SECTION 4

TIME-LIMITED AGING ANALYSES

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4 Time-Limited Aging Analyses

4.1 Identification of Time-Limited Aging Analyses

This section addresses the identification of time-limited aging analyses (TLAAs). The applicant discusses the TLAAs in license renewal application (LRA) Sections 4.2 through 4.7. The staff's review of the TLAAs can be found in Sections 4.2 through 4.7 of this safety evaluation report (SER).

The TLAAs are certain plant-specific safety analyses that are based on an explicitly assumed 40-year plant life. Pursuant to Section 54.21(c)(1) of Title 10 of the *Code of Federal Regulations* (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 addition, pursuant to 10 CFR 54.21(c)(2), an applicant must provide a list of plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs. For any such exemptions, the applicant must provide an evaluation that justifies the continuation of the exemptions for the period of extended operation.

4.1.1 Summary of Technical Information in the Application

The applicant evaluated calculations for the Fort Calhoun Station, Unit 1 (FCS) against the six criteria specified in 10 CFR 54.3 to identify the TLAAs. The applicant indicated that calculations that meet the six criteria were identified by searching the current licensing basis (CLB), which includes the updated safety analysis report (USAR), design basis documents, the Statements of Consideration for 10 CFR Part 54, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, " dated July 2001 (SRP-LR), and Nuclear Energy Institute (NEI) 95-10. The applicant listed the following TLAAs in Table 4.1-1 of the LRA:

- reactor vessel neutron embrittlement; including analyses for upper shelf energy, pressurized thermal shock, low-temperature overpressure protection (LTOP) power operated relief valve (PORV) setpoints, and pressure-temperature limits
- metal fatigue; including analysis of ASME Section III Class 1 vessels, RCS piping, and Class II and III components
- environmental equipment qualification
- concrete containment prestress
- containment liner and penetration sleeve fatigue
- reactor coolant pump flywheel fatigue
- leak-before-break analysis
- high-energy line break

Pursuant to 10 CFR 54.21(c)(2), the applicant stated that no exemptions granted under 10 CFR 50.12 that were based on a TLAA, as defined in 10 CFR 54.3, were identified.

4.1.2 Staff Evaluation

In LRA Section 4.1, the applicant identified the TLAAs applicable to FCS and discussed exemptions based on TLAAs. The staff reviewed the information to determine whether the applicant provided adequate information to meet the requirements of 10 CFR 54.21(c)(1) and 10 CFR 54.21(c)(2).

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

- involve systems, structures, and components within the scope of license renewal, as delineated in Section 54.4(a)
- consider the effects of aging
- involve time-limited assumptions defined by the current operating term, for example, 40 years
- were determined to be relevant by the applicant in making a safety determination
- 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 Section 54.4(b)
- are contained or incorporated by reference in the CLB

The applicant listed the TLAAs applicable to FCS in Table 4.1-1 of the LRA. Tables 4.1-2 and 4.1-3 in the SRP-LR identify potential TLAAs determined from the review of other license renewal applications. In RAI 4.1-1, the staff requested that the applicant discuss whether there are any calculations or analyses at FCS that address the topics listed in Tables 4.1-2 and 4.1-3 of the SRP-LR and were not included in Table 4.1-1 of the LRA.

In its RAI response dated December 12, 2002, the applicant indicated that documentation existed for two topics listed in the SRP-LR that were not identified as TLAAs at FCS. The first topic is metal corrosion allowance. The applicant indicated that corrosion allowances were made consistent with the requirements of the design codes; however, there are no discrete analyses related to metal corrosion allowances meeting the criteria of 10 CFR 54.3(a). The staff concludes that, because the applicant found no analyses related to metal corrosion allowances that meet the TLAA criteria, the applicant’s response regarding the corrosion allowance is acceptable.

The second topic is the polar crane fatigue analysis. The applicant indicated that the crane was purchased using a specification based on Electric Overhead Crane Institute Standard (EOCI)-61, which does not address fatigue failure. The applicant indicated that it also reviewed the polar crane against the fatigue criteria of American Society of Mechanical Engineers (ASME) NOG-1 and Crane Manufacturers Association of America (CMAA)-70 and concluded that the number of anticipated cycles is well below the limits which would require fatigue analyses in accordance with NOG-1 and CMAA-70. The applicant’s assessment is consistent with assessments performed by other license renewal applicants. The staff considers that the polar crane evaluation meets the definition of a TLAA in accordance with the criteria of 10 CFR 54.3(a). However, the staff agrees that the number of anticipated cycles will be well below the limits which would require fatigue analyses. On this basis, the staff finds that the applicant’s evaluation of the polar crane, as discussed above, is acceptable and therefore the crane need not be evaluated as a TLAA.

Following the issuance of the SER with open items, the staff identified an additional TLAA regarding a repair of a temperature nozzle in the pressurizer lower shell. The staff informed the applicant of this additional TLAA, and the associated Open Item 4.7.4-1, by letter dated May 15, 2003. By letter dated July 7, 2003, the applicant responded to the open item. A summary of the open item and it's resolution is provided in Section 4.7.4 of this SER.

10 CFR 54.21(c)(2) requires an applicant to provide a list of all exemptions granted under 10 CFR 50.12 which are determined to be based on a TLAA, and an evaluation and justification for continuation through the period of extended operation. In the LRA, the applicant stated that it performed a search of the FCS electronic docket and each exemption was reviewed for TLAA applicability. No TLAA-based exemptions were identified. On the basis of the information provided by the applicant with regard to the process used to identify TLAA-based exemptions, and the results of the applicant's search, the staff finds that the applicant has found no TLAA-based exemptions which would require justification for continuation through the period of extended operation to satisfy 10 CFR 54.21(c)(2)

4.1.3 Conclusions

On the basis of its review, including its identification of the additional TLAA discussed in Section 4.7.4 of this SER, the staff concludes that all TLAAs have been identified for FCS, as required by 10 CFR 54.21(c)(1), and has confirmed that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA, as required by 10 CFR 54.21(c)(2).

4.2 Reactor Vessel Neutron Embrittlement

The applicant has identified four analyses affected by irradiation embrittlement that have been identified as TLAAs. These analyses are discussed in Sections 4.2.1 through 4.2.4 of the LRA. The analyses identified as TLAAs are:

- pressure/temperature (P/T) curves
- low-temperature overpressure protection (LTOP) power-operated relief valve (PORV) setpoints
- pressurized thermal shock (PTS)
- reactor vessel upper shelf energy

Neutron embrittlement is a significant aging mechanism for all ferritic materials that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) at the end of the period of extended operation. The relevant calculations use predictions of the cumulative damage to the reactor vessel from neutron embrittlement and were originally based on the 40-year expected life of the plant. The reactor pressure vessel contains the core fuel assemblies and is made of thick steel plates that are welded together. Neutrons from the fuel in the reactor irradiate the vessel as the reactor is operated and change the material properties of the steel. The most pronounced and significant changes occur in the material property known as fracture toughness. Fracture toughness is a measure of the resistance to crack extension in response to stresses. A reduction in this material property due to irradiation is referred to as embrittlement. The largest amount of embrittlement usually occurs at the section of the vessel's wall closest to the reactor fuel referred to as the vessel's beltline. FCS uses a "low leakage" PWR core design that reduces the number of neutrons that reach the vessel wall and thus limits the vessel's embrittlement. However, the rate at which the vessel's steel embrittles also depends on its chemical composition. The amounts of two elements in the steel, copper and nickel, are the most important chemical components in determining how sensitive the steel is to neutron irradiation.

4.2.1 Plant Heatup/Cooldown (P/T) Curves and LTOP PORV Setpoints

4.2.1.1 Summary of Technical Information in the Application

The current P/T analyses are valid out to 40 effective full power years (EFPY), which extends beyond the current operating license period but not to the end of the period of extended operation. LTOP limits are considered as part of the calculation of P/T curves. The technical specifications will continue to be updated as required by either Appendix G or H of 10 CFR Part 50, or as operational needs dictate. This will assure that operational limits remain valid for current and projected cumulative neutron fluence levels.

4.2.1.2 Staff Evaluation

In response to RAI 4.2-1, the applicant indicated that the NRC has approved the revised limits and issued Technical Specification Amendment 207 for FCS. Using the methodology approved with the issuance of Technical Specification Amendment 207, the applicant has projected the P/T and LTOP limits to the end of the period of extended operation and determined that the reactor pressure vessel can be operated with the projected P/T and LTOP limits. The technical specifications will continue to be updated as required by either Appendix G or H of 10 CFR Part 50, or as operational needs dictate. This will assure that operational limits remain valid for current and projected cumulative neutron fluence levels. Since the technical specifications will continue to be updated as required by either Appendix G or H of 10 CFR Part 50, additional analysis at this time is not required. On this basis, the staff concludes that the applicant has a process for updating the plant heatup/cooldown (P/T) curves and LTOP PORV setpoints at FCS for the period of extended operation, which satisfies 10 CFR 54.21(c)(1)(iii) and Appendices G and H of 10 CFR Part 50.

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.2.2 Pressurized Thermal Shock

10 CFR 50.61 provides the fracture toughness requirements protecting the reactor vessels (RVs) of PWRs against the consequences of pressurized thermal shock (PTS). Licensees are required to perform an assessment of the reactor vessel materials' projected values of the PTS reference temperature, RT_{PTS}, through the end of their operating license. The rule requires each licensee to calculate the end-of-life RT_{PTS} value for each material located within the beltline of the reactor pressure vessel. The RT_{PTS} value for each beltline material is the sum of the unirradiated nil ductility reference temperature (RT_{NDT}) value, a shift in the RT_{NDT} value caused by exposure to high-energy neutron irradiation of the material (i.e., ΔRT_{NDT} value), and an additional margin value to account for uncertainties (i.e., M value). 10 CFR 50.61 also provides screening criteria against which the calculated RT_{PTS} values are to be evaluated. For reactor vessel beltline base-metal materials (forging or plate materials) and longitudinal (axial) weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 270 °F. For reactor vessel beltline

circumferential weld materials, the materials are considered to provide adequate protection against PTS events if the calculated RT_{PTS} values are less than or equal to 300 °F. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of the shift in the RT_{NDT} value caused by exposure to high-energy neutron irradiation and the margin value to account for uncertainties. In RG 1.99, the shift in the RT_{NDT} value caused by exposure to high-energy neutron irradiation is the product of a chemistry factor and a fluence factor. The fluence factor is dependent upon the neutron fluence, and the chemistry factor may be determined from surveillance material or from the tables in RG 1.99. If the reactor vessel beltline material is not represented by surveillance material, its chemistry factor and the shift in the RT_{NDT} value caused by exposure to high-energy neutron irradiation may be determined using the methodology documented in Position 1.1 and the tables in RG 1.99. The chemistry factor determined from the tables in RG 1.99 depends upon the amount of copper and nickel in the beltline. If the reactor vessel beltline material is represented by surveillance material, its chemistry factor may be determined from the surveillance data using the methodology documented in Position 2.1 of RG 1.99.

4.2.2.1 Summary of Technical Information in the Application

The applicant indicates that it has completed the projected RT_{PTS} calculation, and the NRC has concluded that RT_{PTS} values for the FCS reactor vessel beltline materials will remain below the 10 CFR 50.61 PTS screening criteria until 2033, the end of the period of extended operation. Therefore, the analyses have been projected to the end of the period of extended operation.

4.2.2.2 Staff Evaluation

In a license amendment dated August 3, 2000, and letters dated November 17, 2000, and February 14, 2001, the applicant provided RT_{PTS} analyses for the materials in the FCS reactor vessel. The August 3, 2000, license amendment contains report CEN-636, "Evaluation of Reactor Vessel Surveillance Data Pertinent to the Fort Calhoun Reactor Vessel Beltline Materials." Table 10 in CEN-636, Revision 2, provides the chemistry factor and the predicted RT_{PTS} value through 2033 for each plate and weld in the FCS reactor vessel beltline. Many of the materials' RT_{PTS} values are dependent upon surveillance data, which could affect their RT_{PTS} value.

