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NUREG-2101

Safety Evaluation Report

Related to the License Renewal of Salem Nuclear Generating Station

Docket Numbers 50-272 and 50-311

PSEG Nuclear, LLC

Office of Nuclear Reactor Regulation

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ABSTRACT

This safety evaluation report (SER) documents the technical review of the Salem Nuclear Generating Station, Units 1 and 2, (Salem) license renewal application (LRA) by the U.S. Nuclear Regulatory Commission (NRC) staff (the staff). By letter dated August 18, 2009, PSEG Nuclear, LLC (PSEG or the applicant) submitted the LRA in accordance with Title 10, Part 54, of the *Code of Federal Regulations*, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.” PSEG requests renewal of the operating licenses (Facility Operating License Numbers DPR-70 and DPR-75) for a period of 20 years beyond the current expiration at midnight August 13, 2016, for Unit 1, and at midnight on April 18, 2020, for Unit 2.

Salem is located approximately 40 miles from Philadelphia, PA, and 8 miles from Salem, NJ. The NRC issued the construction permits for Unit 1 and Unit 2 on August 25, 1968. The NRC issued the operating license for Unit 1 on December 1, 1976, and for Unit 2 on May 20, 1981. Both units are pressurized water reactors that were designed and supplied by Westinghouse. License Amendment Nos. 243 (Salem Unit 1) and 224 (Salem Unit 2), dated May 25, 2001, authorized a 1.4 percent increase in the licensed rated power level of each unit to 3,459 megawatt thermal (MWt).

This SER presents the status of the staff’s review of information submitted through May 18, 2011, the cutoff date for consideration in this SER. The staff has resolved all issues associated with requests for additional information and closed all open items since publishing the SER with Open Items. The staff did not identify any new open items that must be resolved before any final determination can be made on the LRA.

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A metal component that is subjected to cyclic loads may fail at load levels lower than its design load carrying capacity due to a well-known phenomenon known as fatigue. Fatigue involves crack initiation and propagation. The fatigue life of a structural component depends on the material used for the structure, the environment to which the structural component is exposed, and the number of occurrences or repetitions of cyclic loads and the magnitude of the applied fluctuating loads.

LRA Section 4.3 states that metal fatigue was evaluated in the design process for pressure boundary components, including the reactor vessel, reactor coolant pumps (RCPs), SGs, pressurizer, piping, valves, and components of primary, secondary, auxiliary, steam, and other systems. Furthermore, the applicant stated that fatigue TLAAs for pressure boundary components are characterized by determining the applicable design codes and specifications that specify the fatigue design requirements.

Fatigue is age-related degradation caused by cyclic stressing of a component by either mechanical or thermal stresses. Fatigue analyses are TLAAs if they meet the six defined elements pursuant to 10 CFR 54.3(a). If the analyses are based on a number of cycles estimated for the current license term, they may meet the 10 CFR 54.3(a)(3) criterion of "defined by the current operating term." The applicant evaluated the TLAAs in accordance with 10 CFR 54.21(c)(1).

4.3.1 Nuclear Steam Supply System Pressure Vessel and Component Fatigue Analyses

4.3.1.1 Summary of Technical Information in the Application

LRA Section 4.3.1 summarizes the evaluation of the pressure vessel components for the period of extended operation. This TLAA is based on the analysis in UFSAR Section 5.2. The applicant stated that metal fatigue evaluation was performed for the nuclear steam supply system (NSSS) pressure vessel and its components that included reactor vessel, reactor vessel closure head, pressurizer, SGs, and RCP casings. The applicant also stated that these components were designed in accordance with ASME Boiler and Pressure Vessel (B&PV) Code Section III for Class A or Class 1 and, therefore, were subject to fatigue analyses. The applicant further stated that these analyses were based upon the number and the amplitudes of design basis transients described in the design specifications and summarized in LRA Table 4.3.1-2, "Design Transient Cycles for NSSS Class A and Class 1 Components at Salem Units 1 and 2." The applicant reviewed fatigue monitoring data to determine the number of cumulative cycles of each transient that occurred during plant operation. Based on this data, the applicant derived the 60-year projected number of cycles and compared these values to the design basis number of cycles. The applicant concluded that the 60-year projected number of cycles remained bounded by the design-basis number of cycles and that the design-basis fatigue analyses will remain valid for the 60 years of operation. In this TLAA, the applicant dispositioned the TLAA pressure vessel and component fatigue analyses based on the criterion in 10 CFR 54.21(c)(1)(i).

4.3.1.2 Staff Evaluation

The staff reviewed the TLAAs in LRA Section 4.3.1 for NSSS pressure vessel and components against the acceptance criteria in SRP-LR Section 4.3.2.1.1.1 and review procedures in SRP-LR Section 4.3.3.1.1.1 in order to verify, in accordance with 10 CFR 54.21(c)(1)(i), that the NSSS pressure vessel and its components fatigue analyses remain valid for the period of extended operation.

The staff also reviewed the following additional documents that are relevant to the staff's evaluation of this TLAA:

- TS 5.7, "Component Cyclic or Transient Limit"
- UFSAR Section 5.2, "Integrity of Reactor Coolant Pressure Boundary"
- UFSAR Table 5.2-10, "Design Thermal and Loading Cycles – AREVA NP Model 61/19T SG – Unit 2"
- UFSAR Table 5.2-10a, "Design Thermal and Loading Cycles – Model F SG – Unit 1"
- 10 CFR 50.55a, "Codes and Standards"

The staff reviewed the applicant's cycle projection methodology in LRA Section 4.3.1 and the actual 60-year transient projection data in LRA Tables 4.3.1-3 and 4.3.1-4 against the design basis limits in LRA Table 4.3.1-2 to determine whether the applicant provided an acceptable basis to disposition the TLAAs in accordance with 10 CFR 54.21(c)(1)(i).

During its review, the staff noted that the applicant is using a linear basis to project the cumulative cycles for the design basis transients to the end of the period of extended operation. The staff noted the applicant's projection methodology is based on 28.5 years of operation for Unit 1 and 25.6 years of operation for Unit 2. The staff confirmed that the applicant derived an average rate of past transient occurrences using 28.5 and 25.6 years of operation for Units 1 and 2. The staff determined that the applicant derived the 60-year cycle projections by adding the cumulative number of occurrences as of December 31, 2007, to the number of cycles predicted to occur in the 31.5 and 34.4 years of future operation for Units 1 and 2, respectively. The staff concluded that this projection methodology is based on the assumption that all monitored transients would not exhibit increasing trends. During its audit and based on the additional information provided by the applicant as referenced in the audit report, the staff confirms that none of the transients listed in LRA Table 4.3.1-2 exhibited increasing trends over the period of operation for which they were assessed (i.e., operations through December 31, 2007). The staff notes that this supports the applicant's conclusion that the linear extrapolation basis is conservative because the linear averaging used in the projection basis is bounding for the actual decreasing trend in transient cycle occurrences over time.

However, the staff also noted that the applicant's 60-year transient occurrence projection basis did not indicate whether there were any gaps in the counting of the design basis transients since the initial startup of the Salem units. By letter dated June 14, 2010, the staff issued RAI 4.3-01 requesting that the applicant clarify whether the cycle counting for the design basis transients at Units 1 and 2 has been performed during the entire period of past operation.

In its response dated July 13, 2010, the applicant stated that it conducted a review of past plant documents to establish cycle counts, which included licensee event reports, monthly operating reports, and the plant's computer-based data archive system. The applicant stated that this review confirmed there were no unmonitored periods during the entire period of past operation. The applicant stated that the review included the entire time of operation except during periods of hot shutdown or cold shutdown conditions. The applicant stated that for each of the design basis transients listed in LRA Tables 4.3.1-3 and 4.3.1-4, the applicant used the larger of the two values for current cycles that either came from the 2007 annual cyclic data report or the review of plant historical information.

Based on its review, the staff finds the applicant's response to RAI 4.3-01 acceptable because the applicant has performed cycle counting during the entire period of past operation, and the applicant has performed a review of plant records to identify any uncounted transients. Further, the applicant has used the highest cycle count resulting from either of the two processes in its evaluation cycles. The staff's concern described in RAI 4.3-01 is resolved.

The staff notes that LRA Section 4.3.1 does not reference the design-basis documents used to confirm the design basis transient limits provided in LRA Table 4.3.1-2. By letter dated June 14, 2010, the staff issued RAI 4.3-02 requesting that the applicant clarify which CLB documents or design-basis documents were used to determine the design basis transient limits for those listed in LRA Table 4.3.1-2, "Design Transient Cycles for NSSS Class A and Class 1 Components at Salem Units 1 and 2."

