for Units 2 and 3 will be used for Unit 1.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
56. Appendix F.7	Install ATWS features.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005

TABLE 3: UNIT 1 RESTART COMMITMENTS THAT ARE DISCUSSED IN LRA APPENDIX F

Item Number/Title	Commitment	LRA Appendix A (UFSAR)	Implementation Schedule	Source
57. Appendix F.8	Remove Reactor Vessel Head Spray piping in drywell, and seal the primary containment penetrations	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
58. Appendix F.9	Implement the Hardened Wetwell Vent modification.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
59. Appendix F.10	Cap Service Air and Demineralized Water Primary Containment Penetrations.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
60. Appendix F.11	Modify Auxiliary Decay Heat Removal System to serve Unit 1.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
61. Appendix F.12	Fully implement the Maintenance Rule Unit 1's temporary exemption ceases to be effective.	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005
62. Appendix F.13	Replace RWCU piping outside of primary containment with IGSCC resistant piping. Implement actions requested in GL 89-10 for RWCU	N/A	Prior to Unit 1 restart	• LRA Appendix F • TVA response dated January 31, 2005

APPENDIX B

CHRONOLOGY

This appendix contains a chronological listing of the routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and the Tennessee Valley Authority (TVA), and other correspondence regarding the NRC staff's reviews of the Browns Ferry Nuclear (BFN), Units 1, 2 and 3 (under Docket Numbers 50-259, 50-260 and 50-296) license renewal application (LRA).

July 12, 1984	TVA letter to NRC, in regards to NUREG 0737, Item II.K.3.28, "Qualification of ADS Accumulators"
July 24, 1985	NRC letter to TVA, "NUREG 0737, Item II.K.3.28, Qualification of ADS Accumulators"
March 1, 1988	TVA letter, R. Gridley to NRC, "Browns Ferry Nuclear Plant (BFN) - Anticipated Transients Without Scram (ATWS) Rule (10 CFR 50.62) - Plant Specific Design"
August 1, 1988	TVA letter to NRC, "Browns Ferry Nuclear Plant (BFN) - Response to Bulletin (sic) 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, dated January 25, 1988"
October 24, 1988	TVA letter, S. A. White to NRC, "Browns Ferry Nuclear Plan (BFN) Nuclear Performance Plan, Revision 2"
December 8, 1988	NRC letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2 and 3 - Appendix R Safe Shutdown System Analysis"
January 22, 1989	NRC letter to TVA, "Compliance with Rule 10 CFR 50.62 Relating to Alternate Rod Injection and Reactor Pump Trip Systems"
January 26, 1989	NRC letter to TVA, "Technical Specifications on Anticipated Transients Without Scram (ATWS) - Recirculation Pump Trip (RPT), Browns Ferry Nuclear Plants, Units 1, 2, and 3" (Accession No. ML020020476)
October 30, 1989	NRC letter to All Operating Licensees with Mark I Containments, "Installation of a Hardened Wetwell Vent (Generic Letter 89-16)," dated September 1, 1989. 2. TVA letter, M. J. Ray to NRC, "Response to Generic Letter 89-16, Installation of Hardened Wetwell Vent"
November 2, 1989	NRC letter to TVA, "Supplemental Safety Evaluation on Post-Fire Safe Shutdown Systems and Final Review of the National Fire Protection Association Code Deviations - Browns Ferry Nuclear Plant, Unit 2"