In response to RAI B.1.7-1, the applicant provided for each beltline material the projected neutron fluence at the end of the period of extended operation and the neutron flux assumed for future core loadings. The fluence values were obtained from WCAP-15443, Revision 0, which was reviewed and approved by the NRC for Technical Specification Amendments 197 and 199. The overall exposure evaluation methodology is based on guidance provided in Draft Regulatory Guide DG-1053 and makes use of the latest ENDF/B-VI neutron transport and dosimetry cross-sections included in the BUGLE-93 library. This fluence report also describes how the fluence was calculated and includes the benchmark of the fluence model and the azimuthal distribution for fluence across the reactor vessel. The fluence values for each material conservatively correspond to the end of fuel Cycle 41 (September 2033). The staff has reviewed the methodology documented in WCAP-15443 and endorses its use for calculating the neutron fluence to be used in the PTS and the reactor vessel upper shelf energy analyses.

In response to RAI B.1.7-1, the applicant also explained how the reactor vessel integrity program (RVIP) will monitor future core loadings to ensure that no beltline material will exceed the PTS screening criteria in 10 CFR 50.61. The applicant indicates that compliance with 10 CFR 50.61 is monitored as part of the program basis document for the RVIP. This program is administered by the FCS Design Engineering-Nuclear Engineering Department. The Nuclear Engineering Department also performs core reload analyses in-house, including core design. During core loading development, core patterns are quantitatively evaluated to ensure that neutron flux to the limiting 3-410 welds is maintained approximately the same as that of Cycle 15, which formed the basis of the fluence analysis. This is done by summing the peripheral fuel assembly relative power densities multiplied by weighting factors derived from the fluence analysis adjoint flux solution. Thus, values from a new fuel cycle can be compared to that of Cycle 15 to determine if there has been a net increase or decrease, with a goal of having a time average value the same as Cycle 15. Periodic updates of the fluence analysis are planned. RT_{PTS} is also tracked on an ongoing basis.

According to 10 CFR 50.61(b), “Requirements,” each licensee is required to update its PTS assessment whenever there is a significant change in the projected value of RT_{PTS}. Therefore, if the applicant’s core loading pattern should deviate from that assumed in the PTS analysis, the applicant would be required to provide the staff with an updated assessment.

Using the neutron fluences contained in RAI B.1.7-1 and the chemical composition data and surveillance data reported in CEN-636, Revision 2, the staff calculated the predicted RT_{PTS} through 2033. The results of the staff’s analysis are documented in Table 4.2.2 below. In the staff’s analysis, weld 3-410 A/C, which was fabricated using tandem electrodes with weld wire heat numbers 12008 and 13253, was projected to be closest to the PTS screening (1 ^{o}F below the PTS screening criteria) at the expiration of extended operation in 2033. Surveillance data were used in the analysis to determine the chemistry factor for plate heat number A1768-1 and welds fabricated using weld wire heat numbers 12008, 13253, and 27204. All other beltline materials did not have surveillance material. Therefore, the chemistry factors were determined using the tables in RG 1.99. On the basis of its evaluation, the staff confirmed that all beltline materials will be below the PTS screening criterion at the expiration of extended operation in 2033 to satisfy 10 CFR 54.21(c)(1)(ii).

On this basis, the staff concludes that the applicant has adequately evaluated PTS at FCS for the period of extended operation by projecting the PTS analyses to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

Table 4.2.2

Plate or Weld Identification	Plate Heat No. or Weld Heat No.	PTS Screening Criteria (°F)	Method of Predicting Chemistry Factor	Predicted RT_{PTS} through 2033 (°F)
Plate D4802-1	C2585-3	270	Table	143
Plate D4802-2	A1768-1	270	Surveillance	131
Plate D4802-3	A1768-2	270	Table	131
Plate D4812-1	C3213-2	270	Table	144
Plate D4812-2	C3143-2	270	Table	120
Plate D4812-3	C3143-3	270	Table	120
Weld 2-410 A/C	51989	270	Table	120
Weld 3-410 A/C	12008/13253(T)[1]	270	Table	269
Weld 3-410 A/C	13253 (T)[1]	270	Surveillance	225
Weld 3-410 A/C	12008/27204 (T)[1]	270	Surveillance	245
Weld 3-410 A/C	27204 (T)[1]	270	Surveillance	256
Weld 9-410	20291	300	Table	260

[1] T indicates welds were fabricated using weld wires in tandem

4.2.3 Reactor Vessel Upper Shelf Energy

The NRC regulations that provide screening criteria for the upper shelf energy (USE) are in 10 CFR Part 50, Appendix G. Appendix G requires that reactor vessel beltline materials have Charpy USE values in the transverse direction for the base metal and along the weld for the weld material, according to the ASME Code, of no less than 75 ft-lb (102 J) initially, and must maintain Charpy USE values throughout the life of the vessel of no less than 50 ft-lb (68 J). However, Charpy USE values below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. RG 1.99 provides an expanded discussion regarding the calculations of Charpy USE values and describes two methods for determining Charpy USE values for reactor vessel beltline materials, depending on whether a given reactor vessel beltline material is represented in the plant's reactor vessel material surveillance program (i.e., 10 CFR Part 50, Appendix H program). If surveillance data are not available, the Charpy USE is determined in accordance with Position 1.2 in RG 1.99. If two or more credible surveillance data (as defined in Section B, "Discussion," of RG 1.99) are available, the Charpy

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[1] T indicates welds were fabricated using weld wires in tandem

USE should be determined in accordance with Position 2.2 in RG 1.99. These methods refer to Figure 2 in RG 1.99, which indicates that the percentage drop in Charpy USE is dependent upon the amounts of copper and the neutron fluence. Since the analyses performed in accordance with Appendix G to 10 CFR Part 50 are based on a flaw with a depth 1/4 through wall, the neutron fluence used in the Charpy USE analysis is the neutron fluence at the 1/4T (thickness) depth location.

4.2.3.1 Summary of Technical Information In the Application

The applicant indicates that preliminary calculations have shown that the vessel beltline Charpy USE for the limiting weld will be approximately 54.6 ft-lbs based on Position 1.2 of RG 1.99. This value remains above the regulatory approved minimum of 50 ft-lbs through the period of extended operation. The existing Appendix G analysis will be finalized and formally revised to reflect that it bounds the minimum approved fluence value at the end of plant life. However, the analyses had not been projected to the end of the period of extended operation at the time that the LRA was submitted for staff review.

4.2.3.2 Staff Evaluation

In response to RAI 4.2-2, the staff requested that the applicant provide for each beltline material (a) the projected peak neutron fluence at a depth of 1/4T at the end of the period of the extended operation, (b) the unirradiated Charpy USE, (c) the amount of copper, (d) the Charpy USE at the end of the period of extended operation, (e) the method of determining the decrease in Charpy USE at the end of the period of extended operation, and (f) the impact of surveillance data on the Charpy USE analysis. The applicant did not provide the impact of surveillance data representing welds fabricated using tandem electrodes with weld wire heat number 13253 and welds fabricated using tandem electrodes with wire heat number 27204. The applicant must provide all the surveillance data applicable to its plant and must determine the impact of all the surveillance data on the Charpy USE analysis. By letter dated February 20, 2003, the staff issued POI-13(a) requesting this information. By letter dated March 14, 2003, the applicant provided the requested information.

In response to this POI, the applicant performed a revised analysis and documented the results of its analysis in Table A.3.1.4-1,"Fort Calhoun Station Upper Shelf Energy Data for Operation to 48 EFPY." In this analysis, the Charpy USE for each reactor vessel beltline material was determined in accordance with position 1.2 of RG 1.99. The lowest predicted Charpy USE at 48 EFPY was 54.6 ft-lb. Position 1.2 states that the percent drop of Charpy USE is a function of the percent copper and neutron fluence, as indicated in Figure 2 of RG 1.99. The applicant has determined the percent drop of Charpy USE for each weld wire heat used in the FCS beltline weld based on: (a) the best-estimate copper for each heat of weld wire; (b) the projected neutron fluence at the 1/4-T depth location at 48 EFPY; and (c) Figure 2 of RG 1.99. The projected Charpy USE at the 48 EFPY is the difference between the unirradiated Charpy USE and the percent drop in Charpy USE. The staff has performed an independent evaluation in accordance with the methodology in RG 1.99 and confirmed the projected Charpy USE values at 48 EFPY for the FCS reactor vessel beltline materials.

In addition, the applicant provided in Table 4.2-2-2 the results of its analysis of irradiated Charpy USE surveillance weld data from other plants that have surveillance data that is applicable to the FCS reactor vessel beltline welds. The data was from surveillance capsules from DC Cook Unit 1, Diablo Canyon Unit 1, Salem Unit 2, and Mihama Unit 1. The surveillance welds for DC Cook Unit 1 and Salem Unit 2 were fabricated using the same weld wire heat number (heat number 13253) as used in the FCS reactor vessel beltline Weld Number 3-410. Weld Number 3-410 also utilized weld wire from heat numbers 27204 and 12008/27204. The Diablo Canyon Unit 1 surveillance weld was fabricated from weld wire heat number 27204 and the Mihama Unit 1 surveillance weld was fabricated using tandem weld wires from heat numbers 12008 and 27204. FCS weld Number 3-410 was projected to have a neutron fluence at 1/4T depth at 48 EFPY of 1.62 x 10^{19} n/cm^2. The applicant indicated that the capsule with the highest neutron fluence was from the third Mihama Unit 1 surveillance capsule with a neutron fluence of 2.1 x 10^{19} n/cm^2. The surveillance weld samples from the third Mihama surveillance capsule demonstrated a Charpy USE value of 61 ft-lb. Surveillance weld samples from all other applicable capsules demonstrated higher Charpy USE values. The applicant concluded that the projected values of Charpy USE for the FCS beltline welds, ranging from 54.6 to 66 ft-lb, are consistent with the values of Charpy USE measured for the surveillance materials from Diablo Canyon, Mihama, DC Cook, and Salem.

The staff has performed an independent analysis of the surveillance data from DC Cook Unit 1, Diablo Canyon Unit 1, Salem Unit 2, and Mihama Unit 1 to determine whether the methodology described in position 1.2 of RG 1.99 is conservative for the weld wires used in fabricating the FCS reactor vessel beltline. Figure 2 of RG 1.99 describes the relationship between neutron fluence and percent drop in Charpy USE as linear on a log-log scale for a specified amount of copper. Hence, the change in Charpy USE with neutron fluence for each beltline heat of weld wire would be described as a line on the log-log plot in Figure 2 and determined by the copper content of each weld wire heat. The staff compared the surveillance data from DC Cook Unit 1, Diablo Canyon Unit 1, Salem Unit 2, and Mihama Unit 1 with the values for the corresponding line on the log-log plot in Figure 2 for the weld wires used in the beltline welds. All of the surveillance data, except for the Salem 2 and DC Cook Unit 1 data, were on or below the lines on the log-log plot in Figure 2 for the corresponding weld wire heats. For the Diablo Canyon Unit 1 and Mihama Unit 1 surveillance data, where the surveillance data is on or below the lines on the log-log plot in Figure 2 for the corresponding beltline heat of weld wire, the surveillance data indicates that the methodology represented by Figure 2 and position 1.2 of RG 1.99 is conservative and the values of Charpy USE determined using this methodology are acceptable. Since surveillance data from Salem Unit 2 and DC Cook Unit 1 exceeded the line representing the FCS weld wire heat 13253, the position 1.2 methodology would be non-conservative for this heat of weld wire. The staff determined the impact of this data using the methodology specified in position 2.2 of RG 1.99. Position 2.2 specifies that the percent drop in Charpy USE may be obtained by plotting the surveillance data on Figure 2 of the RG and fitting the data with a line drawn parallel to the existing lines as the upper bound of all the data. Using this methodology, the projected Charpy USE for heat number 13253 weld wire would be reduced from 66 ft-lb to 61.6 ft-lb at 48 EFPY. Since this value was determined in accordance with position 2.2 of RG 1.99, and is greater than 50 ft-lb, the welds in FCS that were fabricated using weld wire heat 13253 will have Charpy USE greater than 50 ft-lb at 48 EFPY.