In its response dated July 13, 2010, the applicant provided a table that lists the CLB or design-basis documents referenced for each of the transients listed in LRA Table 4.3.1-2. The list of references includes:

- Units 1 and 2 TSs, Table 5.7-1, "Component Cyclic or Transient Limits"
- UFSAR Table 5.2-10a, "Design Thermal and Loading Cycles*, Model F SG – Unit 1," Revision 24
- UFSAR Table 5.2-10, "Design Thermal and Loading Cycles*, AREVA NP Model 61/19T SG – Unit 2," Revision 24
- WCAP-12914, "Structural Evaluation of Salem Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," Revision 1
- PSEG Calculations 3SC-013, "Salem Unit 1 & 2 – NRC Bulletin 88-08 Evaluation of Aux. Spray Line," Revision 0
- Safety Evaluation SGS/M-SE-006, "Safety Injection Transients, 1 and ½ Inch Injection Nozzles – Reactor Coolant System, No. 1 Unit," Revision 0, February 9, 1977

The staff reviewed these documents and concluded that they do provide design basis transient limiting values provided in LRA Table 4.3.1-2. The staff's concern described in RAI 4.3-02 is resolved.

Therefore, based on this review, the staff concludes that the applicant's 60-year transient projection basis is acceptable because the linear extrapolation methodology is conservative relative to the actual decreasing trend in transient occurrences from recent plant operations.

The staff reviewed the 60-year cycle projections for the transients in LRA Tables 4.3.1-3 and 4.3.1-4 against the design basis limit values listed for the transients in LRA Table 4.3.1-2. The staff confirmed that the 60-year projected cycles were based on the projection methodology as described above and that for these transients, the 60-year projected number of cycles listed in LRA Tables 4.3.1-3 and 4.3.1-4 are bounded by the design basis limit values listed for the transients in LRA Table 4.3.1-2. Therefore, the staff finds that the applicant has provided a valid basis for dispositioning the TLAAs in accordance with 10 CFR 54.21(c)(1)(i) because the applicant's 60-year linear extrapolation methodology bounds the actual trend in transient occurrences and the 60-year projections results listed for the transients in LRA Tables 4.3.1-3, 4.3.1-4, 4.3.2-1, 4.3.2-2, 4.3.6-1, and 4.3.6-2 are bounded by the design basis limit values listed for these transients in LRA Table 4.3.1-2.

4.3.1.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of NSSS pressure vessel components fatigue analyses in LRA Section A.4.3.1. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's actions to address NSSS pressure vessel components fatigue analyses is adequate.

4.3.1.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.1.1.1, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that for the metal fatigue TLAA, the analyses for the NSSS pressure vessel and components remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the NSSS pressure vessel and its components, as required pursuant to 10 CFR 54.21(d).

4.3.2 Pressurizer Safety Valve and Pilot-Operated Relief Valve Fatigue Analyses

4.3.2.1 Pressurizer Safety Valve

4.3.2.1.1 Summary of Technical Information in the Application

LRA Section 4.3.2 summarizes the evaluation of pressurizer safety valves for the period of extended operation. In this TLAA, the applicant stated that the fatigue analyses for pressurizer safety valves are a TLAA that require evaluation for the period of extended operation. The applicant also stated that for the design basis analyses, the pressurizer safety valves are based on a total of 50 design cycles. The applicant derived the 60-year projected number of cycles used in fatigue analyses of the pressurizer safety valves based on fatigue monitoring data recorded during plant operation. The applicant concluded that the total number of cycles projected for 60 years for the transients of concern (loss of load, feedwater line break, RCP locked rotor, and control rod ejection) remained bounded by the design basis number of cycles, and thus the design basis fatigue analyses will remain valid for the period of extended operation.

In this TLAA, the applicant dispositioned the TLAA for fatigue of pressurizer safety valve fatigue analyses based on the criterion in 10 CFR 54.21(c)(1)(i).

4.3.2.1.2 Staff Evaluation

The staff reviewed the TLAAs in LRA Section 4.3.2.1 for fatigue of the pressurizer safety valves against the acceptance criteria in SRP-LR Section 4.3.2.1.3 and the review procedures in SRP-LR Section 4.3.3.1.3 in order to verify, in accordance with 10 CFR 54.21(c)(1)(i), that the pressurizer safety valves fatigue analyses remain valid for the period of extended operation.

The staff also reviewed the following additional documents that are relevant to the staff's evaluation of this TLAA:

- TS 5.7, "Component Cyclic or Transient Limit"
- UFSAR Section 5.5, "Components and Subsystem Design"
- UFSAR Table 5.2-10, "Design Thermal and Loading Cycles – AREVA NP Model 61/19T SG – Unit 2"
- UFSAR Table 5.2-10a, "Design Thermal and Loading Cycles – Model F SG – Unit 1"
- 10 CFR 50.55a, "Codes and Standards"

The staff notes that the applicant's metal fatigue analysis assessment for the pressurizer safety valves is based on a design specification that limits the total number of transient occurrences (for all transients applicable to the valves) to a value of 50. The staff also notes that the applicant identified that the following design basis transients are applicable to the applicant's TLAA for the pressurizer pilot-operated relief valves (PORVs): (1) "Loss of Load," (3) "Feedwater Line Break," (3) "RCP Locked Rotor" and (4) "Control Rod Ejection."

The staff notes that LRA Table 4.3.2-1 lists the current total number of occurrences to date and the 60-year projection results for the applicable design basis transients. During its review, the staff confirms that the applicant is using a linear basis to determine the 60-year cycle projections, consistent with the projection methodology evaluated and found to be acceptable by the staff in SER Section 4.3.1.

The staff notes that the applicant's evaluation is based on a projection of one occurrence each, of the "Feedwater Line Break," "RCP Locked Rotor," and "Control Rod Ejection" transients during the period of extended operation, even though there have been no occurrences of these transients at the plant during current licensed operations. The staff finds this assumption to be acceptable because the applicant has programs, requirements, or design features to minimize the probability for the occurrence of these transients. The staff confirms that, for the pressurizer safety valves, the total number of transient occurrences projected for 60 years of operation for all applicable transients is 7 and 4 for Salem Units 1 and 2, respectively. The staff notes that this demonstrates the number of transient occurrences remains bounded by the total number of transient occurrences allowed in the design specification for the valves.

The staff held a teleconference with the applicant on August 1, 2010, to discuss the disposition of the TLAAs on the pressurizer safety valves and pressurizer PORVs as discussed in LRA

Sections 4.3.2.1 and 4.3.2.2. The staff noted that the analyses the applicant claimed to be TLAAs for the pressurizer safety valves (LRA Section 4.3.2.1) and pressurizer PORVs (LRA Section 4.3.2.2) appeared to be limited only to the total number of cycles and thus, the analyses for these valve types do not appear to be associated with the evaluation of an aging effect. The staff noted that the applicant would not normally have to identify these analyses as TLAAs because they do not appear to conform to Criterion 2 in 10 CFR 54.3(a) (i.e., consider the effects of aging).

By letter dated August 26, 2010, the applicant stated that, upon further review, it determined there are no TLAAs associated with the pressurizer safety valves and PORVs, since the design analyses associated with these valves do not meet all of the criteria of a TLAA as defined in 10 CFR 54.3(a).

The applicant further stated that as part of the detailed TLAA documentation search, it found Westinghouse design specifications for component cycles associated with the valves; however, these design specifications do not consider the effects of aging of the pressurizer safety valves and PORVs. The staff noted that the second criterion of a TLAA, as defined in 10 CFR 54.3(a), states that a TLAA are those licensee calculations and analyses that consider the effects of aging. Furthermore, the staff noted that, since these analyses did not consider the effects of aging, they would not normally have been considered TLAAs; however, the LRA conservatively identified these analyses as TLAAs, evaluated the projected number of cycles associated with the valves' operations, and dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(i). The applicant amended its LRA such that the applicable sections, LRA Sections 4.3.2 and A.4.3.2, are deleted to remove the analyses associated with the valves as TLAAs.

Based on its review, the staff finds it acceptable that LRA Sections 4.3.2 and A.4.3.2 were deleted and that the fatigue analyses for the pressurizer safety valves are not TLAAs because these analyses did not consider the effects of aging and, therefore, do not meet the definition of a TLAA, as defined in 10 CFR 54.3(a).

4.3.2.1.3 UFSAR Supplement

By letter dated August 26, 2010, the applicant amended its LRA to delete LRA Section A.4.3.2. The staff's review of this amendment is documented in SER Section 4.3.2.1.2.