November 29, 1990	TVA letter, E. G. Wallace to NRC, “Browns Ferry Nuclear Plant (BFN) - Anticipated Transient Without Scram (ATWS) Response to NRC Followup Items Received During ATWS Inspection
January 23, 1991	NRC letter to TVA, “NUREG 1232, Volume 3, Supplement 2 Browns Ferry, Unit 2"
March 6, 1991	NRC letter to TVA, “Issuance of Amendment” (Accession No. ML020090226)
May 5, 1992	NRC letter to TVA, “Request for Additional Information to Review Compliance with NUREG 0737, Item II.E.4.2 and 10 CFR 50, Appendix J
December 28, 1992	TVA letter to NRC, “Browns Ferry Nuclear Plant – Unit 3 - Supplemental Response to Generic Letter (GL) 88-01, NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping”
March 31, 1993	NRC letter to TVA, “Fire Protection Program - Browns Ferry Nuclear Plant Units 1, 2 and 3"
December 3, 1993	NRC letter to TVA, “Browns Ferry Nuclear Plant Unit 3 - Safety Evaluation of Supplemental Response to Generic Letter 88-01"
January 2, 1995	NRC letter to TVA, “Browns Ferry Nuclear Plant Units 1 and 3, NUREG-0737, Item II.E.4.2, Containment Isolation Dependability”
November 2, 1995	NRC letter to TVA, “Safety Evaluation of Post-Fire Safe Shutdown Capability and Issuance of Technical Specification Amendments for the Browns Ferry Nuclear Plant Units 1, 2, and 3" (Accession No. ML020040025)
April 25, 1997	BWRVIP letter, C. Terry to B. Sheron (NRC), “BWR Utility Commitments to the BWRVIP”
August 9, 1999	NRC letter to TVA, ”Issuance of Temporary Partial Exemption from 10 CFR 50.65, Browns Ferry Nuclear Plant, Unit 1” (Accession No. ML020040329)
November 25, 2002	NRC Meeting Summary, S.T. Hoffman, “Summary Of Meeting to Discuss Planned License Renewal Application” (Accession No. ML023300013)
June 2, 2003	NRC Meeting Summary, S.T. Hoffman, Summary Of Meeting To Discuss Planned License Renewal Application (Accession No. ML031540295)
October 30, 2003	NRC Meeting Summary, S.T. Hoffman, “Summary Of Meeting to Discuss Browns Ferry Units 1, 2, and 3 Planned License Renewal Application” (Accession No. ML033080369)

December 3, 2003	NRC letter to TVA, “Browns Ferry Nuclear Plant Unit 3 - Safety Evaluation of Supplemental Response to Generic Letter 88-01"
December 31, 2003	Letter from Mr. Mark. J. Burzynski, Tennessee Valley Authority (TVA) to the NRC, submitting the application for the renewal of the operating Licenses for Browns Ferry Nuclear Units 1,2, and 3 (Accession No. ML040060361)
January 7, 2004	Letter from P.T.Kuo, NRC, to J.A.Scalice,TVA forwarding the Notice of Receipt and Availability of the application for the renewal of the operating license for the BFN Units 1,2 and 3 (Accession No. ML040090370)
February 19, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC- Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - January 28, 2004 Meeting Follow-Up - Additional Information - Supplemental Information - Unit 1 Wet Lay-Up (Accession No. ML040510241)
March 4, 2004	Letter from P.T.Kuo, NRC to J. A. Scalice, TVA indicating acceptability and sufficiency for docketing and opportunity for a hearing regarding the application from Tennessee Valley Authority for renewal of the operating licenses for the BFN, units 1, 2, and 3 (Accession No. ML040650206)
March 25, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC stating use of the BFN license renewal boundary drawings to obtain scoping results (Accession No. ML040860596)
March 31, 2004	Letter from P.T.Kuo, NRC, to J.A.Scalice,TVA forwarding the review schedule for application for renewal of the operating licenses for the BFN Units 1,2 and 3 (Accession No. ML040910016)
May 4, 2004	In a memorandum (signed by Jimi Yerokum), NRC summarized the April 7, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff’s review of the LRA. (Accession No. ML041310015)
May 6, 2004	In a memorandum (signed by Jimi Yerokum), NRC summarized the March 24, 2004 and March 30, 2004 teleconferences between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff’s review of the LRA. (Accession No. ML041310029)
May 10, 2004	In a memorandum (signed by Jimi Yerokum), NRC summarized the April 14, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff’s review of the LRA. (Accession No. ML041310206)