FCS has surveillance weld metal and plate material being irradiated within its reactor vessel. The weld metal was prepared using a weld wire heat number that was not used in the FCS reactor vessel beltline; therefore, it does not represent any FCS reactor vessel beltline weld. The surveillance plate material was removed from beltline plate D4802-2, heat number A1768-1. Therefore, the decrease in Charpy USE observed on this surveillance plate would be representative of the decrease in Charpy USE that would be expected for the FCS beltline plate. Using the position 2.2 methodology in RG 1.99, the staff determined that the Charpy USE plate surveillance data would result in a Charpy USE at 48 EFPY for plate D4802-2 of 84 ft-lb. Position 2.2 was used for the evaluation of the plate, because the surveillance plate was removed from the beltline plate and they have equivalent chemical compositions.

Based on the staff and applicant evaluation of surveillance data and using the methodology from RG 1.99, all FCS reactor vessel beltline materials are projected to have Charpy USE at 48 EFPY greater than 50 ft-lb and will meet the screening criteria for Charpy USE in Appendix G, 10 CFR Part 50 at the expiration of the extended license. This completes the staff evaluation of Reactor Vessel USE and resolves POI-13(a).

The USAR Supplement did not contain the Charpy USE analysis that was performed in response to RAI 4.2-2. Since this analysis applies through the end of the period of extended operation, the applicant must revise the USAR Supplement to include the results of the Charpy USE performed in response to RAI 4.2-2. By letter dated February 20, 2003, the staff issued POI-13(b) requesting the applicant to revise the USAR Supplement. By letter dated March 14, 2003, the applicant provided a revised USAR Supplement Section A.3.1.4, which incorporated the results of the Charpy USE analysis. The staff reviewed the revised USAR Supplement and finds that it is an adequate description of the Charpy USE TLAA. POI-13(b) is resolved.

On this basis, the staff concludes that the applicant has adequately evaluated the RV USE at FCS for the period of extended operation by projecting the analysis to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.2.4 Conclusions

The staff has reviewed the TLAAs regarding the maintenance of acceptable Charpy USE levels for the FCS RV materials and the ability of the FCS RV to resist failure during postulated PTS events. On the basis of this evaluation, the staff concludes that the applicant's TLAAs for Charpy USE and PTS meet the respective requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.61, for the FCS RV beltline materials, as evaluated to the end of the period of extended operation, and therefore satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for 60 years of operation.

The staff will evaluate the P/T limit curves and the LTOP PORV setpoints, as described in LRA Section A.3.1.1, for the period of extended operation upon submittal by the applicant. The staff's review of the P/T limit curves, when submitted, will ensure that the operation of the RCS for FCS will be done in a manner that ensures the integrity of the RCS during the period of extended operation and that the curves, when submitted, will satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for the period of extended operation.

The staff has also reviewed the USAR Supplements for the P/T curves, LTOP PORV setpoints, PTS, and reactor vessel USE TLAAs, and finds that they provide adequate descriptions of the TLAAs, as required by 10 CFR 54.21(d).

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.

4.3.1 Summary of Technical Information in the Application

The applicant discussed the design requirements for components of the RCS at FCS. The RV and major RCS components were designed to the ASME Boiler and Pressure Vessel Code, Section III requirements for Class A components. The reactor coolant loop piping and fittings were designed and fabricated in accordance with the requirements of United States of America Standard (USAS) B31.1, “Power Piping Code.” The reactor coolant loop attached piping was designed and fabricated in accordance with the requirements of USAS B31.7, “Draft Code for Nuclear Power Piping.” The fatigue analyses of both the reactor coolant loop and attached piping were performed in accordance with USAS B31.7.

The applicant listed the transients used in the design of RCS components in Section 4.3.1 of the LRA. The applicant indicated that it does not expect the number of design cycles for the transients that are counted to be exceeded during the period of extended operation. The applicant uses the fatigue monitoring program (FMP) to verify its conclusion. The FMP is discussed in Section B.2.4 of the LRA and evaluated in Section 3.0.3.8 of this SER. The applicant described the actions taken to address the issue of environmentally-assisted fatigue in Section 4.3.2 of the LRA.

The applicant describes its evaluation of the following fatigue-sensitive component locations:

- reactor vessel shell and lower head
- reactor vessel inlet and outlet nozzles
- pressurizer surge line elbow
- charging system nozzle
- safety injection system nozzle
- shutdown cooling system Class 1 piping

The applicant indicated that the evaluation found all locations were acceptable for the period of extended operation, with the exception of the pressurizer surge line. The applicant indicated that the pressurizer surge line will require further evaluation prior to the period of extended operation.

The applicant discussed the further evaluation of the pressurizer surge line in Section 4.3.3 of the LRA. The applicant indicated that the pressurizer surge line bounding locations will be included in the FMP. The applicant further indicated that actual operating data will be used to perform a reevaluation of the surge line prior to the period of extended operation.

The applicant discussed the evaluation of Class II and III components in Section 4.3.4 of the LRA. These components were designed to the requirements of USAS B31.1. USAS B31.1 specifies that a stress reduction factor be applied to the allowable thermal bending stress range if the number of full range cycles exceeds 7000. The applicant indicated that most piping systems within the scope of license renewal are only subject to occasional cyclic operation, and consequently, the analyses will remain valid during the period of extended operation. However, the applicant did indicate that the RCS hot leg sample line could exceed the 7000 cyclic limit during the period of extended operation and that it would be included in the FMP.

4.3.2 Staff Evaluation

As discussed in the previous section, components of the RCS at FCS were designed to the Class 1 requirements of the ASME Code, and the RCS piping was evaluated using the fatigue requirements of USAS B31.7. These requirements contain explicit criteria for the fatigue analysis of components. Consequently, the applicant identified the fatigue analysis of these components as TLAAs. The staff reviewed the applicant's evaluation of the RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for fatigue analysis of RCS components involves calculating the cumulative usage factor (CUF). The fatigue damage in the component caused by each thermal or pressure transient depends on the magnitude of the stresses caused by the transient. The CUF sums the fatigue damage resulting from each transient. The design criterion requires that the CUF not exceed 1.0. The applicant indicated that review of the FCS plant operating histories indicates that the number of cycles and severity of the transients assumed in the design of these components envelops the expected transients during the period of extended operation. In RAI 4.3-1, the staff requested that the applicant provide the following information for each of the transients described in Section 4.3.1 of the LRA:

- the current number of operating cycles and a description of the method used to determine the number of the design transients from the plant operating history

- the number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years

- a comparison of the design transients listed in the LRA with the transients monitored by the FMP described in Section B.2.4 of the LRA; identification of any transients listed in the LRA that are not monitored by the FMP and an explanation of why it is not necessary to monitor these transients

The applicant's December 19, 2002, response provided a table which lists the current cycle counts for the design transients. The applicant indicated that these cycles were recorded in accordance with plant Standing Order (SO) O-23 on a monthly basis. The applicant indicated that the pressure differential transients due to RCP stops and starts are not counted because the number specified (4000) is conservative. The applicant also identified several transients that are not counted under this procedure. These cycles involve power changes, operating pressure and temperature variations, and feedwater additions with the plant in hot standby conditions. The applicant indicated that these cycles will be conservatively estimated from a review of plant operating records to predict current cycles under the FMP. Once current number of cycles has been established, a review will be performed to determine if there is a potential for exceeding the allowable cycles and should be managed. If so, they'll be counted and managed by the FMP.

The applicant's response did not provide a cycle count for chemical and volume control system (CVCS) transients identified in LRA Section 4.3. In addition, Note 1 to the response to Item 1 implies that some transients may not be monitored by the FMP, whereas the response to Item 3 indicates that all transients will be monitored either directly or indirectly by the FMP. By letter dated February 20, 2003, the staff issued POI-13(c), requesting the applicant to provide additional clarification regarding how these transients are monitored by the FMP. Specifically, the applicant was requested to provide a cycle count for CVCS transients and clarify the difference between the Item 1, Note 1, and Item 3 responses.

By letter dated March 14, 2003, the applicant provided its response to POI-13(c). The applicant provided the current cycle counts for the CVCS transients identified in LRA Section 4.3. The applicant indicated that the cycle counts for some of the CVCS transients are based on gross estimates due to incomplete data logs. The applicant stated that a condition report (CR) would be generated to obtain a more accurate transient count prior to entry into the period of extended operation. The applicant indicated that transients associated with the regenerative heat exchanger isolation and loss of letdown are the most limiting transients with regard to thermal fatigue of the CVCS system. The applicant's statement is consistent with the results of the charging nozzle evaluation presented in NUREG/CR-6260. The applicant's response also indicates that the regenerative heat exchanger and loss of letdown transients are based on actual cycle counts. The staff notes that the estimated cycles for all the CVCS transients are well below the number of design transients and that the number of CVCS transients would not be expected to exceed the number of design cycles during the period of extended operation.

The applicant indicated that transients with low volume control tank level and boron dilution would not be counted by the FMP because they resulted in insignificant fatigue usage. The applicant also indicated that FCS is a base-loaded plant whereas several of the cycle estimates provided in item 1 of the December 19, 2002, response were based of the assumption of load following. These include the operating and power change cycles. The applicant indicated that the number of cycles of these transients is expected to be well below the number of design cycles for the period of extended operation. This is consistent with the results presented in NUREG/CR-6260. The staff concludes that the applicant has provided sufficient information to assure that the thermal design transients that are significant contributors to the fatigue usage of RCS components will be monitored by the FMP. Therefore, POI-13(c) is resolved.

In response to RAI 4.3.1-1, Item 3, submitted to the staff by letter dated December 19, 2003, the applicant indicated that all design basis transients will be included in the FMP. The applicant indicated that a program basis document (PBD) would be generated to capture both the current and increased scope of the FMP which includes incorporation of automated cycle counting and the analysis for environmentally-assisted fatigue. The applicant committed to complete the PBD and implement the enhanced FMP prior to the period of extended operation.

Section A.2.10 of the LRA provides the FMP USAR Supplement. The Supplement indicates that the automated cycle counting software "FatiguePro" will be used to monitor thermal fatigue of the components in the program. The Supplement also indicates that an FCS site-specific evaluation is being performed to address environmental fatigue and that appropriate program enhancements will be made prior to the period of extended operation. However, Section 4.3.2 of the LRA indicates that the environmental fatigue evaluations are complete. In RAI 4.3.2-2, the staff requested the applicant to address the apparent discrepancy.

The applicant's December 12, 2002, response to RAI 4.3.2-2 indicated that the environmental fatigue evaluations are complete, and the analysis shows that the surge line is the only location where the CUF may exceed 1.0 during the period of extended operation. The applicant further indicated that the environmental fatigue of the surge line will be included in the FMP. By letter dated February 20, 2003, the staff issued POI-13(d) requesting the applicant to revise the USAR Supplement to describe the completed environmental fatigue evaluation. By letter dated April 4, 2003, the applicant provided the requested USAR Supplement revision. Therefore, this part of POI-13(d) is resolved.