4.3.2.1.4 Conclusion

On the basis of its review, the staff concludes that the fatigue analyses for the pressurizer safety valves are not TLAAs, as defined in 10 CFR 54.3(a). The staff also concludes that a UFSAR supplement is not required.

4.3.2.2 Pressurizer Pilot-Operated Relief Valve Fatigue Analyses

4.3.2.2.1 Summary of Technical Information in the Application

LRA Section 4.3.2 summarizes the evaluation of pressurizer PORVs for the period of extended operation. In this TLAA, the applicant stated that the fatigue analyses for pressurizer PORVs are a TLAA that requires evaluation for the period of extended operation. The applicant also stated that for pressurizer PORVs, the design basis analyses are based on a total of 20,000 design cycles. Based on fatigue monitoring data recorded during plant operation, the applicant derived the 60-year projected number of cycles used in the fatigue analyses of the pressurizer PORVs.

The applicant concluded that the total number of cycles projected for 60 years of operation remain bounded by the design basis number of cycles and that the design basis fatigue analyses will remain valid for the period of extended operation. The applicant dispositioned the TLAA for fatigue of pressurizer PORVs based on the criterion in 10 CFR 54.21(c)(1)(i).

4.3.2.2.2 Staff Evaluation

The staff reviewed the TLAAs in LRA Section 4.3.2.2 for fatigue of the pressurizer PORVs against the acceptance criteria in SRP-LR Section 4.3.2.1.3 and the review procedures in SRP-LR Section 4.3.3.1.3 in order to confirm, in accordance with 10 CFR 54.21(c)(1)(i), that the pressurizer PORVs fatigue analyses remain valid for the period of extended operation.

The staff also reviewed the same additional documents as described in SER Section 4.3.2.1.2. The staff noted that the applicant's metal fatigue analysis for the pressurizer PORVs is based on a design specification that limits the number of transient cycles to 20,000 occurrences for all transients that are applicable to the valves. The staff also noted that the applicant identified that the following design basis transients are applicable to the applicant's TLAA for the pressurizer PORVs: (1) large step load with steam dump, (2) loss of load, (3) loss of flow, and (4) loss of power.

The staff noted that LRA Table 4.3.2-2 lists the total number of cumulative occurrences for these transients to date and the 60-year projection results for these transients. The staff confirms that these projections are based on the applicant's projection methodology provided in LRA Section 4.3.1. The staff evaluated this projection methodology in SER Section 4.3.1 and determined that the applicant's 60-year design basis transient projection basis and results were acceptable and conservative. The staff confirmed that, for the pressurizer PORVs, the total number of 60-year projected cycles is 91 and 40 for Salem Units 1 and 2, respectively. The staff notes that this projected number of transient occurrences is bounded by the number of transient occurrences allowed in the design specification for the pressurizer PORVs (i.e., less than 20,000).

The staff held a teleconference with the applicant on August 1, 2010, to discuss the disposition of the TLAAs on the pressurizer safety valves and PORVs, as discussed in LRA Sections 4.3.2.1 and 4.3.2.2. The staff noted that the analyses that the applicant claimed to be TLAAs for the pressurizer safety valves (LRA Section 4.3.2.1) and pressurizer PORVs (LRA Section 4.3.2.2) appeared to be limited only to the total number of cycles and thus, the analyses for these valve types do not appear to be associated with the evaluation of an aging effect. The staff noted that the applicant would not normally have to identify these analyses as TLAAs because they do not appear to conform to Criterion 2 in 10 CFR 54.3(a) (i.e., consider the effects of aging).

By letter dated August 26, 2010, the applicant stated that, upon further review, it determined there are no TLAAs associated with the pressurizer safety valves and PORVs, since the design analyses associated with these valves do not meet all of the criteria of a TLAA as defined in 10 CFR 54.3(a).

The staff's review of the August 26, 2010, letter and the deletion of LRA Sections 4.3.2 and A.4.3.2 are documented in SER Section 4.3.2.1.2.

Based on its review, the staff finds it acceptable that LRA Sections 4.3.2 and A.4.3.2 were deleted and that the fatigue analyses for the pressurizer PORVs are not TLAAs because these analyses did not consider the effects of aging and, therefore, do not meet the definition of a TLAA as defined in 10 CFR 54.3(a).

4.3.2.2.3 UFSAR Supplement

By letter dated August 26, 2010, the applicant amended its LRA to delete LRA Section A.4.3.2. The staff's review of this amendment is documented in SER Section 4.3.2.1.2.

4.3.2.2.4 Conclusion

On the basis of its review, the staff concludes that the fatigue analyses for the pressurizer PORVs are not TLAAs, as defined in 10 CFR 54.3(a). The staff also concludes that a UFSAR supplement is not required.

4.3.3 American Standards Association/United States of America Standards B31.1 Piping Fatigue Analyses

4.3.3.1 Summary of Technical Information in the Application

LRA Section 4.3.3 summarizes the evaluation of American Standards Association/United States of America Standards (ASA/USAS) B31.1 piping for the period of extended operation. This TLAA is based on the analysis in UFSAR Section 5.2. In this TLAA, the applicant stated that the piping was designed in accordance with ASA/USAS B31.1 piping code and, therefore, fatigue analyses were not required, but cyclic load was considered in a simplified manner in the design process. The applicant determined that the total number of 60-year projected cycles does not exceed 7,000 cycles, which is the minimum number of cycles required that would result in application of an allowable stress reduction factor. Therefore, the applicant concluded that the existing analyses of ASA/USAS B31.1 piping for which the allowable range of secondary stresses depends on the number of assumed thermal cycles, remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

4.3.3.2 Staff Evaluation

The staff reviewed the TLAA in LRA Section 4.3.3 for fatigue of ASA/USAS B31.1 piping against the acceptance criteria in SRP-LR Section 4.3.2.1.2.1 and the review procedures in SRP-LR Section 4.3.3.1.2.1 in order to verify, in accordance with 10 CFR 54.21(c)(1)(i), that the ASA/USAS B31.1 piping fatigue analyses remain valid for the period of extended operation.

The staff reviewed the applicant's cycle projection methodology in LRA Section 4.3.1 and found the applicant's methodology acceptable. From the information provided in LRA Tables 4.3.1-3 and 4.3.1-4, the staff determined that the total number of projected cycles for the design transients applicable to the ASA/USAS B31.1 piping used 4,936 and 4,264 for Salem Units 1 and 2, respectively, and will not exceed the 7,000-cycle limit. Therefore, the staff concludes that the applicant's design transient cycle projection for the period of extended operation will be less than the limit of 7,000 cycles and thus the analysis remain valid for the period of extended operation.

4.3.3.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of ASA/USAS B31.1 piping fatigue analyses in LRA Section A.4.3.3. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's metal fatigue TLAA for the ASA/USAS B31.1 piping is adequate.

4.3.3.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.1.2.1, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the metal fatigue analyses for the ASA/USAS B31.1 piping remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the ASA/USAS B31.1 piping, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.3.4 Supplementary ASME Code Section III, Class 1 Piping and Component Fatigue Analyses

4.3.4.1 NRC Bulletin 88-08, Thermal Stresses in Piping Connected to Reactor Coolant Systems

4.3.4.1.1 Summary of Technical Information in the Application

LRA Section 4.3.4 summarizes the evaluation of supplementary ASME Code Section III, Class 1 piping and component fatigue analysis for the period of extended operation. This TLAA is based on the analysis in response to NRC Bulletin 88-08. In this TLAA, the applicant stated that Units 1 and 2 piping systems were originally designed in accordance with the ASA/USAS B31.1 piping code, however, a number of updated fatigue analyses have been performed for some piping systems and components to address transients that have been identified based on industry practice that were not originally considered. The applicant further stated that these transients include those associated with potential valve leakage transients identified in GL 88-08 for the auxiliary spray line.

The applicant stated that the staff approved Salem's response to NRC Bulletin 88-08, which included the evaluation of the fatigue analyses of the normal and alternate charging lines and the auxiliary spray lines. The applicant also stated that the analyses were based on the requirements of ASME Code Section III, 1986 Edition, Subsection NB-3653 and the fatigue curves of I-9.2.1 and I-9.2.2 and concluded that the cumulative usage factor (CUF) would remain less than 1.0 for the normal and alternate charging lines.

The applicant also performed a fatigue evaluation of the auxiliary spray line for a life of 40 years. The analysis showed that the inadvertent auxiliary spray transient controlled the calculated fatigue usage. The resulting fatigue usage was calculated to be less than 1.0 for 40 years.