May 27, 2004	Browns Ferry Nuclear Plant (BFN) Units 1, 2, and 3 - March 30-31, 2004 meeting follow-up-additional information for License Renewal Environmental Review
May 28, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC updating the LRA application sections 4.2 and 4.3 to reflect extended power uprate conditions (Accession No. ML041550393)
June 15, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the May 19, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML041700550)
June 16, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the April 21, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML041700505)
June 16, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the May 27, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) concerning activities on BFN units 1, 2 and 3 LRA. (Accession No. ML041700523)
June 18, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the May 5, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML041700572)
June 23, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on section 3.5 of the LRA. (Accession No. ML041760076)
June 28, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the May 27, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML041810168)
July 7, 2005	Response to Request for Additional Information (RAI) regarding severe accident mitigation alternatives for Browns Ferry Nuclear Plant, Units 1, 2, and 3

July 9, 2004	TVA letter to NRC, “Browns Ferry Nuclear Plant (BFN) Unit 1 - Technical Specification (TS) 436 - Increased Main Steam Isolation Valve (MSIV) Leakage Rate Limits and Exemption from 10 CFR 50, Appendix J” (Accession No. ML041980222)
July 10, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the June 16, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff’s review of the LRA. (Accession No. ML041950508)
July 19, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the April 28, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff’s review of the LRA. (Accession No. ML042010388)
July 19, 2004	Letter from Mr. M.J.Burzynski, Tennessee Valley Authority (TVA) to the NRC regarding lay-up effects of Unit 1 Structures and Component Supports (Accession No. ML042040231)
July 21, 2004	TVA letter, T. E. Abney to NRC, “Browns Ferry Nuclear Plant (BFN) Unit 1 – Supplemental Response to Generic Letter 88-01, NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping” (Accession No. ML042040274)
July 28, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the July 1, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff’s review of the LRA. (Accession No. ML042110485)
July 30, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on section 2.1 of the LRA. (Accession No. ML042120186)
August 3, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC detailed explanation of how the LRA application Bounds the BFN extended power uprate (EPU) submittals (Accession No. ML042180449)
August 5, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC update of application sections 4.2 and 4.3 to reflect extended power uprate conditions –supplemental information (Accession No. ML042220285)

August 23, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on sections 3.1, 3.2, 3.3, and 3.4 of the LRA. (Accession NO. ML042360590)
August 23, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on sections 2.3, 3.3, 4.4, B.2.0 of the LRA (Accession NO. ML042360762)
August 23, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the July 28, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML042390497)
August 26, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the July 24, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML042400550)
August 31, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the July 12, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML042450211)
August 31, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on sections 2.1, 2.2, and 2.3 of the LRA. (Accession No. ML042450260)
September 3, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC NRC scoping and screening audit – request for additional information (RAI) (Accession No. ML042520374)
September 16, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the August 19, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML042600522)
September 27, 2004	NRC letter to TVA, "Browns Ferry Nuclear Plant, Units 1, 2, and 3 — Issuance of Amendments Regarding Full- Scope Implementation of Alternative Source Term" (Accession No. ML042730028)

September 30, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC NRC scoping and screening audit –request for additional information (Accession No. ML042750259)
October 6, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on 2.3.1, 2.3.2, and 2.3.3 of the LRA. (Accession No. ML042860015)
October 8, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC NRC scoping and screening audit –request for additional information (Accession No. ML042870422)
October 8, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on section 2.3 of the LRA. (Accession No. ML042860051)
October 8, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC NRC scoping and screening staff audit at BFN – request for additional information (Accession No. ML042870428)
October 8, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on sections 2.3.2 and 2.3.3 of the LRA (Accessiion No. ML042860066)
October 12, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on section 3.3 of the LRA (Accession No. ML042860133)
October 15, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the September 15, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML042920201)
October 18, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC NRC scoping and screening audit – request for additional information (Accession No. ML042930471)
October 19, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC – request for additional information - Sections 2.1, 2.2, and 2.3, related to the Scoping and Screening: Mechanical Systems (Accession No. ML042930931)