The applicant indicates that the FMP will continue during the period of extended operation and will assure that design cycle limits are not exceeded. The applicant's FMP tracks transients and cycles of RCS components that have explicit design transient cycles to assure that these components stay within their design basis. Generic Safety Issue (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:

> The results of the probabilistic analyses, along with the sensitivity studies performed, the iterations with industry (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 breaks as plants continue to operate. Thus, the staff concludes 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 Combustion Engineering (CE) plant for the effect of the environment on the fatigue life of the components. The applicant also indicated that the later environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," were considered in the evaluation. In RAI 4.3.2-1, the staff requested that the applicant provide the results of the usage factor evaluation for each of the six component locations listed in NUREG/CR-6260.

The applicant's December 19, 2002, response provided the calculated environmental usage factors for the six component locations listed in NUREG/CR-6260. The calculated usage factors are less than 1.0 for all components except for the surge line elbow. The applicant's response indicates that the usage factors for two components, the surge line elbow and the charging line nozzle, were based on anticipated cycles for a 60-year plant life consistent with Table 5-32 of NUREG/CR-6260. The statement in the applicant's response is not clear to the staff. By letter dated February 20, 2003, the staff issued POI-13(e), requesting that the applicant clarify that the evaluations are based on the number of anticipated cycles for 60 years of operation at FCS. The staff also requested the applicant to clarify that the number of cycles assumed in these evaluations is included in the FMP to provide assurance that the evaluations remain valid during the period of extended operation.

By letter dated March 14, 2003, the applicant responded to POI-13(e). The applicant indicated that the number of cycles assumed for the evaluation of the charging line nozzle will be included in the FMP basis document to assure that the CUF of 1.0 is not exceeded during the period of extended operation. The staff finds the applicant’s commitment to include the number of cycles used in the evaluation of the charging nozzle in the FMP acceptable. The applicant also indicated that further evaluation of the surge line elbow will be required prior to entry into the period of extended operation. The applicant will include this commitment in its USAR Supplement as discussed in its response to POI-13(d). The staff concludes the applicant’s commitments are sufficient to assure that the effect of the environment on the fatigue life of the charging nozzle and the surge line elbow will be adequately addressed during the period of extended operation. Therefore, POI-13(e) is resolved.

The results of the applicant’s evaluation are consistent with the results presented in NUREG/CR-6260 for an older vintage CE plant. NUREG/CR-6260 also identified the surge line elbow as the only component where the environmental usage factor may exceed 1.0 during the period of extended operation. The staff concludes that the applicant’s results are reasonable based on comparison with the results presented in NUREG/CR-6260.

Section 4.3.2 of the LRA contained a discussion of the proposed AMP to address fatigue of the FCS pressurizer surge line. The discussion indicated that the AMP will consist of an inspection program. The LRA also indicated that the results of the surge line inspections will be used to assess the appropriate approach for addressing environmentally-assisted fatigue of the surge lines. However, Section 4.3.3 of the LRA indicated that a reevaluation of the fatigue usage of critical areas of the surge line will be performed prior to the period of extended operation and that the bounding locations will be included in the FMP. In RAI 4.3.2-3, the staff requested that the applicant describe how the effect of the reactor water environment will be considered in the reevaluation of the critical areas of the surge line and how the results of this evaluation will be monitored by the FMP.

The applicant’s December 19, 2002, response indicated that the limiting surge line welds would be inspected prior to the period of extended operation. The applicant further indicated that the results of these inspections will be used to assess the appropriate approach for addressing environmentally-assisted fatigue of the surge lines. The applicant indicated that the approach would include one or more of the following four options.

1. further refinement of the fatigue analysis to lower the CUF(s) to below 1.0
2. repair of the affected locations
3. replacement of the affected locations
4. management of the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic nondestructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC)

The applicant committed that, if Option 4 were to be selected, it will provide the inspection details, including scope, qualification method, and frequency, to the NRC staff for review and approval prior to the period of extended operation. An AMP under this option would be a departure from the design basis CUF evaluation described in the USAR Supplement, and therefore would require a license amendment pursuant to 10 CFR 50.59. This was identified as Confirmatory Item 4.3.2-1.

By letter dated July 7, 2003, the applicant formalized this commitment. The staff finds this acceptable. Confirmatory Item 4.3.2-1 is closed.

The staff finds the applicant's proposed program an acceptable plant-specific approach to address environmentally-assisted fatigue during the period of extended operation in accordance with 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), this information needs to be added to the USAR Supplement. By letter dated February 20, 2003, the staff issued POI-13(d), requesting this addition to the USAR Supplement.

By letter dated April 4, 2003, the applicant provided the requested USAR Supplement revision. Therefore, this part of POI-13(d) is resolved.

Section 4.3.4 of the LRA contained a discussion of the analysis of Class II and III components at FCS. American National Standards Institute (ANSI) B31.1 requires that a reduction factor be applied to the allowable bending stress range if the number of full-range thermal cycles exceeds 7000. The LRA indicated that the USAS B31.1 limit of 7000 equivalent full-range cycles may be exceeded during the period of extended operation for the nuclear steam supply system (NSSS) sampling system and that the affected portions of the NSSS sampling system would be tracked by the FMP. In RAI 4.3.4-1, the staff requested that the applicant provide the calculated thermal stress range for these affected portions of the NSSS sampling system.

The applicant's December 12, 2002, response indicated that the small bore piping at FCS was designed and supported based on nomographs developed in accordance with the USAS B31.1 code.   Because the applicant used nomographs, there are currently no specific stress calculations. The applicant committed that, as part of the FMP, the sampling piping will be analyzed and a stress calculation performed to determine the thermal stress range for the line when the sampling line exceeds 7000 cycles. The applicant should confirm that the results, when completed, will meet USAS B31.1. This was identified as Confirmatory Item 4.3.2-2.

By letter dated July 7, 2003, the applicant formalized this commitment and confirmed that the stress calculation results for the small bore sampling system piping, when completed, will meet USAS B31.1 requirements. The staff finds this acceptable. Confirmatory Item 4.3.2-2 is closed.

The applicant's USAR Supplement for metal fatigue is provided in Sections A.2.10 and A.3.2 of the LRA, which includes a description of the FMP. By letter dated April 4, 2003, the applicant updated Section A.2.10 of the USAR Supplement to provide a more detailed discussion of the proposed program to address environmental fatigue effects. The staff finds the applicant's proposed USAR Supplement provides an acceptable description of the FCS fatigue TLAA evaluation and the FCS program to manage thermal fatigue during the period of extended operation to satisfy 10 CFR 54.21(d).

4.3.3 Conclusions

The staff has reviewed the applicant's metal fatigue TLAA and concludes that the applicant's actions and commitments will ensure that the subject components will be adequately managed during the period of extended operation to satisfy 54.21(c)(1).

The staff has also reviewed the revised USAR Supplement for the TLAA and finds that it is an adequate description of the metal fatigue TLAA to satisfy 10 CFR 54.21(d).

4.4 Environmental Qualification

4.4.1 Environmental Qualification Program TLAA

The 10 CFR 50.49 environmental qualification (EQ) program has been identified as a TLAA for the purpose of license renewal. EQ components include all long-lived, passive and active electrical and I&C components and commodities that are located in a harsh environment and are important to safety, including safety-related and Q-list equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment.

The staff has reviewed Section 4.4, "Environmental Qualification," of the LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1) for evaluating the EQ TLAA. 10 CFR 54.21(c)(1) requires that a list of EQ TLAAs be provided. It also requires demonstration that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effect of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also reviewed Section 4.4.3.4, "EQ Generic Safety Issue (GSI-168) for Electrical Components," of the LRA.

4.4.1.1 Summary of Technical Information in the Application

The FCS EQ program complies with all applicable regulations and manages equipment thermal, radiation, and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. Environmentally qualified equipment must be refurbished, replaced, or have its qualification extended prior to reaching the aging limits established in the aging evaluation. Aging evaluations for environmentally-qualified equipment that specify a qualified life of at least 40 years are considered TLAAs for license renewal.

The FCS Electrical Equipment Qualification (EEQ) program has been established to implement the requirements of the EQ Rule, 10 CFR 50.49. The program provides for necessary procedural controls to ensure that appropriate and timely changes are implemented. The qualified life of an equipment type is determined by the ambient environmental conditions to which it is exposed for the predicted period, internal heat rise, and cyclic stresses. Also, the qualified life of equipment can be affected by changes in plant design and operating conditions; therefore, the qualified life of equipment is frequently revisited to determine if any changes have occurred that would potentially affect the life of the equipment. The applicant routinely performs recalculations of qualified life as well as updates to equipment performance characteristics under the current EQ program. The applicant's EQ program addresses the effects of aging to ensure that the required electrical equipment function is maintained and qualified throughout its installed life. The EEQ program at FCS accomplishes the following to meet the requirements of the EQ Rule:

- reviews original qualified life bases
- establishes margin/uncertainty limits for qualified life
- reviews available aged specimen test data for impact on, and validation of, margin/uncertainty
- reviews any data for impact on, and validation of, margin/uncertainty
- adjusts qualified life based on consideration of analytical and test data and refurbishment without violating the qualification margin/uncertainty limits
- establishes new replacement dates for qualified equipment based on emergent issues, new data, industry experience, etc., as appropriate in accordance with plant and 10 CFR 50.49 program procedures

The applicant states in Section 4.4.3 of the LRA that all significant effects from normal service conditions are considered in accordance with 10 CFR 50.49 requirements. The normal service conditions include expected aging effects from normal temperature exposure, any radiation effects during normal plant operation, and cyclic aging. The applicant states that during the period of extended operation, a reevaluation of the aging effects will be performed to determine whether the equipment can continue to support the intended pre-accident service while continuing to maintain the capability to perform its post-accident intended function. The applicant states that existing analyses for thermal aging of all equipment within the FCS EQ program will be reviewed to determine if the existing calculations remain valid for the period of extended operation, or if additional analysis will be required to demonstrate qualification through the period of extended operation. Also, the total integrated dose for the 60 years will be established by making the assumption that it is equal to 1.5 times the normal operating dose for 40 years. The total integrated dose for the period of extended operation (60 years) will then be compared to the qualification level to ensure that the required total integrated dose is enveloped for the equipment. If the total integrated dose is higher than the qualification value of the equipment, then the equipment qualified life will be reassessed prior to the end of 40 years of qualified life.

The applicant has chosen option iii of the 10 CFR 54.21(c)(1) in its TLAA evaluation to demonstrate that the aging effects of the EQ equipment identified in this TLAA will be managed during the period of extended operation by the EQ program activities. The applicant states in Section 4.4.4, "Conclusion," of the LRA that aging effects of the EQ equipment identified in this TLAA will be managed during the period of extended operation consistent with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," dated July 2001, program X.E1, "Environmental Qualification (EQ) of Electrical Components."

4.4.1.2 Staff Evaluation

The staff reviewed Sections 4.1.1 and 4.4 of the LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1).

For the electrical equipment identified in LRA Table 4.1-1, the applicant uses 10 CFR 54.21(c)(1)(iii) in its TLAA evaluation to demonstrate that the aging effects of the EQ equipment identified in this TLAA will be adequately managed during the period of extended operation.

The staff reviewed the EQ program information in the LRA to determine whether it will assure that the electrical and I&C components covered under this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. The staff's evaluation of the component qualification focused on how the EEQ program manages the aging effects to meet the requirements delineated in 10 CFR 50.49.

The applicant stated that the EEQ program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49 qualification methods. Also, the applicant stated that during the period of extended operation, a re-evaluation of the aging effects will be performed to determine whether the equipment can continue to support the intended pre-accident service while continuing to maintain the capability to perform its post-accident intended function.