In this TLAA, the applicant dispositioned the TLAA for the auxiliary spray lines in accordance with 10 CFR 54.21(c)(1)(i) and the normal and alternate charging lines in accordance with 10 CFR 54.21(c)(1)(iii) for the period of extended operation using the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

4.3.4.1.2 Staff Evaluation

During its review, the staff noted that the applicant is using a linear basis to project the cumulative cycles for the design basis transients to the end of the period of extended operation. The staff accepted the applicant's methodology in SER Section 4.3.1. The staff determined that the applicant revised the auxiliary spray lines fatigue analyses to reduce the original design basis transients from 10 to 5 inadvertent auxiliary spray transients, in response to GL 88-08, in 1999.

The staff confirmed that the 60-year projected cycles for the inadvertent auxiliary spray transient are 2 and 3 for Units 1 and 2, respectively, from LRA Tables 4.3.1-3 and 4.3.1-4. These projected cycle counts are less than the design basis of 10 for this transient. Based on this review, the staff finds that the applicant has provided an acceptable basis for demonstrating that the metal fatigue TLAA for the auxiliary spray lines are acceptable in accordance with 10 CFR 54.21(c)(1)(i) because the staff has confirmed that the number of auxiliary spray transient occurrences, as projected through the period of extended operation, will be bounded by the number of occurrences allowed under the applicant's design basis for this transient.

The staff's review of the normal and alternate charging lines determined that the applicant previously revised the charging lines fatigue analyses to include additional transients, in response to GL 88-08. During aging management program (AMP) audit interviews of the applicant's technical staff, the NRC staff clarified that additional transients incorporated into the charging lines fatigue analyses were included in LRA Tables 4.3.1-3 and 4.3.1-4. These transients are inadvertent auxiliary spray to pressurizer and inadvertent safety injection transients. To address the reactor coolant environmental effects, the applicant re-evaluated the charging lines (the charging to pipe weld) fatigue analysis. The applicant presented the results of this re-evaluation in LRA Section 4.3.7. The staff's evaluation and acceptance of the fatigue analyses for the charging lines is documented in SER Section 4.3.7.

4.3.4.1.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of supplementary ASME Code Section III, Class 1 piping and components fatigue analyses in LRA Section A.4.3.4. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's metal fatigue TLAA for the supplementary ASME Code Section III, Class 1 piping and components is adequate.

4.3.4.1.4 Conclusion

On the basis of its review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the auxiliary spray lines remain valid for the period of extended operation. The staff's evaluation and acceptance of the charging lines are documented in SER Section 4.3.7. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the supplementary ASME Code Section III, Class 1 piping and components, as required by 10 CFR 54.21(d).

4.3.4.2 NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification

4.3.4.2.1 Summary of Technical Information in the Application

LRA Section 4.3.4 summarizes the evaluation of supplementary ASME Code Section III, Class 1 piping and component fatigue analysis for the period of extended operation. This TLAA is based on the analysis in response to NRC Bulletin 88-11. The applicant stated that Units 1 and 2 piping systems were originally designed in accordance with the ASA/USAS B31.1 piping code, however, a number of updated fatigue analyses have been performed for some piping systems and components to address transients that have been identified based on industry practice that were not originally considered.

The applicant further stated that these transients include those associated with thermal stratification of the pressurizer surge line as described in NRC Bulletin 88-11. LRA Section 4.3.4 also stated that a plant-specific WESTEMS™ model was developed for the pressurizer and surge line to evaluate the effects of pressurizer insurge and outsurge transients and surge line stratification on the pressurizer surge nozzle safe end to pipe weld and the surge line hot leg nozzle. These results were also used in the evaluation of the reactor water environmental effects on the surge line.

In this TLAA, the applicant dispositioned the TLAA for the pressurizer surge line based on the criterion in 10 CFR 54.21(c)(1)(ii).

4.3.4.2.2 Staff Evaluation

The staff's review of the pressurizer surge line thermal stratification determined that the applicant previously evaluated the effects of thermal stratification and plant-specific transients on the pressurizer surge line, in response to GL 88-11. This evaluation demonstrated that the surge line weld to the pressurizer surge nozzle is a controlling location for the pressurizer surge line. To address reactor coolant environmental effects, the applicant re-evaluated the pressurized surge line (the pressurizer surge line hot leg nozzle and pressurizer nozzle to safe end weld) using ASME B&PV Code Section III, Class 1 fatigue analysis. The applicant presented the results of this re-evaluation in LRA Section 4.3.7.

During its review of the LRA, the staff identified concerns regarding the results determined by the WESTEMS™ program as a part of the ASME Code fatigue evaluation process. For example, Westinghouse's response to NRC questions regarding the AP1000 Technical Report (see Agencywide Document Access and Management System (ADAMS) Accession No. ML102300072, dated August 13, 2010) describes the ability of users to modify intermediate data (peak and valley stresses/times) used in the analyses. In addition, a response provided on August 20, 2010 (ADAMS Accession No. ML102350440), describes different approaches for summation of moment stress terms. These items can have significant impacts on calculated fatigue CUF. The staff issued an RAI requesting information on how WESTEMS™ was used in the Salem analyses, whether these issues apply to the Salem analyses, the environmentally-assisted fatigue (EAF) analyses, and the differences between the stress models used in WESTEMS™ and the stress models used in the current governing analysis of record and the EAF analysis of record. The staff also requested a benchmarking evaluation to compare calculated stresses and CUF using WESTEMS™ to the same results from the initial design basis analyses of record. This was identified as Open Item OI 4.3.4.2-1. This Open Item was closed and its resolution is discussed in SER Section 3.0.3.2.18.

The staff's evaluation of the fatigue analyses for the pressurizer surge line is documented in SER Section 4.3.7.

4.3.4.2.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of supplementary ASME Code Section III, Class 1 piping and components fatigue analyses in LRA Section A.4.3.4. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, and the closure of Open Item OI 4.3.4.2-1, the staff concludes that the summary description of the applicant's metal fatigue TLAA for the supplementary ASME Code Section III, Class 1 piping and components is adequate.

4.3.4.2.4 Conclusion

The staff's evaluation and acceptance of the pressurizer surge line are documented in SER Section 4.3.7. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the supplementary ASME Code Section III, Class 1 piping and components, as required by 10 CFR 54.21(d).

4.3.4.3 Salem Unit 1 Steam Generator Feedwater Nozzle Transition Piece

4.3.4.3.1 Summary of Technical Information in the Application

LRA Section 4.3.4 summarizes the evaluation of supplementary ASME Code Section III, Class 1 piping and component fatigue analysis for the period of extended operation. This TLAA is based on the replacement of the Unit 1 SGs. In this TLAA, the applicant stated that Units 1 and 2 piping systems were originally designed in accordance with the ASA/USAS B31.1 piping code, however, a number of updated fatigue analyses have been performed for some piping systems and components to address transients that have been identified based on industry practice that were not originally considered. The applicant also stated that, as a part of the Salem Unit 1 SG replacement, a new feedwater nozzle transition piece forging was designed in accordance with ASME B&PV Code Section III, Class 1.

In this TLAA, the applicant dispositioned the TLAA for the feedwater nozzle transition piece forging based on the criterion in 10 CFR 54.21(c)(1)(iii) for the period of extended operation using the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

4.3.4.3.2 Staff Evaluation

The staff's review of the feedwater nozzle transition piece determined that hot standby operation transients were replaced with thermal stratification loadings in the updated fatigue analysis for the feedwater nozzle transition piece forging. For the remaining plant life of 15 cycles, the applicant assumed 800 hours of auxiliary feedwater flow per cycle, resulting in a design limit of 12,000 hours of auxiliary feedwater operation. The applicant stated that the thermal stratification loads are managed by the Metal Fatigue of Reactor Coolant Pressure Boundary Program, where the number of auxiliary feedwater flow operational hours will be tracked and compared to the design limit of 12,000 hours. However, the LRA does not provide sufficient information for the staff to determine how the Metal Fatigue of Reactor Coolant Pressure Boundary Program tracks and compares the design limit of 12,000 hours for the auxiliary feedwater flow operation, and which transients tracked by the Metal Fatigue of Reactor Coolant Pressure Boundary Program will assure that the design limit of 12,000 hours for the auxiliary feedwater flow operation is not exceeded. By letter dated June 14, 2010, the staff issued RAI 4.3-04 requesting that the applicant justify why the enhancement of the Metal Fatigue of Reactor Coolant Pressure Boundary Program for tracking of the hourly operations of this transient is an acceptable basis to disposition this TLAA in accordance with 10 CFR 54.21(c)(1)(iii).