October 21, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the September 22, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML042990519)
October 22, 2004	In a memorandum (signed by Yoira Diaz-Sanabria), NRC summarized the August 18, 2004 teleconference between the NRC staff and Tennessee Valley Authority (TVA) regarding draft Request for Additional Information (D-RAI) concerning the staff's review of the LRA. (Accession No. ML043000040)
October 25, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - request for additional information on Appendix F (Accession No. ML043000149)
October 28, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Fire Protection Section Verbal Request on October 20, 2004 - Response to NRC Request for Additional Information (Accession No. ML043030434)
November 1, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on section 2.5 of the LRA (Accession No. ML043060492)
November 3, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 License Renewal Application – Heating, Ventilation, and Cooling (HVAC) Systems Sections 2.3.2 and 2.3.3 – Request for Additional Information (RAI) - (Accession No. ML043090545)
November 3, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) – Units 1, 2, and 3 License Renewal Application – Reactor Systems Section 2.3.1, 2.3.2, and 2.3.3 – Request for Additional Information (Accession No. ML043100588)
November 3, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Auxiliary Systems Section 3.3 - Response to NRC Request for Additional Information (Accession No. ML043090343)
November 4, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on sections 3.1.2.4, B.2.1.13, and 4.7.8 of the LRA (Accession No. ML043090573)

November 4, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on section 3.6 of the LRA (Accession No. ML043090577)
November 18, 2004	Letter from Ram Subbaratnam, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Sections 3.2 and 3.4 of the LRA (Accession No. ML043270655)
December 1, 2004	Letter from Ram Subbaratnam, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Sections 3.1 of the LRA (Accession No. ML043360401)
December 1, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – electrical and instrument and control systems (I&C) systems section 2.5- Response to NRC Request for Additional Information (Accession No. ML043370173)
December 3, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Heating, Ventilation, and Cooling (HVAC) Systems Sections 2.3.2 and 2.3.3 - Response to NRC Request for Additional Information (Accession No. ML043380353)
December 7, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Chemistry Control Program, Section B.2.1.5 of the LRA (Accession No. ML043490336)
December 9, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Mechanical Systems Sections 3.1.2.4, B.2.1.13, and 4.7.8- Response to NRC Request for Additional Information (Accession No. ML043440080)
December 9, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Electrical and Instrument and Control Systems (I&C) Systems Section 3.6- Response to NRC Request for Additional Information (Accession No. ML043440226)
December 9, 2004	Letter from Ram Subbaratnam, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding status of staff review of the Browns Ferry Nuclear Plant Units 1, 2, and 3 License Renewal Application (Accession No. ML043490470)

December 10, 2004	Letter from Ram Subbaratnam, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2, and 3 license renewal application on Section 3.5 and B.2.1.34 of the LRA (Accession No. ML043500140)
December 13, 2004	Letter from Ram Subbaratnam, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on ASME Section XI Subsection IWF Program, Section B.2.1.33 of the LRA (Accession No. ML043500210)
December 14, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application Section 4.7.1 of the LRA (Accession No. ML043500508)
December 16, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Mechanical Systems Sections 3.2 and 3.4 - Response to NRC Request for Additional Information (Accession No. ML043520395)
December 16, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application Section 3.0 of the LRA (Accession No. ML043560502)
December 20, 2004	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 4.4-2 Mechanical and Environmental Qualifications - Response to NRC Request for Additional Information (Accession No. ML043550381)
December 20, 2004	Letter from Yoira Diaz-Sanabria, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Section 2.4 of the LRA (Accession No. ML043560382)
January 6, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections B.2.1.5 Chemistry Control Program - Response to NRC Request for Additional Information (Accession No. ML050070179)
January 7, 2005	Letter from Mr. M. D. Skaggs, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application - Meeting Summary and Plant Visit (ML050100180)