The applicant also stated that the EEQ program is consistent with GALL program XE1, "Environmental Qualification (EQ) of Electric Components." The continued application of 10 CFR 50.49 to EQ components that are qualified for the current qualified life for license renewal is acceptable to the staff because the EQ program has provided satisfactory management of electrical components within the program. The staff concludes that the EEQ program is capable of programmatically managing the qualified life of the components falling within the scope of the program for license renewal. The continued implementation of the FCS EEQ program provides assurance that the aging effects will be managed and that components falling within the scope of the EEQ program will continue to perform their intended functions for the period of extended operation. Thus, because the applicant will manage electrical components within the EQ program in accordance 10 CFR 50.49 for the period of extended operation, the staff finds the applicant's approach meets the requirements of 10 CFR 54.21(c)(1)(iii) and is acceptable.

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.4.1.3 Conclusions

On the basis of the review described above, the staff has determined that the applicant has evaluated the TLAA for EQ of electrical equipment consistent with 10 CFR 54.21(c)(1)(iii). The commitment made in the LRA that aging effects of the EQ equipment identified in the TLAAs will be managed during the period of extended operation consistent with GALL program X.E1, is in agreement with the GALL Report conclusion that plant EQ programs, which implement the requirements of 10 CFR 50.49, are viewed as acceptable aging management programs for license renewal under 10 CFR 54.21(c)(1)(iii).

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.4.2 GSI-168, "Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables"

During the staff's review of license renewal issues, the EQ process was found to be a significant issue. Of particular concern was whether the EQ requirements for older plants, whose licensing bases differ from newer plants, are adequate for license renewal. Further, a question was raised as to whether the EQ requirements for older plants should be reassessed for the current licensing term. Upon subsequent review, additional concerns were raised related to the EQ process, and it was concluded that differences in EQ requirements constituted a potential generic issue that should be evaluated for backfit, independent of license renewal. This came to be identified as Generic Safety Issue (GSI)-168. Key items to be addressed in GSI-168 are:

- the adequacy of older EQ requirements for license renewal, as well as the current licensing term
- the adequacy of accelerated aging techniques to simulate long-term natural service aging
- the possibility that unique failure mechanisms exist for bonded jacket and multi-conductor cable configurations that are not adequately addressed in EQ
- the feasibility of using condition monitoring techniques to monitor current cable condition in situ as a means of offsetting uncertainties in the process used to predict long-term service aging

The staff has provided guidance to the industry (letter dated June 2, 1998 from the NRC (Grimes) to the Nuclear Energy Institute (NEI) (Walters)), which states:

- GSI-168 issues have not been identified to a point that a license renewal applicant can be reasonably expected to address these issues specifically at this time
- an acceptable approach is to provide a technical rationale demonstrating that the CLB for EQ will be maintained in the period of extended operation.

For the purpose of license renewal, as discussed in the Statements of Consideration (SOC) (60 FR22484, May 8, 1995), there are three options for addressing issues associated with a GSI:

- If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution in the LRA.
- An applicant can submit a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.
- An applicant can develop a plant-specific aging management program that incorporates the resolution of the aging issue.

4.4.2.1 Summary of Technical Information in the Application

The applicant states that since environmental qualification is a TLAA for license renewal, outstanding GSIs that could affect the validity of any credited analyses must be dispositioned as part of the application process. GSI-168 remains unresolved, and for the purposes of license renewal, there are three options for resolving issues associated with a GSI.

1. If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution in the LRA.

2. An applicant can submit a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.

3. An applicant can develop a plant-specific aging management program that incorporates the resolution of the aging issue.

The applicant states that it has chosen to pursue the second option, so until GSI-168 is resolved, aging management of qualified cables will be addressed through plant-specific programs. At that time, one or more reasonable options should be available to adequately manage the effects of aging.

4.4.2.2 Staff Evaluation

As stated above, there are three options for addressing issues associated with a GSI:

1. If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution in the LRA.

2. An applicant can submit a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging.

3. An applicant can develop a plant-specific aging management program that incorporates the resolution of the aging issue.

The applicant has chosen to pursue Option 2 with regard to GSI-168. This option requires the applicant to provide a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging. The research and technical assessment of GSI-168 is limited to low-voltage instrumentation and control (I&C) cables in harsh environments. GSI-168 does not encompass any other electrical equipment or components. As such, the applicant's technical rationale provided in the LRA on GSI-168 addresses cables that are captured in GSI-168. No additional rationale is needed from the applicant.

The existing EQ program at FCS complies with all applicable regulations and manages equipment thermal, radiation and cyclic aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. The plant will continue to use these methods to manage the qualification of I&C cables until such time as GSI-168 is resolved. The applicant has committed to incorporate the resolution of GSI-168 into the extended period of operation. The staff finds the applicant's response acceptable.

4.4.3 Conclusions

On the basis of its review, the staff concludes that the applicant has addressed the issues associated with GSI-168. The applicant will continue to manage the effects of aging through plant-specific programs in accordance with the CLB until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging. The staff finds that the applicant has 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 on GSI-168 requires no actions on the part of addressees. Therefore, the staff considers the GSI-168 issue to be closed.

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.5 Concrete Containment Tendon Prestress

The prestressing tendons in prestressed concrete containments lose their prestressing forces with time due to creep and shrinkage of concrete and relaxation of the prestressing steel. During the design phase, engineers estimate these losses to arrive at the prestressing forces at the designated operating life, normally 40 years. The operating experience with the trend of prestressing forces indicates that the prestressing tendons lose their prestressing forces at a rate higher than predicted due to sustained high temperature. Thus, it is necessary to perform TLAAs for the period of extended operation. The adequacy of the prestressing forces in prestressed concrete containments is reviewed for the period of extended operation.

4.5.1 Summary of Technical Information in the Application

The applicant describes its TLAA for prestressing force for the FCS containment as follows:

> Pre-stressing tendon integrity is monitored and confirmed by the containment ISI program (B.1.3). The program provides for tendon inspection 1, 2 and 4 years after initial pre-tensioning, and every five years thereafter for the remaining life of the plant. The pre-stressing tendon surveillances are performed in accordance with NRC Regulatory Guide 1.35 revision 3, as implemented in Amendment 139 to the FCS operating license.
>
> Curves showing anticipated variation of tendon force with time, together with the lower limit curves to be applied to surveillance readings, are shown in the FCS USAR. The curves are given in terms of net force in the tendon and as a percentage of the initial tendon load. The calculated pre-stress at end of plant life exceeds by a reasonable margin the intensity required to meet the design criteria. This margin is the basis of the limits set for deviation with time of the tendon forces as measured by the periodic lift-off readings. If at any time surveillance testing indicates a decrease in the tendon force below the given limit line, corrective action will be taken in accordance with the Technical Specifications. The USAR curves will be extended to 60 years of plant life to cover the period of extended operation. This will also show that the pre-stressing force is acceptable for continued service at the end of the period of extended operation considering the assumed time dependent nature of pre-stress losses. The tendon surveillance program will be continued into the period of extended operation using the updated curves. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

In Section A.3.4 of the USAR Supplement, the applicant summarizes the TLAA and concludes that the calculated prestress at the end of plant life exceeds by a reasonable margin the intensity required to meet the design criteria.

4.5.2 Staff Evaluation

The staff's review of the TLAA indicated that the applicant was missing an important acceptance criterion in the description of the TLAA. In RAI 4.5-1, the staff requested information regarding this acceptance criterion as follows:

> For acceptance criterion for tendon prestressing force, the LRA states: "If at any time surveillance testing indicates a decrease in the tendon force below the given limit line, corrective action will be taken in accordance with the Technical Specifications." This is one of the criterion (sic) in IWL-3221. Additionally, 10 CFR 50.55a(b)(2)(viii)(B) requires: "When evaluation of consecutive surveillance's of prestressing forces for the same tendon or tendons in a group indicates a trend of prestressing loss such that the tendon forces will be less than the minimum design prestress requirements before the next inspection interval, an evaluation must be performed and reported in the Engineering Evaluation Report as prescribed in IWL-3300." Based on these requirements, the staff requests the applicant to clarify whether the acceptance criterion in the LRA complies with the requirements of IWL-3221 and 10 CFR 50.55a(b)(2)(viii)(B).

In response, the applicant stated that the acceptance criterion in the LRA does comply with IWL-3221 and 10 CFR 50.55a(b)(2)(viii)(B). A regression analysis of forces measured on specific tendons was conducted and included in the tendon testing report. The analysis showed satisfactory results for the next surveillance. Furthermore, the applicant provided detailed information regarding the process used to comply with the regulation in Appendix C attached to its letter dated March 14, 2003. The staff reviewed the process and the curves indicating future trends with respect to the minimum required prestress for each group of tendons. The staff found that the process satisfied 10 CFR 50.55a(b)(2)(viii)(B), and is therefore acceptable.

The applicant did not provide adequate quantitative evaluation based on the prior tendon inspections. In RAI 4.5-2, the staff requested the following information:

> Title 10 CFR 50.55a(b)(2)(viii)(B) requires the development of a trend line of measured prestressing forces so that the licensee can decide whether the prestressing tendon forces during the next inspection interval will remain above the "Lower Limit - Dome," and "Lower-Limit-Wall," as plotted in USAR Figure 5.10-3. The applicant addresses this TLAA using Section X.S1 of the GALL Report, as part of its operating experience. In order to confirm that the prestressing tendon forces will remain above the lower limits for the dome and wall during the period of extended operation, the staff requests that the applicant provide information related to the trend lines for wall and dome tendons compared to the established lower limits. Guidance for statistical considerations in developing the trend lines is given in Attachment 3 of IN 99-10, Revision 1, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments."

In response, the applicant stated that because it is using 10 CFR 54.21(c)(1)(iii), i.e managing the TLAA, it need not provide such information. However, the staff needs the quantitative data of trend lines, as part of the operating experience, to make a conclusion regarding this TLAA for the period of extended operation. In Appendix C to its March 14, 2003, letter, the applicant provided the quantitative trend lines based on the containment tendon inspections performed thus far at FCS. It should be noted that the future prestressing force measurements could change the predictions. However, because the applicant is going to continue monitoring the tendon forces as required by ASME Section XI, Subsection IWL, the staff finds the process, and the quantitative data provided by the applicant in its March 14, 2003, letter, acceptable.

The staff reviewed the USAR Supplement for this TLAA and concluded that it provides an adequate summary description of the TLAA.

4.5.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that based on the operating experience related to the tendon prestressing forces, the identified aging management program will adequately mange the containment tendon prestressing forces during the period of extended operation. The staff also reviewed the USAR Supplement and concluded that it contains an appropriate summary description of the concrete containment tendon prestress TLAA evaluation for the period of extended operation, as reflected in the current licensing basis, to satisfy 10 CFR 54.21(d). Therefore, the staff finds 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 reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.6 Containment Liner Plate and Penetration Sleeve Fatigue

The interior surface of a concrete containment structure is lined with thin metallic plates to provide a leak-tight barrier against the uncontrolled release of radioactivity to the environment, as required by 10 CFR Part 50. The thickness of the liner plates is generally between 1/4 inch (6.2 mm) and 3/8 inch (9.5 mm). The liner plates are attached to the concrete containment wall by stud anchors or structural rolled shapes, or both. The design process assumes that the liner plates do not carry loads. However, normal loads, such as from concrete shrinkage, creep, and thermal changes, imposed on the concrete containment structure are transferred to the liner plates through the anchorage system. Internal pressure and temperature loads are directly applied to the liner plates. Thus, under design basis conditions, the liner plates could experience significant strains.