In its response dated July 13, 2010, the applicant stated that it has revised its management of the Salem Unit 1 SG feedwater nozzle transition piece and rather than manually tracking hours of the auxiliary feedwater pump during the period of extended operation, the applicant will use WESTEMS™ to automatically compute the CUF for the Unit 1 SG feedwater nozzle transition piece. The applicant further stated that a design limit will be determined for cumulative usage, based on auxiliary feedwater operation, at the transition piece as opposed to tracking the number of auxiliary feedwater flow operational hours. The applicant stated that the design limit is a CUF of 1.0. The applicant stated that all the design basis transients considered in the original analysis will remain the same and these transients are monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary Program. The applicant stated that the hot standby transient was replaced with the thermal stratification loads, which are caused by the auxiliary feedwater pump. The applicant further stated that if the fatigue usage for this location approaches 80 percent of the design limit, the corrective action program will be initiated to evaluate the condition and determine corrective actions.

Based on its review, the staff finds the applicant's response to RAI 4.3-04 acceptable because the applicant has modified its approach for aging management based on the pump operation hours to CUF values and the applicant's Metal Fatigue of Reactor Coolant Pressure Boundary Program ensures that the cumulative usage design limit of 1.0 is not exceeded. During its review of the LRA, the staff identified concerns regarding the results determined by the WESTEMS™ program as a part of the ASME Code Section III fatigue evaluation. This concern was identified as Open Item OI 4.3.4.2-1 and its resolution is discussed in SER Section 3.0.3.2.18. The staff's concern with the issue on the use of WESTEMS™ as described in RAI 4.3-04 is resolved.

4.3.4.3.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of supplementary ASME Code Section III, Class 1 piping and components fatigue analyses in LRA Section A.4.3.4. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's metal fatigue TLAA for the supplementary ASME Code Section III, Class 1 piping and components is adequate.

4.3.4.3.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.1.1.3, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the feedwater nozzle transition piece forging intended functions will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the supplementary ASME Code Section III, Class 1 piping and components, as required by 10 CFR 54.21(d).

4.3.4.4 Salem Unit 1 Steam Generator Primary Manway Studs

4.3.4.4.1 Summary of Technical Information in the Application

LRA Section 4.3.4 summarizes the evaluation of supplementary ASME Code Section III, Class 1 piping and component fatigue analysis for the period of extended operation. This TLAA is based on the qualification of the SG primary manway studs for a longer life. In this TLAA, the applicant stated that Units 1 and 2 piping systems were originally designed in accordance with the ASA/USAS B31.1 piping code, however, a number of updated fatigue analyses have been performed for some piping systems and components to address transients that have been identified based on industry practice that were not originally considered. The applicant also stated that, as a part of the Unit 1 SG replacement, the design basis for Unit 1 SG manway studs was updated to include fatigue considerations.

In this TLAA, the applicant dispositioned the TLAA for the Salem Unit 1 SG manway studs based on the criterion in 10 CFR 54.21(c)(1)(i).

4.3.4.4.2 Staff Evaluation

The staff's review of the SG manway studs fatigue analysis determined that, as specified in the LRA, Westinghouse conducted a series of tests to qualify the SG manway studs for 40 years of plant operation. The staff also noted that, although LRA Section 4.3.4.4 indicated that the 60-year projected cycles for the Unit 1 SG manway studs were bounded by the number of cycles assumed in the 40-year design basis fatigue analysis, the LRA did not provide sufficient information to identify which transients were used in the design basis analysis and the 60-year fatigue analysis of the SG manway studs. By letter dated June 14, 2010, the staff issued RAI 4.3-03 requesting that the applicant identify what transients were used in the 40-year fatigue analysis of the SG manway studs and clarify whether limiting cycle numbers for these transients were equivalent to the design basis transient limits.

In its response dated July 13, 2010, the applicant stated that Westinghouse conducted a series of tests to qualify the SG manway studs for a 40-year life. The applicant further stated that these tests were performed for Westinghouse Model F SGs in accordance with ASME Code Section III, Appendix II, 1989 Edition. The applicant stated that the test parameters were determined by using the design transients from the general design specification for the Westinghouse Model F SG. The applicant stated that because the transients used for the fatigue qualification tests considered a larger population of SGs, the testing parameters included additional transients (i.e., reactor coolant pipe break, steam pipe break, operating basis earthquake (OBE), etc.). The applicant further stated that all of the 40-year design transients in the general design specification for Model F SGs were determined to bound the corresponding 40-year design transients for the Unit 1 SGs. The applicant stated that the 40-year design transients for the Unit 1 Model F SGs are bounded by those presented in LRA Table 4.3.1-3 and that there are no other 40-year design transients that are applicable to the Unit 1 Model F SG primary manway studs fatigue analyses that were not listed in LRA Table 4.3.1-3. The applicant further stated that the 60-year cycle projections contained in LRA Table 4.3.1-3 are bounded by the test parameters used for the primary manway stud fatigue qualification testing. The applicant also stated that Westinghouse concluded after fatigue testing that the CUF was less than 1.0. The applicant further stated that because the 60-year cycle projections are bounded by the test parameters, the 60-year projected CUF is also less than 1.0.

Based on its review, the staff finds the applicant's response to RAI 4.3-03 acceptable because: (1) the applicant indicated that the Steam Generator Primary Manway Studs have been fatigue tested in accordance with the ASME Code and (2) this fatigue testing bounds the design bases transient limits and the 60-year projected cycles are less than the design bases limits, which means that the fatigue testing also bounds the period of extended operation. The staff's concern described in RAI 4.3-03 is resolved.

4.3.4.4.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of supplementary ASME Code Section III, Class 1 piping and components fatigue analyses in LRA Section A.4.3.4. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's metal fatigue TLAA for the supplementary ASME Code Section III, Class 1 piping and components is adequate.

4.3.4.4.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.1.1.1, the staff concludes that the applicant has demonstrated pursuant to 10 CFR 54.21(c)(1)(i), that the Unit 1 SG manway studs fatigue analyses remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the supplementary ASME Code Section III, Class 1 piping and components, as required pursuant to 10 CFR 54.21(d).

4.3.5 Reactor Vessel Internals Fatigue Analyses

4.3.5.1 Summary of Technical Information in the Application

LRA Section 4.3.5 summarizes the evaluation of reactor vessel internals for the period of extended operation. In this TLAA, the applicant stated that the Salem reactor vessel internals were designed and constructed prior to the development of ASME code requirements for core support structures, and the RCS functional design requirements were considered. The applicant also stated that the reactor vessel internals were implicitly designed for low cycle fatigue based upon the RCS design basis transients and were identified as a TLAA. In this TLAA, the applicant dispositioned the TLAA for reactor vessel internals fatigue analyses based on the criterion in 10 CFR 54.21(c)(1)(i).

4.3.5.2 Staff Evaluation

The staff reviewed the TLAA in LRA Section 4.3.5 for reactor vessel internals fatigue analyses against the acceptance criteria in SRP-LR Section 4.3.2.1.3 and the review procedures in SRP-LR Section 4.3.3.1.3 in order to verify, in accordance with 10 CFR 54.21(c)(1)(i), that the reactor vessel internals fatigue analyses remain valid for the period of extended operation.

During its review, the staff noted that LRA Section 4.3.5 states that the reactor vessel internals were designed based on the RCS design transient projections for 40 years. During the AMP audit and based on the additional information provided by the applicant as referenced in the Audit Report, the staff clarified that the RCS design transient projections for 40 years refer to the RCS design-basis transients. The staff reviewed the 60-year cycle projections, as summarized in LRA Tables 4.3.1-3 and 4.3.1-4, and confirmed that these projections were based on the projection methodology as described in SER Section 4.3.1. The staff further confirmed that, for transients used in the reactor vessel internals fatigue analyses, the 60-year projected number of transient cycles for the reactor vessel internals are bounded by the design basis number of cycles. Therefore, the staff concludes that the applicant has provided a valid basis for dispositioning the metal fatigue TLAA for the reactor vessel internals in accordance with the criterion in 10 CFR 54.21(c)(1)(i) because: (1) the applicant’s 60-year linear extrapolation basis for the transients in LRA Tables 4.3.1-3 and 4.3.1-4 bounds the actual trend in transient occurrences for the Salem units, and (2) the staff has confirmed that the 60-year transient occurrence projections for these components are bounded by the design-basis limit values listed for these transients.

4.3.5.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of reactor vessel internals fatigue analyses in LRA Section A.4.3.5.