January 12, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 4.7.1 Reactor Building Crane Load Cycle - Response to NRC Request for Additional Information (Accession No. ML050130333)
January 18, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections B.2.1.33 ASME Section XI Subsection IWF Program - Response to NRC Request for Additional Information (Accession No. ML050180505)
January 18, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 2.5 and 3.6 Electrical and Instrument and Control - Response to NRC Request for Additional Information (Accession No. ML050180537)
January 20, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 3.1 Aging of Mechanical Systems During the Extended Outage - Response to NRC Request for Additional Information (Accession No. ML050210334)
January 24, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 2.4 - Response to NRC Request for Additional Information (Accession No. ML050250264)
January 25, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 4.4 - Response to NRC Request for Additional Information (Accession No. ML050260327)
January 27, 2005	Letter from Harold O. Christensen, NRC, to K. W. Singer, Tennessee Valley Authority (TVA) regarding Browns Ferry Nuclear Plant - Inspection Report 05000259/2004012, 05000260/2004012, and 05000296/2004012 (Accession No. ML050270022)
January 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Annual Update (Accession No. ML050310428)
January 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – The Integration of Unit 1 Restart and License Renewal Activities. Response to NRC Request for Additional Information (Accession No. ML050320137)

January 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 3.1, 4.2, and B.2.1 Reactor Vessel and Internals Mechanical Systems - Response to NRC Request for Additional Information (Accession No. ML050320145)
January 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 3.5, 4.7.4, and B.2.1.32 - Response to NRC Request for Additional Information (Accession No. ML050320149)
January 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 3 Unit 1 layup questions - Response to NRC Request for Additional Information (Accession No. ML050320208)
January 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Section 2.1, status of response to RAI 2.1-2, A.3 - Response to NRC Request for Additional Information (Accession No. ML050310442)
February 28, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 2.1, final status of response to RAI 2.1-2, A.3 - Response to NRC Request for Additional Information (Accession No. ML050600274)
February 28, 2005	Browns Ferry Nuclear Plant - Units 1, 2, and 3 License Renewal Application - LRA Section 3.5 - response to NRC request for follow-up question for RAI 3.5-7
March 2, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 2.5 and 4.7.8 - Response to NRC Request for Additional Information (Accession No. ML050620258)
March 3, 2005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Section 4.7.7 of the LRA (Accession No. ML050620592)
March 11, 2005	Letter from Yoira Diaz-Sanabria, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Sections 3.1.2.4 and 3.5 of the LRA (Accession No. ML050700309)

March 11, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 3.3 - Response to NRC Request for Additional Information (Accession No. ML050700463)
March 16, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 4.6.2 T-Quenchers within Reactor Vessel Vents and Drains System - Response to NRC Request for Additional Information (Accession No. ML050760230)
March 16, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 3.1.2.4 and 4.3 Reactor Coolant Pressure Boundary Bolting Clarifications - Response to NRC Request for Additional Information (Accession No. ML050770041)
March 25, 2005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) forwarding request for additional information for the review of the BFN units 1, 2 and 3 license renewal application on Section 2.4 and 3.5 of the LRA (Accession No. ML050840483)
April 5, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 7.2.5.2 ASME Equivalent Supports and Components - Response to NRC Request for Additional Information (Accession No. ML050950189)
April 5, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 3.1.2.4-7 and 3.5-16 AMR Small Bore Piping and Fittings and Submerged Reinforced Concrete - Response to NRC Request for Additional Information (Accession No. ML050950311)
April 8, 2005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) Forwarding Request for Additional Information for the Review of the BFN Units 1, 2, and 3 License Renewal Application on Section 2.3.3.21 (Accession No. ML050980086)
April 14, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 2.4 and 3.5 Radwaste and Service Building - Response to NRC Request for Additional Information (Accession No. ML051040164)
April 19, 005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Trip Report of staff visit to Browns Ferry Nuclear Units 1,2, and 3 on March 28, 29, 2005 (Accession No. ML051090488)

April 28, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 2.3.3.21 Reactor Water Cleanup System - Response to NRC Request for Additional Information (Accession No. ML051190272)
May 12, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 7.2.5.2 ASME Equivalent Supports and Components - Response to NRC Request for Additional Information (Accession No. ML051330038)
May 18, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 2.3 and 2.4 - Response to NRC Request for Additional Information (Accession No. ML051380504)
May 18, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 3.0-9 Unit 1 Layup Program - Response to NRC Request for Additional Information (Accession No. ML051390237)
May 19, 2005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Summary of teleconference with Tennessee Valley Authority (TVA) License Renewal Staff (Accession No. ML051400190)
May 24, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 2.3.3.21 Reactor Water Cleanup System - Response to NRC Request for Additional Information (Accession No. ML051440261)
May 24, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 3.3 Diesel Generator - Response to NRC Request for Additional Information (Accession No. ML051440779)
May 24, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 2.4, 3.5 and 4.7.4 - Response to NRC Request for Additional Information (Accession No. ML051450126)
May 25, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 7.1.19 and 7.1.22 GALL Audit - Response to NRC Request for Additional Information (Accession No. ML051460348)