Fatigue of the liner plates may be considered in the design based on an assumed number of loading cycles for the current operating term. The cyclic loads include reactor building interior temperature variation during the heatup and cooldown of the reactor coolant system, a LOCA, annual outdoor temperature variations, thermal loads due to high-energy containment penetration piping lines (such as steam and feedwater lines), seismic loads, and pressurization due to periodic Type A integrated leak rate tests.

The containment liner plates, 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 will be adequately managed for the period of extended operation.

The adequacy of the fatigue analyses of the containment liner plates (including welded joints), penetration sleeves, dissimilar metal welds, and penetration bellows is reviewed for the period of extended operation. The fatigue analyses for the pressure boundary of process piping are reviewed in Section 4.3 of this SER, following the guidance in Section 4.3 of the SRP-LR.

4.6.1 Summary of Technical Information in the Application

The applicant discussed the design of the FCS containment liner and penetration sleeves in Section 4.6 of the LRA. The applicant indicated that the containment linear and penetration sleeves were designed using the ASME Code, Section III, “Nuclear Vessels,” as a guide in the determination of acceptable strains. The applicant also indicated that the liner reinforcement at all penetrations meets the requirements of the ASME Code, Section III, “Class B Vessels,” and that the penetration design and materials conform to the requirements of the ASME Code, Section III, “Nuclear Vessels.” The applicant indicated that fatigue considerations were of prime importance in the design of the liner and attachments and that the following fatigue loadings were assumed for the design:

- thermal cycling caused by one loss-of-coolant accident
- thermal cycling caused by variation of annual outdoor temperatures (40 cycles)
- thermal cycling caused by variation of internal temperature between shutdown and operating conditions (500 cycles)

The applicant indicated that the design CUF for the liner plate and attachments was 0.05. The applicant indicated that this value was computed based on an assumed inward curvature of the liner plate between stiffeners of 1/16 inch. The applicant indicated that actual measurements of the containment liner found values of 1/4 to 3/4 inch. The applicant indicated that this condition was evaluated and found acceptable for the current term.

4.6.2 Staff Evaluation

The design of the FCS liner and anchorage system is described in Sections 5.5 and 5.6 of the USAR. The USAR indicates that the 1/4-inch thick liner is anchored at 14-1/2 inch centers by continuous structural tees. Section 5.6 of the USAR indicates that an analysis of the liner steel was performed for 500 cycles of operating conditions, and the calculated CUF of 0.05 was compared with an allowable value of 1.0 permitted by ASME, Section III, N415.2(e)(6). The USAR also indicates that an inward curvature of 1/16 inch of a single panel was assumed in the analysis of the linear plate for the most critical case. As discussed previously, the applicant indicated that actual measurements found larger displacements and that an analysis of the as-found displacements for the 60-year period would be completed prior to the period of extended operation. In RAI 4.6-1, the staff requested that the applicant describe the analysis that was performed to show the containment liner plate/penetration sleeve meets acceptance criteria for the current term and to provide the calculated usage factor obtained from this analysis.

In its December 12, 2002, response, the applicant indicated that the recent analysis of the as-found buckling of the liner plate was performed using non-linear, 3D finite element analysis with loads applied in a fashion similar to the original analysis. The applicant indicated that an undeformed panel was evaluated using the new model to benchmark the new model against results from a comparable model from the original analysis. The applicant indicated that the new analysis resulted in a CUF of 0.141 for the 500 cycles of internal temperature variation due to heatup and cooldown. The applicant further indicated that 500 cycles is greater than the number of cycles expected for 60 years of plant operation. This is consistent with the applicant's response to RAI 4.3.1-1, which indicates that there have been 66 cycles of heatup and cooldown of the RCS in approximately 30 years of plant operation. The staff also notes that the number of heatup and cooldown cycles is being tracked by the FCS FMP. By letter dated February 20, 2003, the staff issued POI-13(h), requesting that the applicant verify that the thermal cycling due to outdoor temperature variation does not result in significant fatigue usage. The staff also requested that the applicant clarify whether the current evaluation bounds the fatigue usage in the penetration area.

By letter dated March 14, 2003, the applicant responded to POI-13(h). The applicant indicated that the analysis of the as-found buckling of the liner plate included cyclic conditions for outdoor air annual temperature changes and LOCA transients, and that the contribution to the fatigue usage factor from the outdoor air temperature variations was insignificant. The applicant further indicated that the containment liner plate buckling was remote from the penetration area and, therefore, the buckling had no effect on the stresses and fatigue usage at the penetration. The staff finds the applicant has adequately addressed the cyclic design loads in the fatigue evaluation of the liner plate buckling. Therefore, POI-13(h) is resolved.

The applicant provided a summary description of the containment liner plate and penetration sleeve fatigue TLAA in Section A.3.5 of the USAR Supplement. The applicant indicated that an evaluation of the liner plate as-found buckling for a 60-year life will be completed prior to the period of extended operation. By letter dated February 20, 2003, the staff issued POI-13(i) requesting the applicant to update the USAR Supplement to indicate that the evaluation is complete and to provide the evaluation results. By letter dated March 14, 2003, the applicant provided the requested USAR revision. POI-13(i) is resolved.

4.6.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1) that, for the containment liner plate and penetrations fatigue TLAA, (i) the analyses remain valid for the period of extended operation. The staff also concludes that the USAR Supplement contains an appropriate summary description of the containment liner plate and penetration sleeve fatigue TLAA, as required by 10 CFR 54.21(d).

4.7 Other TLAAs

There are certain plant-specific safety analyses that may have been based on an explicitly assumed 40-year plant life and may, therefore, be TLAAs. Pursuant to 10 CFR 54.21(c)(1), a license renewal applicant is required to evaluate TLAAs, as defined in 10 CFR 54.3. License renewal reviews focus on the period of extended operation.

The applicant has identified three additional TLAAs for license renewal:

- reactor coolant pump flywheel fatigue
- leak-before-break analysis for resolution of USI A-2
- high-energy line break (HELB)

The staff's evaluation of these TLAAs is provided below.

4.7.1 Reactor Coolant Pump (RCP) Flywheel Fatigue

4.7.1.1 Summary of Technical Information in the Application

General Electric Designed (GE-design) RCP Flywheels

The applicant stated that General Electric (GE) manufactured the original RCP motors and that each GE pump motor is provided with a flywheel that reduces the rate of flow decay upon loss of pump power. The applicant stated that conservative design bases and stringent quality control measures have been taken to preclude failure of the flywheel and that the following design features for the GE-designed RCP flywheels ensure that the requirements for structural soundness were met:

- division of the mass into three separate discs
- a keyway fillet radius not less than 1/8 inch to minimize stress concentrations
- fabrication of the discs using forged carbon steel plate having different tensile strengths

The applicant stated that the resistance of the GE-designed RCP flywheels to rupture was examined at 120 percent overspeed, and that the critical crack length for the disc most susceptible to crack propagation was found to be 3 inches, as based on fracture mechanics data furnished by GE and the assumption that the crack extended radially outward from the flywheel's keyway and penetrated completely through the thickness of the disc. Using the crack growth prediction techniques provided by GE, the applicant concluded that more than 185,000 complete cycles from 0 to 120 percent overspeed would be required to cause a 0.5 inch long crack extending radially from the keyway to grow to critical size. This number of cycles will not be exceeded if the licensing period is extended to 60 years. To do so would require in excess of eight pump starts per day, which far exceeds actual and projected pump use. Since the cycle limit will not be exceeded, the applicant concluded that fatigue crack growth analysis for the GE-design RCP flywheels remains valid for the period of extended operation.

ABB Design RCP Flywheel (ABB-design, flywheel for the RCP No. RC-3B)

The applicant stated that during the 1996 refueling outage, the RCP RC-3B motor was replaced with a motor manufactured by ABB Industries. The applicant stated that the flywheel was conservatively designed and made with closely controlled quality material such that the probability of a flywheel failure is sufficiently small and that, therefore, a steel shroud was not included in the flywheel design.

The applicant stated that the ABB-designed RCP flywheel was made from forged ASTM A508 4/5 steel and shrink-fitted to the shaft collar and that the flywheel was designed, manufactured, and tested per the guidance of RG 1.14, Revision. 1, "Reactor Coolant Pump Flywheel Integrity," dated August, 1975. The applicant stated that a crack growth analysis was performed by ABB, which demonstrated that critical flaw growth would not occur with fewer than 10,000 complete cycles (RCP startups) from 0 to 120 percent overspeed. The applicant stated that this number of cycles will not be exceeded even if the licensing period is extended to 60 years, because to do so, the applicant would have to start the pumps approximately once every two days, which far exceeds actual and projected pump use at the plant. The applicant stated that, since the cycle limit for the flywheel will not be exceeded, the analysis for the flywheel remains valid for the period of extended operation.

4.7.1.2 Staff Evaluation

10 CFR 54.21(c)(1) requires applicants for license renewal to demonstrate that TLAAs for license renewal have been projected through the end of the period of extended operation for their facilities, remain valid for the period of extended operation, or demonstrate that the effects of aging that are applicable to the components evaluated by the TLAAs will be managed during the period of extended operation. PWR RCP flywheels are designed with rotors and discs that revolve at high speeds that can make the components susceptible to crack initiation and growth by fatigue, which is a time-dependent aging mechanism. The regions of the flywheels that are most susceptible to low-cycle fatigue are located at the corners of the locking mechanisms in the flywheel discs. These corners act as stress risers which make the corners more highly susceptible to the initiation and growth of cracking induced by fatigue.

The scope of Section X of the GALL Report, Volume 2, does not currently include recommended guidelines for performing TLAAs of fatigue crack growth analyses for PWR RCP flywheels. However, RG 1.14, Revision. 1, “Reactor Coolant Pump Flywheel Integrity,” dated August, 1975, provides acceptable guidelines for ensuring the structural integrity of RCP flywheels in PWR-designed nuclear plants1 against critical-fracture or fatigue-induced failures. The applicant did not initially provide its fatigue crack growth analyses for the GE-designed and ABB-designed RCP flywheels in the FCS LRA. The staff issued RAI 4.7.1-1 to request that the applicant provide its fatigue crack growth analyses for the GE-designed and ABB-designed RCP flywheels for staff review to demonstrate that the fatigue crack growth analyses for the GE-designed and ABB-designed RCP flywheels remain valid for the period of extended operation for FCS.

The applicant provided its response to RAI 4.7.1-1 by letter dated December 19, 2002. For the GE-design RCP flywheel, the applicant clarified that the details of the analysis are adequately summarized in Section 4.3.5 of the FCS USAR. For the ABB-design RCP flywheel, the applicant provided proprietary calculation FC06608 for staff review.

The staff’s evaluation of the TLAAs for fatigue-induced crack growth in the GE-designed RCP flywheels and the ABB-designed RCP flywheel is discussed below.

<u>General Electric Design (GE-design) RCP Flywheels</u>

The applicant’s fatigue crack growth analysis for the GE-design RCP flywheel is summarized in USAR Section 4.3.5. Staff review of USAR Section 4.3.5 indicates that the USAR section provides sufficient technical information to address the staff’s request in RAI 4.7.1-1, as it relates to the fatigue crack growth analysis for the GE-designed RCP flywheel. The fatigue crack growth analysis summarized in the USAR postulates the occurrence of a surface flaw that extends 0.5 inches from the corner of the flywheel keyway locking component for the limiting flywheel disc. The fatigue crack growth analysis is based on the number of startups of a GE-designed RCP from 0 to 120 percent operational overspeed. The analysis is therefore dependent on the number of accumulated RCP trips over the licensed period for the plant. The postulated flaw size represents the crack size that could exist in the flywheel disc and be detected in the flywheel discs during inservice inspections (ISI) of the disc. The USAR section indicates that based on a 17.5 ksi loading stress, which is attributed to rotation of the disc at 120 percent of the flywheel's normal operational design speed, it would take 185,000 trips of the RCP to extend the 0.5 inch flaw beyond the critical flaw size for the disc (i.e., 3.0 inches). This would require the applicant to trip the RCP associated with the GE-designed flywheel at a frequency exceeding eight pump trips per day. This frequency conservatively exceeds the number of anticipated GE-design RCP trips assumed in the design basis through the end of the period of extended operation for FCS.