On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's metal fatigue TLAA for the reactor vessel internal components is adequate.

4.3.5.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.1.3, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), that the metal fatigue TLAA for the reactor vessel internals remains valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the reactor vessel internal components, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.3.6 Spent Fuel Pool Bottom Plates Fatigue Analyses

4.3.6.1 Summary of Technical Information in the Application

LRA Section 4.3.6 summarizes the evaluation of fatigue on spent fuel pool (SFP) bottom plates for the period of extended operation. This TLAA is based on a response to a staff RAI dated February 26, 1996, for when an analysis was performed to show that the SFP liner and anchors would not experience significant deformations as a result of thermal loadings. Because the SFP liner and anchors were identified as a TLAA for the 40-year plant life, the applicant performed an evaluation of these components for the period of extended operation. The applicant further stated that based on these analyses, the resulting number of allowable cycles for the SFP liner bottom plates plant normal heatup and cooldowns is 1,638 cycles. This number of allowable cycles is much greater than the projected number of plant heatups and cooldowns (266 for Unit 1 and 312 for Unit 2).

The applicant also stated that a separate analysis of the SFP liner bottom plate and anchors determines a CUF of 0.00063 under upset conditions, based on one design-basis event (DBE) and 20 OBE cycles. The applicant projects 1 DBE and 2 OBEs for Unit 1, and 1 DBE and 3 OBEs for Unit 2.

The applicant stated that because the 60-year projected number of cycles used in fatigue analyses of the SFP liner and anchors remained bounded by the design basis number of cycles, the design basis fatigue analyses will remain valid for 60 years of operation. The applicant dispositioned the TLAA for fatigue of SFP bottom plates based on 10 CFR 54.21(c)(1)(i).

4.3.6.2 Staff Evaluation

The staff reviewed the TLAA in LRA Section 4.3.6 for fatigue of SFP bottom plates against the acceptance criteria in SRP-LR Section 4.3.2.1.3 and the review procedures in SRP-LR Section 4.3.3.1.3 in order to verify that the SFP liner and anchors fatigue analyses remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

The staff also reviewed the following additional documents that are relevant to the staff's evaluation of this TLAA:

- UFSAR Section 9.1.2, "Spent Fuel Pool"
- UFSAR Table 5.2-10, "Design Thermal and Loading Cycles – AREVA NP Model 61/19T SG – Unit 2"
- UFSAR Table 5.2-10a, "Design Thermal and Loading Cycles – Model F SG – Unit 1"
- 10 CFR 50.55a, "Codes and Standards"

During its review, the staff noted that the applicant's 60-year cycle projections for plant heatups and cooldowns were based on the projection methodology accepted by the staff in SER Section 4.3.1. The staff further confirmed that the total number of 60-year projected cycles is 266 and 312 for Units 1 and 2, respectively, and would remain bounded by the 1,638 allowable cycle limit.

Since the plant has experienced neither an OBE nor a DBE, the staff further confirms that the 60-year cycle projections would remain bounded by 1 DBE and 20 OBE cycles. Therefore, the staff concludes that the applicant's design transient cycle projection provides a conservative estimate of the number of transients occurring through the period of extended operation because the transients are not expected to go over the design-basis value based on the observed operating experience.

4.3.6.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of the SFP bottom plates fatigue analyses in LRA Section A.4.3.6. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes that the summary description of the applicant's actions to address the SFP bottom plates fatigue analyses is adequate.

4.3.6.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.1.3, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(i), the analyses for the SFP bottom plate liner and anchors will remain valid for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation for the SFP bottom plate liner and anchors, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

4.3.7 Environmentally-Assisted Fatigue Analyses

4.3.7.1 Summary of Technical Information in the Application

LRA Section 4.3.7 summarizes the evaluation of EAF for the period of extended operation. This TLAA evaluates the effects of the RCS environment on the following fatigue life representative components that are identified in NUREG/CR-6260 for older vintage Westinghouse plants:

- reactor vessel shell and lower head
- reactor vessel inlet and outlet nozzles
- surge line
- charging system nozzle
- safety injection system nozzle
- residual heat removal system Class 1 piping

In this TLAA, the applicant stated that the plant-specific components were identified for the NUREG/CR-6260 sample locations and EAF calculations followed the guidance of NUREG/CR-6583 for components made of carbon and low-alloy steels and the guidance of NUREG/CR-5704 for components made of austenitic stainless steel. The applicant further stated that no CUF values considering environmental effects will exceed the code limit of 1.0 for 60 years of operation. In this TLAA, the applicant dispositioned the TLAA for EAF based on 10 CFR 54.21(c)(1)(ii).

4.3.7.2 Staff Evaluation

The staff reviewed the TLAAs in LRA Section 4.3.7 for EAF against the acceptance criteria in SRP-LR Section 4.3.2.2 and the review procedures in SRP-LR Section 4.3.3.2 in order to verify, in accordance with 10 CFR 54.21(c)(1)(ii), that the analyses for the NUREG/CR-6260 sample locations have been projected to the end of the period of extended operation.

During its review, the staff determined that, using plant-specific design fatigue results, the applicant identified the plant-specific components and limiting components locations for the NUREG/CR-6260 sample locations and performed EAF calculations for these components to evaluate the effects of the RCS environment on fatigue life. However, the LRA does not provide sufficient information on the methodology used in determining the plant-specific components and limiting component locations for the NUREG/CR-6260 sample locations. By letter dated June 14, 2010, the staff issued RAI 4.3-05 requesting that the applicant justify the methodology, assumptions, component locations, and results that the applicant included in the EAF evaluation for the LRA.

In its response dated July 13, 2010, the applicant provided the methodology used to determine the Salem plant-specific locations that bound the locations provided in the NRC guidance document NUREG/CR-6260.

For the reactor vessel shell and lower head, the applicant stated that it selected the core support guide welds as the limiting component based on guidance provided in Section 5.5.1 of NUREG/CR-6260. The applicant further stated that the controlling fatigue location is the outer corner of the weld that connects the core support guide to the reactor vessel inner wall. For the reactor vessel inlet and outlet nozzles, the applicant selected the reactor vessel inlet and outlet nozzles as the limiting components based on the guidance provided in Section 5.5.2 of NUREG/CR-6260. The applicant further stated that the controlling fatigue location is the outside surface of the nozzle-to-shell juncture. For the pressurizer surge line, the applicant stated that it evaluated fatigue in WCAP-12913, “Structural Evaluation of Salem Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification,” Revision 1. The applicant further stated that additional fatigue analysis was conducted for the pressurizer lower head and surge nozzles in WCAP 16194, “Evaluation of Pressurizer Insurge/Outsurge Transients for Salem Units 1 and 2,” Revision 0. The applicant stated that it used both these fatigue calculations and the information provided in NUREG/CR-6260 Section 5.5.3 to select the surge line hot leg nozzles as a limiting component for the pressurizer surge line. For the RCS piping charging system nozzles, the applicant stated that both the normal and alternating charging nozzles were chosen based on the guidance provided in NUREG/CR-6260 Section 5.5.4. The applicant further stated that it developed a detailed model of the nozzles and applied a stress analysis for the nozzles and connections to determine the exact limiting locations. The staff noted that this limiting location is the weld that connects the nozzle to the charging line piping. For the RCS piping safety injection nozzles, the applicant stated that it reviewed the safety injection system nozzles connected to the RCS cold leg based on the guidance provided in NUREG/CR-6260 Section 5.5.5. Based on this review, the applicant stated that the 1.5-inch boron injection tank nozzles were selected to represent this location. The applicant further stated that it developed a detailed model of the 1.5-inch boron injection tank nozzles and applied a stress analysis, which determined the fatigue controlling location was the boron injection tank piping region at the socket weld that connects the nozzle to the safety injection line piping. The applicant stated that for the residual heat removal system Class 1 piping, it used guidance in NUREG/CR-6260 Section 5.5.6 to review the residual heat removal system Class 1 piping, specifically the letdown path and return path to the RCS primary loop. Based on this review, the applicant stated it determined the 10-inch accumulator/residual heat removal injection cold leg nozzles to be the limiting fatigue location. The applicant further stated that it developed a detailed model and applied stress analyses for the 10-inch accumulator/residual heat removal injection cold leg nozzles and their connections to determine that the controlling fatigue location is the weld that connects the accumulator nozzle to the residual heat removal line piping.