May 25, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Sections 4.2.6, 4.2.7 and 4.7.6 Reactor Pressure Vessel Internals - Response to NRC Request for Additional Information (Accession No. ML051460354)
May 26, 2005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Summary of teleconference with Tennessee Valley Authority (TVA) License Renewal Staff (Accession No. ML051460418)
May 27, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 3.0 Unit 1 Layup Program - Response to NRC Request for Additional Information (Accession No. ML051470354)
May 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 2.1.2 and 2.3.4.4 Secondary Containment and Main Steam System - Response to NRC Request for Additional Information (Accession No. ML051520081)
May 31, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Section 4.7.7 TLAA Core Plate Relaxation of Bolts - Response to NRC Request for Additional Information (Accession No. ML051520139)
May 31, 2005	Letter from Mr. M D. Skaggs, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application (LRA) - Response to NRC Request for Additional Information (RAI) concerning follow up to RAIs 2.4-3, 3.5-1, 3.5-4, B.2.1.33-1, and B.2.1.36 (Accession No. ML051520084)
May 31, 2005	Letter from M.L. Marshall, Jr., NRC, to K W. Singer, Tennessee Valleu Authority (TVA) - Browns Ferry Nuclear Plant Unit 1 - Safety Evaluation for Relief Request 1-ISI-19 Associated With Reactor Pressure Vessel Circumferential Shell Welds (Accession No. ML051110626)
June 3, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application –Response to NRC Request for Additional Information on Potential Open Item 3.3.2.35-1 (Accession No. ML051540336)
June 9, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Additional Information (RAI) 4.7.3-1-Radiation Dose for Valve Seals (Accession No. ML051610400)

June 9, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Additional Information (RAI) Concerning follow up to RAI 2.3.3.18-1 and follow up to RAI 2.3.3.22-1(Accession No. ML051610592)
June 15, 2005	Letter from Mr. T.E.Abney, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Additional Information (RAI) for the Time-limited Aging Analysis identified in 4.7.2 and 4.7.5 of LRA (Accession No. ML051660547)
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June 22, 2005	Letter from Ram Subbaratnam, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Request for Additional Information on Section 4.7.7 (Accession No. ML051730507)
June 29, 2005	Letter from Mr. W.D. Crouch, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Additional Information (RAI) concerning Follow up to Section 4.7. 7 Time Limited Aging Analysis RAIs (Accession No. ML0519402910)
August 2, 2005	Letter from Yoira Diaz Sanabria, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Summary of teleconference held on July 7, 2005, with Tennessee Valley Authority (TVA) License Renewal Staff, Concerning Information on Browns Ferry Nuclear Plant, Units 1, 2, and 3 License Renewal Application (Accession No. ML052140646)
August 4, 2005	Letter from Mr. W.D. Crouch, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant - Units 1, 2, and 3 License Renewal Application - Description of Unit 1 Periodic Inspection Program (Accession No. ML052170406)
August 9, 2005	Letter from P.T. Kuo, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Safety Evaluation Report (SER) with Open Items Related to the License Renewal of Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Accession No. ML052210459)
August 9, 2005	Letter from Mr. Brian O'Grady, Browns Ferry Nuclear Plant to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Consolidated List of Commitments for License Renewal (Accession No. ML052220070)