<u>ABB Design RCP Flywheel (ABB-design, flywheel for the RCP No. RC-3B)</u>

In its response to RAI 4.7.1-1, by letter dated December 19, 2002, the applicant provided the fracture mechanics and fatigue crack growth analysis for the ABB-designed RCP flywheel. The ABB analysis postulates the existence of a fatigue-induced crack (the length is designated as proprietary information in the calculation) in the flywheel that is more than 30 percent of the acceptable crack length in the flywheel. The crack growth analysis is based on the stress intensity associated with the operating condition that creates the limiting applied stress (load) on the crack. The following loading (stress) conditions were considered:

- loading under normal operations with revolution of the flywheel at synchronous speed
- loading under normal operations with revolution of the flywheel at test speed (i.e., rotation at greater than 120 percent of synchronous speed)
- loading under operational basis earthquake loads with revolution of the flywheel at synchronous speed
- loading under design basis earthquake loads (faulted conditions) with revolution of the flywheel at synchronous speed

For FCS, this limiting applied stress is associated with revolution of the flywheel at synchronous speed under faulted loading conditions, which bound the loading conditions for revolution of the flywheel at test speed under normal operations. The applicant therefore based the loadings for the fatigue crack growth analysis on the loadings for revolution of the flywheel at synchronous speed under faulted loading conditions. This is a conservatism in the analysis. The staff's review of the applicant's proprietary analysis confirmed that it would take more than 10,000 trips of the RCP to exceed the maximum allowable crack size for the ABB-designed RCP flywheel. To achieve this number of pump trips, the applicant would have to trip the RCP associated with the ABB-designed flywheel at a frequency exceeding once every two days. This frequency exceeds the number of anticipated ABB-designed RCP trips assumed in the design basis through the end of the period of extended operation for FCS.

4.7.1.3 USAR Supplement

The applicant provides its USAR Supplement for the TLAAs on the GE-designed and ABB-designed RCP flywheels in Sections A.3.6.1.1 and A.3.6.1.2 of the LRA. The USAR Supplement summarized the fatigue crack growth analysis results and crack growth conclusions for the flywheels. The USAR Supplement also provides enough information to demonstrate that the structural integrity of the GE-designed and ABB-designed RCP flywheels will be acceptable through the expiration of the period of extended operation for FCS. Based on the staff's review of Sections A.3.6.1.1 and A.3.6.1.2 of the LRA, the staff concludes that the

USAR Supplement for the TLAAs on the GE-designed and ABB-designed RCP flywheels are acceptable and satisfy 10 CFR 54.21(d).

4.7.1.4 Conclusions

On the basis of its review, including the applicant's response to the staff's RAI, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1) that, for the fatigue crack growth TLAAs for GE-designed and ABB-designed RCP flywheels, and that the analyses remain valid for the period of extended operation for FCS. The staff also concludes that the USAR Supplement Sections A.3.6.1.1 and A.3.6.1.2 contain appropriate summary descriptions of the applicant's RCP flywheel fatigue TLAA evaluations for the period of extended operation, as required by 10 CFR 54.21(d). Therefore the staff finds that the safety margins established and maintained during the current operating term will be maintained for the period of extended operation.

4.7.2 Leak-Before-Break (LBB) Analysis for Resolution of USI A-2

4.7.2.1 Summary of Technical Information in the Application

In Section 4.7.2 of the applicant's LRA, the applicant states:

> There are two TLAA aspects to LBB, crack growth and thermal aging. While transient cycle fatigue crack growth is a TLAA for FCS and also a design consideration, thermal aging was not evaluated for FCS by either the original design code or the LBB analysis. Consequently, OPPD will perform a plant-specific LBB analysis prior to the period of extended operation. This analysis will consider a 60-year life and thermal aging effects of the cast austenitic stainless steel (CASS) RCS and will be completed before the period of extended operation. Therefore, the analysis will be projected to the end of the period of extended operation.

The staff requested an additional applicant commitment in RAI 4.7.2-1 regarding the evaluation of the impact of the potential for Inconel 82/182 weld PWSCC on the applicant's LBB evaluation. In response to RAI 4.7.2-1, the applicant stated

> "For the period of extended operation of FCS, OPPD will implement actions or perform analyses, as required by the NRC, to confirm continued applicability of existing FCS LBB evaluations. These actions or analyses will be consistent with those required to address the impact of PWSCC on existing LBB evaluations under Part 50 considerations."

4.7.2.2 Staff Evaluation

The staff has evaluated the information provided by the applicant in its LRA and in its response to RAI 4.7.2-1. The staff has concluded that the applicant appropriately identified those TLAAs (fatigue crack growth, aging of cast austenitic stainless steel (CASS) RCS piping and components, and primary water stress-corrosion cracking (PWSCC) of Inconel 82/182 RCS welds), which may impact the extension of the applicant's existing leak before break (LBB) analysis through the period of extended operation. The applicant has committed to perform a plant-specific LBB analysis prior to entering the period of extended operation which will address these TLAAs and project the analysis to the end of the period of extended operation. However, the applicant's commitment did not appear to meet 10 CFR 54.21(c)(1) which requires the applicant to demonstrate that (i) the analysis remains valid for the period of extended operation,

(ii) the analysis has been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff requested that the applicant provide the information needed for the staff to determine whether (i) the applicant's LBB analysis remains valid for the period of extended operation, (ii) the applicant's LBB analysis has been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) of the components within the scope of the LBB analysis will be adequately managed for the period of extended operation. This was identified as Open Item 4.7.2.2-1.

NEI 95-10, Revision 3, provides guidance to applicants who apply for renewal of their operating licenses. In Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," the staff has endorsed this NEI guideline. Section 5.1.4 of NEI 95-10 allows for deferral of TLAA evaluations. The guidance states that, if an applicant decides to defer the completion of an evaluation, it should submit additional information to the staff to support a conclusion that the effects of aging addressed in the TLAA will be adequately managed. This information includes (1) details of the methodology that will be used for the TLAA evaluation, (2) the acceptance criteria that will be used to judge the adequacy of the structure or component, consistent with the CLB, when the TLAA evaluation or analysis is performed, (3) the corrective actions that will be performed to provide reasonable assurance that the structure or component will perform its intended function or will not be outside of its design basis established by the CLB, and (4) information to identify when the completed TLAA evaluation will be submitted to ensure that the evaluation will be performed before the structure or component will be unable to perform its intended function.

By letter dated July 7, 2003, the applicant stated that it will defer completion of the plant-specific LBB evaluation in accordance with Section 5.1.4 of NEI 95-10. The applicant submitted the information below, as provided in NEI 95-10.

- The applicant committed to complete a plant-specific LBB evaluation of the RCS piping using the latest LBB criteria. The LBB analysis will incorporate the effects of thermal aging, plant-specific materials, operating temperatures/pressures, loads at welds in the primary loops, and weld fabrication. The plant-specific methodology will also use the existing plant's RCS leak detection capability and the piping stress analysis loads for the FCS RCS configuration. The analysis will be applicable for the period of extended operation, and will use a methodology from the Westinghouse Electric Company for thermal aging considerations. Westinghouse has performed over 30 plant-specific LBB analyses approved by the NRC, and addressed thermal aging effects of the cast materials as applicable. For the primary loop piping, the latest LBB SER which includes the Westinghouse analysis methodology was for D.C. Cook Units 1 and 2. This SER was issued in December 1999 (docket numbers 50-315 and 50-316).

The staff reviewed this information and finds that it adequately describes the methodology that will be used for the applicant's LBB analysis.

- Acceptance criteria used to determine the adequacy of the structure or component when the LBB analysis is performed will be in accordance with draft Standard Review Plan (SRP) 3.6.3, "Leak-Before-Break Evaluations Procedures," published for comment in Volume 52, Number 167 of the *Federal Register*, dated, Friday, August 28, 1987, and NUREG-1061, Volume 3.

The staff reviewed this information and finds that the applicant has identified the acceptance criteria that will be used to judge the adequacy of the structures or components when the LBB analysis is performed.

- The plant-specific LBB analysis will include evaluation of corrective actions that can be performed to provide reasonable assurance that the component in question will perform its intended function when called upon, or will not be outside of its design basis established by the plant's CLB. One such corrective action is to maintain the CLB RCS leak rate program as defined in FCS Technical Specification (TS) 2.1.4 during the period of extended operation. The leak detection capability of the systems noted in TS 2.1.4 meet the intent of Regulatory Guide 1.45 and will be capable of performing their designed function during the period of extended operation.

The staff reviewed this information and finds that the applicant has identified the corrective actions it will perform to ensure that the structures and components will continue to perform their intended functions.

- The applicant committed to submit a License Amendment Request containing the plant-specific LBB evaluation described above to the NRC no later than December 2006, which is well before the period of extended operation. This submittal schedule supports the applicant's planning decisions for possible changes to RCS operation or configuration.

The staff reviewed this information and finds that the applicant has identified the submittal date for the LBB analysis. Further, the staff concludes that this submittal date should provide sufficient time to address aging issues before loss of intended function of the applicable SCs.

On the basis of the applicant's response to Open Item 4.7.2.2-1, the staff concludes that the applicant has followed the guidance to support the deferral of the submittal of its LBB analysis. The characteristics of the LBB analysis, as proposed by the applicant, is sufficient to allow the staff to conclude that the effects of aging addressed in the TLAA will be adequately managed, as required by 10 CFR 54.21(c)(1)(iii). Open Item 4.7.2.2-1 is closed.

With regard to the identified fatigue crack growth and CASS thermal aging TLAAs, the staff has determined that adequate assurance exists regarding the ability of the applicant to perform acceptable analyses of these issues. Each of these issues has been adequately addressed by other license renewal applicants in support of extending existing LBB evaluations through the period of extended operation. The NRC staff has concluded that there are no known unique concerns regarding FCS which would prevent the applicant from performing acceptable TLAAs for each of these issues prior to entering the period of extended operation for FCS.

Regarding the impact of Inconel 82/182 PWSCC on LBB evaluations, the NRC staff has concluded that this is a generic current licensing basis issue outside of the scope of license renewal. The staff is continuing to review the generic implications of PWSCC on LBB approvals. The staff may consider the need for additional applicant actions or analyses, as appropriate, to ensure that the underlying basis for approval of LBB for the FCS main coolant loop remains valid. Therefore, the staff finds the applicant's commitment to "implement actions or perform analyses, as required by the NRC, to confirm continued applicability of existing FCS

LBB evaluations....consistent with those required to address the impact of PWSCC on existing LBB evaluations" (see the applicant's response to POI-7(f) in Section 3.1.2.3.4.2 of this SER) to be acceptable for addressing this TLAA within the scope of the applicant's LRA.

4.7.2.3 USAR Supplement

The applicant provides its USAR Supplement for the LBB analysis in Section A.3.6.2 of the LRA. On the basis of its review, the staff concludes that the USAR Supplement for the TLAAs on LBB is acceptable.

4.7.2.4 Conclusions

On the basis of its review, the staff concludes that the applicant will be able to provide, prior to entering the period of extended operation, an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(ii), regarding the projection of its leak-before-break analysis for resolution of USI A-2 TLAA, through the end of the period of extended operation. The applicant's commitment to submit an updated LBB analysis, which addresses the TLAAs identified above, is documented in Appendix A to this SER. The staff also concludes that the USAR Supplement contains an appropriate summary description of the LBB analysis for resolution of USI A-2 TLAA evaluation for the period of extended operation, as required by 10 CFR 54.21(d). Therefore, the staff finds 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).