The applicant responded to the question on the assumption used for the 60-year EAF calculations by first generating the 60-year CUF for the six sample locations listed in LRA Tables 4.3.7-1 and 4.3.7-2 and then applying the environmental fatigue life correction factor, F_{en}. The first assumption the applicant made was that the 40-year NSSS transient design cycles and auxiliary transient design cycles, or their respective 60-year projected number of cycles, would bound the actual number of cycles experienced during the period of extended operation. The applicant stated that it will validate the basis for this assumption by implementing the Metal Fatigue of Reactor Coolant Pressure Boundary Program to monitor transients and use the WESTEMS™ code to compute the cumulative fatigue at select NUREG/CR-6260 sample locations to ensure that the 60-year CUF values remain less than the design limit.

In the applicant’s response to the request for the assumptions used in the F_{en} calculations, the applicant stated it used the NUREG/CR-6583 and NUREG/CR-5704 methodologies to evaluate the environmental effects on carbon, low-alloy, and stainless steels. For low-alloy steel components, the applicant stated that it set both the temperature and oxygen content parameter to zero, which will maximize the F_{en} value at 2.532 for low-alloy steel components. For stainless steel components, it assumed that the oxygen content was less than 0.05 parts per million (ppm), which is based on normal operations of less than 5 ppb. The applicant further stated that it reviewed the dissolved oxygen data, which indicated that the dissolved oxygen content was less than 0.05 ppm since 2000, except for short periods of time during start-up and shutdown conditions. To determine the strain rate, the applicant stated it used an integrated method known as the modified rate approach. The applicant also stated that transient total stress time histories were used to determine the corresponding strain rates of the tensile producing portion of the stress cycle for the different fatigue pairs for all of the applicable analyzed transients.

The staff notes that the applicant's response did not specify the dissolved oxygen data prior to 2000 and that it is not clear whether the applicant's primary water chemistry specifications maintained dissolved oxygen less than 0.05 ppm since initial plant start-up. The staff notes that if there were extended periods of time, prior to 2000, in which the applicant operated with dissolved oxygen greater than 0.05 ppm, the assumptions used in the determination of the F_{en} value for carbon and low-alloy steels may not be valid. This is important to the carbon and low-alloy steel components because a dissolved oxygen content greater than 0.05 ppm can increase the F_{en} value. The staff notes that the assumption of less than 0.05 ppm dissolved oxygen is conservative when determining the F_{en} value for stainless steel because it increases the F_{en} value. The staff identified this as Open Item OI 4.3.4.2-1.

Regarding the question whether the critical fatigue locations include nickel alloys, the applicant stated that none of the six critical fatigue locations include nickel alloy materials and that low-alloy steel is used to construct the components for the critical fatigue locations associated with the reactor vessel shell and lower head and reactor vessel inlet and outlet nozzles. The applicant also stated that stainless steel is used in the construction of the critical fatigue locations associated with the: (1) pressurizer surge line nozzle, (2) RCS piping charging system nozzles, (3) RCS piping system safety injection nozzles, and (4) residual heat removal system Class 1 piping.

In response to the question requesting if there are other plant-specific locations that may be more limiting than those identified in NUREG/CR-6260, the applicant stated the selection of the locations are compliant with NUREG/CR-6260 and the determination of the limiting locations was presented in response to the first request of this RAI. The applicant stated that because the locations are compliant with NUREG/CR-6260 and the limiting locations were identified and evaluated, no other plant-specific locations were required to be identified and evaluated for EAF. The staff notes that SRP-LR Section 4.3.2.2 states that the critical components should include, as a minimum, those selected in NUREG/CR-6260. Furthermore, the staff notes that there may be more limiting plant-specific locations (e.g., locations with a higher CUF value). It is not clear to the staff whether these locations were also considered or are the locations selected by the applicant for EAF evaluations, the limiting plant-specific locations (e.g., locations with a higher CUF value) for the plant. The staff was concerned whether the applicant verified that the locations per NUREG/CR-6260 are bounding as compared to other plant-specific locations (e.g., locations with a higher CUF value). This was also identified as part of Open Item OI 4.3.4.2-1.

By letter dated November 22, 2010, the staff issued RAI 4.3-08 to address both portions of Open Item OI 4.3.4.2-1. RAI 4.3-08, Part 1 requested the applicant to confirm and justify that the locations selected for EAF analyses, consistent with NUREG/CR-6260, are the most limiting and bounding for the plant. Furthermore, if these locations are not the most limiting and bounding for the plant, clarify the locations that require an EAF analysis and the actions that will be taken for these additional locations. If the most limiting location consists of nickel alloy, the NUREG/CR-6909 methodology for nickel alloy will be used. The staff also requested in RAI 4.3-08, Part 2 that the applicant justify the statement, “Fen is maximized when these two terms are set equal to zero” made in response to RAI 4.3-05. Finally, the staff requested in

Part 3 that the applicant clarify whether dissolved oxygen content has always been maintained less than 0.05 ppm since initial plant start-up, and provide justification to support this clarification. If not, justify why the Fen values provided in LRA Tables 4.3.7-1 and 4.3.7-2 do not account for these periods of time in which dissolved oxygen content was not maintained less than 0.05 ppm, including the "short periods of time during start-up and shutdown conditions."

In its response to Part 1, dated December 21, 2010, the applicant committed (Commitment No. 52) to the following:

> [It] will perform a review of design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 based locations that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting locations for the Salem plant configuration. If more limiting locations are identified, the most limiting location will be evaluated for the effects of the reactor coolant environment on fatigue usage. If any of the limiting locations consist of nickel alloy, NUREG/CR-6909 methodology for nickel alloy will be used in the evaluation.

Based on its review, the staff finds the applicant's responses to RAI 4.3-05; RAI 4.3-08, Part 1; and Commitment No. 52 acceptable because: (1) the applicant will review its design basis ASME Code Class 1 fatigue evaluations to determine whether the NUREG/CR-6260 based locations are the limiting locations for its plant-specific configuration; (2) if more limiting locations are identified, the applicant will perform EAF analyses for the most limiting location; (3) if any of the limiting locations consist of nickel alloy, the NUREG/CR-6909 methodology for nickel alloy will be used in the evaluation; (4) NUREG/CR-6909 will be used for determining a conservative F_{en} factor for any new nickel-alloy components that require EAF analysis; and (5) Commitment No. 52 is consistent with the recommendations in SRP-LR Sections 4.3.2.2 and 4.3.3.2, and GALL AMP X.M1, to consider environmental effects for the NUREG/CR-6260 locations, at a minimum. The staff's concerns described in RAI 4.3-05 and RAI 4.3-08, Part 1 are resolved, and this portion of Open Item OI 4.3.4.2-1 is closed.

In its response to Part 2, dated December 21, 2010, the applicant clarified that the two terms in the statement, "F_{en} is maximized when these two terms are set equal to zero" referred to the correction temperature, T, and the transformed oxygen content parameter, O*. The staff noted that during the applicant's review, it identified a typographical error in its response to RAI 4.3-05 (Part 3), dated July 13, 2010, and amended the term "0.001124T" to "0.00124T." The staff reviewed Equation 6.5b of NUREG/CR-6583 and confirmed that the use of the term "0.00124T" is correct. The applicant stated that it agrees that the above statement is not accurate for all situations, particularly when a negative transformed total strain rate, ε*, is used and the resultant F_{en} value would exceed 2.532.

The applicant stated that it applied a zero term for transformed dissolved oxygen content, O*, making the third term (0.101S*T*O* ε*) of Equation 6.5b from NUREG/CR-6583 equal to zero for its plant-specific environmental fatigue analyses, since the dissolved oxygen content was assumed to be less than 0.05 ppm. The staff noted that the applicant's response to RAI 4.3-08, Part 3 further explains this assumption. The staff's review of RAI 4.3-08, Part 3 is documented below, in SER Section 4.3.7.2. Furthermore, a conservative value of zero was used for the second term (0.00124T) in Equation 6.5b. The applicant stated that the statement, "F_{en} is maximized when these two terms are set equal to zero" is not accurate for analyses other than its plant-specific environmental fatigue analyses. The staff finds that setting the second term (0.00124T) in Equation 6.5b to zero is acceptable because it yields a larger F_{en} factor, which is more conservative. The staff noted that the response to RAI 4.3-05 (Part 3), dated July 13, 2010, was amended to remove the statement, "F_{en} is maximized when these two terms are set equal to zero" and finds this acceptable because the statement is not accurate for all situations of transformed dissolved oxygen content, transformed total strain rate, transformed temperature, and transformed sulfur content.