September 6, 2005	Letter from Mr. Brian O'Grady, Browns Ferry Nuclear Plant to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application - Response to NRC Request for Additional Information Concerning Follow-Up to RAIs 4.7.7-1 and 4.7.7-2 (Accession No. ML052570462)
September 14, 2005	Letter from Mr. Brian O'Grady, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application - Results of Review of Safety Evaluation Report (SER) with Open Items Related to License Renewal of Browns Ferry Nuclear Plant (Accession No. ML052630075)
October 31, 2005	Letter from Yoira Diaz Sanabria, NRC, to K.W. Singer, Tennessee Valley Authority (TVA) - Summary of teleconference held on October 14 and 18, 2005, with Tennessee Valley Authority (TVA) License Renewal Staff, Concerning Information on Browns Ferry Nuclear Plant, Units 1, 2, and 3 License Renewal Application (Accession No. ML053050358)
November 16, 2005	Letter from Mr. W.D. Crouch, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant - Units 1, 2, and 3 License Renewal Application (LRA) - Supplemental Responses to NRC Requests (Accession No. ML053320331)
November 21, 2005	Letter from Mr. W.D. Crouch, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 License Renewal Application – Response to NRC Request for Clarification for BFN's use of Linde 80 Weld Material on Unit 1. (Accession No. ML053260542)
December 12, 2005	Letter from Mr. W.D. Crouch, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Units 1, 2, and 3 License Renewal Application (LRA) - Response to NRC Request for Clarification of Commitment Tables Contained in the Safety Evaluation Report (Accession No. ML053460417)
December 20, 2005	Letter from Mr. W.D. Crouch, Tennessee Valley Authority (TVA) to the NRC - Browns Ferry Units 1, 2, and 3 License Renewal Application (LRA) - Response to NRC Request for Clarification of Commitment Tables Contained in the Safety Evaluation Report (Accession No. 053560328)

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APPENDIX C
PRINCIPAL CONTRIBUTORS

NAME	RESPONSIBILITY
R. Subbaratnam	Project Manager
Y. Diaz Sanabria	Project Manager
K. Naidu	Quality Assurance
G. Galleti	Quality Assurance
B. Rogers	Quality Assurance
S. Jones	Management Supervision
J. Guo	Plant Systems
F. Akstulewicz	Management Supervision
M. Razzaque	Reactor Systems
R. Dennig	Management Supervision
R. Goel	Plant Systems
J. Raval	Plant Systems
S. Weerakkody	Management Supervision
N. Iqbal	Plant Systems
K. Manoly	Management Supervision
D. Jeng	Civil Engineering
H. Ashar	Civil Engineering
A. Lee	Mechanical Engineering
Y. Li	Mechanical Engineering
S. Bailey	Mechanical Engineering
R. McNally	Mechanical Engineering
M. Hartzman	Mechanical Engineering
R. Jenkins	Management Supervision
A. Pal	Electrical Engineering
G. Cranston	Technical Support
T. Chan	Management Supervision
R. Davis	Materials Engineering
S. Coffin	Management Supervision
M. Mitchell	Management Supervision
B. Elliot	Materials Engineering
G. Cheruvenki	Materials Engineering
L. Lambros	Materials Engineering
L. Lund	Management Supervision
K. Parczewski	Chemical Engineering
A. Hodgdon	Office of General Council

CONTRACTORS

CONTRACTOR	TECHNICAL AREA
Brookhaven National Laboratory	GALL Audit
Information Systems Laboratories	Plant Systems
Legin Group, Inc.	SER Support

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APPENDIX D

REFERENCES

This appendix contains a listing of references used in the preparation of the Safety Evaluation Report prepared during the review of the license renewal application for Browns Ferry Nuclear Plant, Units 1, 2, and 3, Docket Numbers 50-259, 50-260, and 50-296, respectively.

(1) NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” April 2001

(2) NEI 95-10, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule, Revision 3,” August 2001

(3) NUREG-1801, “Generic Aging Lessons Learned Report (GALL),” April 2001

(4) Browns Ferry Nuclear Plant, Units 1, 2, and 3, updated final safety analysis report (UFSAR), Amendment 20.

(5) BWR Vessel and Internals Project Report BWRVIP-25: “BWR Core Plate Inspection and Flaw Evaluation Guidelines,” TR-107284, December 1996.

(6) BWR Vessel and Internals Project Report BWRVIP-99: “Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components,” TR-1003018, December 2001.