4.7.3 High-Energy Line Break

4.7.3.1 Summary of Technical Information in the Application

The applicant described its evaluation of high-energy line breaks (HELBs) in Section 4.7.3 of the LRA. The applicant indicated that fatigue analyses were performed for the B31.7 Class I portions of main steam (MS) and main feedwater (MFW) outside containment to identify locations with CUF greater than 0.1, which is the criterion for postulating pipe breaks. The applicant indicated that, for the MFW piping, breaks were postulated at the end of each pipe segment. The applicant indicated that the Class I portions of the MFW outside containment are wrapped in steel "barrel slat" enclosures to prevent lateral pipe movement and the formation of longitudinal and axial jets, which could impact nearby structures and equipment. The applicant further indicated that pipe whip restraints are installed to limit pipe movement due to circumferential breaks. Consequently, the applicant concluded that any additional locations on the Class I portion of the piping will be bounded by the existing break locations. The applicant indicated that a similar design existed for the Class I MS piping with one potential exception. The applicant indicated that an evaluation had not been performed to determine whether the slat enclosures protected the piping connections to the isolation valves. The applicant indicated that the design CUFs at these locations were less than 0.001 and, therefore, would not exceed the 0.1 criterion during the period of extended operation.

4.7.3.2 Staff Evaluation

The applicant's HELB criteria are provided in Appendix M of the USAR. Appendix M indicates that portions of the MS and MFW piping between the containment and the outside isolation valve were designed in accordance with ANSI B31.7. The Class I criteria require a fatigue analysis. As indicated by the applicant, the pipe break criteria for the Class I portions of the MS and MFW piping require postulation of pipe breaks at locations where the CUF may exceed 0.1. The applicant's evaluation indicates that the existing postulated pipe breaks are bounding for all Class I sections of the MS and MFW piping, except the MS connections to the isolation valves. The applicant's evaluation also indicates that the calculated usage factor for those locations will not exceed the criterion of 0.1 for the period of extended operation. Therefore, the applicant concluded that the pipe break analyses remain valid for the period of extended operation and meet the requirements of 10 CFR 54.21(c)(1)(i). The staff finds that the applicant performed an acceptable TLAA of the FCS pipe break criteria.

The staff also reviewed the USAR Supplement for this TLAA and concludes that it provides an adequate summary description of the TLAA to satisfy 10 CFR 54.21(d).

4.7.3.3 Conclusions

On the basis of its review, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(i), that for the HELB TLAA, the analyses remain valid for the period of extended operation. The staff also concludes that the USAR Supplement contains an appropriate summary description of the HELB TLAA evaluation for the period of extended operation, as required by 10 CFR 54.21(d). Therefore the staff finds that the safety margins established and maintained during the current operating term will be maintained during the period of extended operation.

4.7.4 Alloy 600 Weld Repair in a Temperature Nozzle in the Pressurizer Lower Shell

4.7.4.1 Summary of Technical Information in the Application

The application did not initially discuss an Alloy 600 repair in the temperature nozzle in the pressurizer lower shell. As a result of discussions between the staff and the applicant, the applicant in a letter dated July 7, 2003, added a new Section 4.7.4 to the license renewal application. This section indicates that the temperature nozzle in the pressurizer lower shell was repaired by adding a weld pad to the existing weld build-up to the lower shell outer diameter (OD) and welding this pad to the existing nozzle. This moved the pressure boundary from the inner diameter to this location. The Alloy 600 J-weld and original crack were left in place at the inside surface of the pressurizer as part of the repaired configuration.

In a letter dated October 25, 2000, Westinghouse provided Omaha Public Power District (OPPD) the technical justification for the weld on the liquid space Alloy 600 instrument nozzle on the OD of the pressurizer. This letter stated that the subject repair should be made in accordance with later editions of Section III, or the 1992 Edition (or later) of Section XI.

In April 2002, Westinghouse notified OPPD that its technical justification of October 2000 only considered the effects of the repair on the requirements of ASME Section III, and did not consider the Section XI requirements related to leaving the flaw in place after the repair was completed and the vessel returned to service.

In April 2003, OPPD received the "calculation note" titled "Evaluation of Fatigue Crack Growth of Postulated Flaw at Omaha Fort Calhoun Pressurizer Lower Shell Instrumentation Nozzle," dated January 8, 2003, that evaluated the Section XI requirements related to leaving the flaw in place after the repair was completed and the vessel returned to service.

OPPD has evaluated the crack, and any potential future growth of the crack, and determined it does not impact the structural integrity of the vessel for the current licensed 40-year life. OPPD has elected to defer completion of the evaluation that demonstrates that the crack and any potential future growth of the crack does not impact the structural integrity of the vessel for the period of extended operation. On the basis of guidance in Section 5.1.4 of NEI 95-10, Revision 3, the applicant provided details to explain how the effects of aging will be addressed for this evaluation.

OPPD will submit, for staff review and approval, the fracture mechanics evaluation for the period of extended operation of the small-bore instrument nozzle J-weld region at the repaired instrument nozzle. This submittal will be made prior to entering the period of extended operation. This evaluation will include bounding the flaw size by the size of the J-weld itself, and addressing the possibility of corrosion in the presence of a flaw.

4.7.4.2 Staff Evauation

Because the application did not initially discuss an Alloy 600 repair in the temperature nozzle in the pressurizer lower shell, the staff identified the resolution of this issue as Open Item 4.7.4-1.

10 CFR 54.3 contains six criteria that must be satisfied for an analysis to be considered a time-limited aging analysis (TLAA). As a result of the information submitted in their July 7, 2003 letter, the evaluation of flaw growth for a crack that was left in place at the inside surface of the pressurizer and the impact of corrosion on the pressurizer nozzle meet these six criteria and should be considered a TLAA.

Section 5.1.4 of NEI 95-10, Revision 3 indicates that an applicant who elects to defer completing the evaluation of a TLAA at the time of a renewal application should submit the following details in the renewal application to support a conclusion that the effects of aging addressed by that TLAA will be managed for a specific structure or component:

1. Details concerning the methodology which will be used for TLAA evaluation,

2. Acceptance criteria that will be used to judge the adequacy of the structure or component, consistent with the CLB, when the TLAA evaluation or analysis is performed,

3. Corrective actions that the applicant could perform to provide reasonable assurance that the component in question will perform its intended function when called upon or will not be outside of its design basis established by the plant's CLB, and

4. Identification of when the completed TLAA evaluation will be submitted to ensure that the necessary evaluation will be performed before the structure of component in question would not be able to perform its intended functions established by the CLB.

The July 7, 2003 letter contains a methodology and criteria for evaluating the impact of flaw growth on the original crack that was left in place at the inside surface of the pressurizer and specifies that the impact of corrosion will be included in the evaluation. The methodology is summarized as follows:

1. Design drawings are reviewed to determine vessel, nozzle and J-weld dimensions and materials.

2. The initial flaw size to be used in the evaluation is calculated.

3. Manufacturing records are reviewed to determine the reference temperature (RT_{NDT}) of the base metal at the location of interest.

4. Design operation transients are reviewed to determine their appropriateness for use in the generation of stresses for use in the flaw analysis.

5. When the design transients are not appropriate, a realistic bounding transient is developed for analysis purposes.

6. Thermal transient analyses are performed to determine through-wall temperatures for use in the stress analysis.

7. Stress analyses are performed at various time points during each plant operating event of interest.

8. Pressure and mechanical load stresses are calculated.

9. A survey of the combined pressure, thermal and mechanical stresses is conducted to determine the limiting time point for evaluation.

10. Stresses are determined to calculate the applied stress intensity factor, K_I.

11. The applied stress intensity factor is calculated for comparison to allowable values.

12. Fatigue crack growth of the flaw is calculated over the 60 years.

13. The final flaw size is used to confirm flaw stability over the remaining life of the plant.

14. The flaw stability checks defined above are performed for normal and upset conditions and emergency and faulted conditions using the respective allowables defined per ASME Section XI.

15. Primary stress limits per NB-3000 are checked considering the effect of the final flaw size.

This methodology is acceptable because it will determine the impact of plant operation, design transients, material fracture resistance, and flaw growth on pressurizer integrity for the period of extended operation.

The flaw will be acceptable if it satisfies the linear elastic fracture mechanics criteria in ASME Code Section XI, IWB-3611 or IWB-3612, or elastic-plastic fracture mechanics criteria in ASME

Code Section XI, Appendix K, articles K-2200, K-2300, and K-2400. Since the acceptance criteria are in accordance with ASME Code criteria, they are acceptable for use in this TLAA.

The applicant's corrective action includes assuring that the pressure at any temperature should not be any higher than the higher of the following two limits:

1. The saturation pressure plus 200 psi, and

2. 350 psi and the maximum rate of temperature decrease is 200 ^{o}F/hr.

By limiting pressure and the maximum rate of decrease in temperature for the pressurizer, the corrective action will limit the stresses on the flaw remaining in the pressurizer and provides reasonable assurance that the component in question will perform its intended function when called upon or will not be outside of the design basis established by the plant's CLB.

The applicant indicates that the TLAA for this issue will be completed before the period of extended operation and the analyses will be submitted for staff review and approval.

By satisfying the criteria in Section 5.1.4 of NEI 95-10, Revision 3, the staff concludes that the applicant has provided a methodology and criteria for assuring that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation and has satisfied the TLAA criteria 10 CFR 54.21(c)(1)(iii). The applicant's commitment to complete the evaluation is documented in Appendix A of this SER.

On the basis of the staff's evaluation described above, the summary description for the "Pressurizer Alloy 600 J-Weld Left in Place" described in the USAR Supplement (LRA, Appendix A.3.6.4) is acceptable. Open Item 4.7.4-1 is closed.

4.7.4.3 Conclusions

The staff has reviewed the information provided regarding the TLAA for the Alloy 600 repair in the temperature nozzle in the pressurizer lower shell. On the basis of this evaluation and the licensee's commitment to complete and submit the evaluation of the small-bore instrument nozzle J-weld region at the repaired instrument nozzle to the NRC before the period of extended operation, the staff concludes that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation and has satisfied the TLAA criteria 10 CFR 54.21(c)(1)(iii).

In addition, the staff concludes that the applicant's USAR Supplement provides an adequate description of the analysis to be performed to evaluate the pressurizer Alloy 600 J-Weld left in place, as required by 10 CFR 54.21(d).

4.8 Evaluation Findings

The staff has reviewed the information in Section 4 of the LRA. On the basis of its review, the staff concludes that the applicant has provided an adequate list of TLAAs, as defined in 10 CFR 54.3. Further, the staff concludes that the applicant has demonstrated or will demonstrate that the TLAAs (1) will remain valid for the period of extended operation, as required by 10 CFR 54.21(c)(1)(i); (2) have been projected to the end of the period of extended operation, as required by 10 CFR 54.21(c)(1)(ii); or (3) the aging effects will be adequately managed for the period of extended operation, as required by 10 CFR 54.21(c)(1)(iii). In addition, the staff concludes that there are no plant-specific exemptions in effect that are based on TLAAs, as required by 10 CFR 54.21(c)(2). Finally, the staff has reviewed the USAR Supplements and concludes that the applicant has provided or will provide adequate descriptions of the TLAAs credited for license renewal, as required by 10 CFR 54.21(d).

On this basis, the staff finds that the aging effects associated with the structures and components subject to TLAAs are addressed such that the structures and components will perform their intended functions in accordance with the current licensing basis during the period of extended operation, as required by 10 CFR 54.21(a)(3).