In its response to Part 3, dated December 21, 2010, the applicant clarified that during Modes 1 (Power Operations) and 2 (Startup), where the RCS is greater than or equal to 177 °C (350 °F) and reactivity condition (K_{eff}) is greater than 0.99, the dissolved oxygen concentrations are always less than 0.05 ppm (50 ppb), specifically, less than 0.005 ppm (5 ppb) as determined from the RCS quarterly chemistry data since 2000. The applicant stated that the reason for the extremely low dissolved oxygen levels is due to the RCS environment containing a hydrogen concentration of a minimum of 25 cc/kg (cubic centimeters per kilogram), as specified for Westinghouse PWRs to keep the oxygen level in the RCS below the limit of detection (5 ppb). The applicant stated that it had this specification limit of RCS hydrogen imposed since original start-up of the units. The staff finds it reasonable, during Modes 1 and 2, since the applicant has operated with a minimum of 25 cc/kg of RCS hydrogen, that dissolved oxygen was always less than 0.05 ppm (50 ppb), specifically, less than 0.005 ppm (5 ppb) since original start-up of the units.

The staff reviewed Equation 6.5b for low-alloy steels from NUREG/CR-6583 and noted that the transformed temperature, T*, is set to zero when the RCS temperature is less than 150 °C (302 °F), which negates the contribution from dissolved oxygen in this equation. The applicant stated that any dissolved oxygen values exceeding 0.05 ppm (50 ppb) during Mode 5 (Cold Shutdown – RCS temperature less than 93 °C (200 °F)) and Mode 6 (Refueling – RCS temperature less than 60 °C (140 °F)) do not contribute to EAF due to the low RCS temperatures. The staff finds that the transformed oxygen content parameter, O*, in Equation 6.5b can be ignored in Modes 5 and 6 because the RCS temperature during these modes does not exceed the threshold of 150 °C (302 °F) described in NUREG/CR-6583, therefore, setting the term "0.101S*T*O* ε*" equal to zero.

The applicant stated that there are possible short periods of time where the RCS dissolved oxygen levels can exceed 0.05 ppm, while the RCS temperatures exceed 150 °C (302 °F) for carbon and low-alloy steel. These short periods of time are during Mode 3 (Hot Standby – RCS temperature greater than 177 °C (350 °F) and K_{eff} is less than 0.99) and Mode 4 (Hot Shutdown – RCS temperature greater than 93 °C (200 °F) but less than 177 °C (350 °F) and K_{eff} is less than 0.99). The applicant stated that during the time when the RCS is heating from 150 °C (302 °F) (Mode 4) to 177 °C (350 °F) (Mode 3), or cooling from 177 °C (350 °F) (Mode 3) to 150 °C (302 °F) (Mode 4), the RCS dissolved oxygen levels could exceed 0.05 ppm (50 ppb), but are less than or equal to 0.10 ppm (100 ppb). Furthermore, the oxygen control is attained through hydrazine addition to the primary system. The applicant stated that the short periods of time are less than 24 hours per plant heatup and are less than 8 hours per plant cooldown.

The staff noted that the projected number of heatups and cooldowns for Unit 1 are 133 and 133, respectively, and 157 and 155 for Unit 2, respectively. The applicant stated that for additional conservatism, the 40-year NSSS design specification of 200 heatups and 200 cooldowns is multiplied by a time period of 24 hours for the heatup event and 8 hours for the cooldown event, which resulted in 6,400 hours. Furthermore, the projected effective full power hours for each unit is obtained by multiplying the effective full power years of 50 by 8,760 hours in a year, or 438,000 hours. The applicant determined that the percentage of time that the RCS temperature will be heating from 150 °C (302 °F) to 177 °C (350 °F), and cooling from 177 °C (350 °F) to

150 °C (302 °F) is less than 1.5 percent of the total operating time. The applicant determined an adjusted F_{en} value, which considers the dissolved oxygen level effect during Mode 3 and Mode 4, and noted that it results in a 0.4 percent increase in the CUF_{EAF} for the Units 1 and 2 reactor vessel inlet nozzles which are fabricated from low-alloy steel.

The staff finds that the short periods of time when the dissolved oxygen levels can exceed 0.05 ppm does not have a significant impact to the overall F_{en} value because the duration of time that both units operate with dissolved oxygen levels in excess of 0.05 ppm will conservatively be 1.5 percent of the total operating time after 60 years of operation and the resultant increase in F_{en} value is approximately 0.4 percent, which is negligible. The staff noted that this is applicable for both carbon and low-alloy steel components.

The applicant stated that it has not changed the chemistry control with regards to oxygen control in the RCS when the temperature is greater than 150 °C (302 °F) since original plant start-up, therefore, the values observed in the past 10 years (2000 to 2010) are representative of past operations. Furthermore, it will continue to and is committed to maintain its primary water chemistry, including the previously discussed limitations on dissolved oxygen, through the Water Chemistry Program, which incorporates Electric Power Research Institute (EPRI) guidelines.

Based on its review, the staff finds the applicant's response to RAI 4.3-8, Parts 2 and 3 acceptable because: (1) the applicant confirmed that it has always maintained dissolved oxygen levels less than 0.05 ppm since initial plant start-up during Modes 1 and 2; (2) the impact of dissolved oxygen levels greater than 0.05 ppm but less than or equal to 0.10 ppm, during Modes 3 and 4, on the F_{en} value are negligible, as described above; (3) the impact of dissolved oxygen levels greater than 0.05 ppm during Modes 5 and 6, when the temperature is less than 150 °C (302 °F), do not need to be considered, as described above; (4) the applicant will continue to maintain its primary water chemistry during the period of extended operation; and (5) the applicant justified that a F_{en} value of 2.532 for low-alloy steel components is conservative, based on its plant-specific operating conditions. The staff's concerns described in RAI 4.3-05 and RAI 4.3-08, Parts 2 and 3 are resolved, and this part of Open Item OI 4.3.4.2-1 is closed.

The staff also noted that, in LRA Section 4.3.7, the applicant stated that the fatigue analyses for the NUREG/CR-6260 sample locations have been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The staff noted, however, that LRA Section B.3.1.1 indicated that the Metal Fatigue of Reactor Coolant Pressure Boundary Program will be enhanced to address the effects of the reactor coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant, as identified in NUREG/CR-6260. Therefore, it was not evident to the staff whether the applicant had chosen to use its Metal Fatigue of Reactor Coolant Pressure Boundary Program as the basis for accepting the EAF analysis TLAA, in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii), and for managing the effects of environmental fatigue on the intended functions of the applicant's NUREG/CR-6260 sample locations during the period of extended operation. Therefore, in a letter dated June 14, 2010, the staff issued RAI 4.3-06 requesting that the applicant clarify: (1) how the Metal Fatigue of Reactor Coolant Pressure Boundary Program would be used to monitor the effects of the reactor coolant environment on the metal fatigue analyses for the plant's critical NUREG/CR-6260 locations, and (2) whether the AMP would be used to disposition the EAF analyses for these components in accordance with the TLAA acceptance criterion in 10 CFR 54.21(c)(1)(iii).

In its response dated July 13, 2010, the applicant stated that the Metal Fatigue of Reactor Coolant Pressure Boundary Program addresses the effects of the reactor coolant environment on component fatigue life on fatigue limiting locations. The applicant further stated that it would revise site procedures to include the effects of the reactor coolant environment for each of the six locations discussed in LRA Section 4.3.7 in a periodic fatigue monitoring report. In addition, the applicant modified the LRA to indicate that the aging of these fatigue limiting locations will be managed by 10 CFR 54.21(c)(1)(iii) using the Metal Fatigue of Reactor Coolant Pressure Boundary Program.

Based on its review, the staff finds the applicant's response to RAI 4.3-06 acceptable because the Metal Fatigue of Reactor Coolant Pressure Boundary Program monitors the transients to ensure that the CUF considering environmental effects remains below the design basis of 1.0. The staff finds this an appropriate approach because the applicant has modified its LRA to indicate that the aging of these fatigue limited locations is managed in accordance with 10 CFR 54.21(c)(1)(iii). The staff's concern described in RAI 4.3-06 is resolved.

4.3.7.3 UFSAR Supplement

The applicant provided a UFSAR supplement summary description of its TLAA evaluation of EAF analyses in LRA Section A.4.3.7. On the basis of its review of the UFSAR supplement, consistent with SRP-LR Section 4.3.3.3, the staff concludes, with the closure of Open Item OI 4.3.4.2-1, that the summary description of the applicant's actions to address EAF analyses is adequate.

4.3.7.4 Conclusion

On the basis of its review, consistent with SRP-LR Section 4.3.3.2, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of reactor coolant environment on component fatigue life will be adequately managed for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.