4. TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

In Section 4.1 of the LRA, the applicant describes its identification of time-limited aging analyses. The staff reviewed this section of the LRA to determine whether the applicant has identified the TLAAs, as required by 10 CFR 54.21(c).

4.1.1 Summary of Technical Information in the Application

The applicant evaluated calculations for St. Lucie Units 1 and 2 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 from the technical specifications (TS), updated final safety analysis reports (UFSARs), and docketed licensing correspondence. The applicant identified the following TLAAs in Table 4.1-1 of the LRA:

- reactor vessel neutron embrittlement, including analyses for upper-shelf energy (USE), pressurized thermal shock (PTS), and pressure-temperature (P-T) limits
- metal fatigue, including analysis of American Society of Mechanical Engineers (ASME) Section III Class 1 components, ASME Class 2 and 3 components and American National Standards Institute (ANSI) B31.1 components
- environmental equipment qualification calculations
- containment penetration fatigue analyses
- leak-before-break (LBB) analyses
- crane load cycle limit
- Unit 1 core support barrel (CSB) repair fatigue analysis
- Unit 1 core support barrel (CSB) repair plug preload relaxation
- Alloy 600 instrument nozzle repairs

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

As indicated by the applicant, 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 licensee 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 current licensing basis (CLB)

Table 4.1-1 of the LRA did not identify pipe break postulation based on cumulative usage factor (CUF) as a TLAA. Section 3.6.2.2.1 of the St. Lucie Unit 2 UFSAR describes the criteria used to provide protection against pipe whip inside the containment. A part of the criteria specifies the postulation of pipe breaks at locations where the CUF exceeds 0.1. Although the fatigue usage factor calculation was identified as a TLAA, the pipe break criterion was not identified as a TLAA. However, the usage factor calculation used to identify postulated pipe break locations meets the definition of a TLAA, as specified in 10 CFR 54.3, and, therefore, the staff considers the associated criteria for pipe break postulation a TLAA. In the staff's request for additional information (RAI) 4.1-1, it requested that the applicant provide a description of the TLAA performed to address the pipe break criteria for St. Lucie Unit 2. The staff also requested the applicant to identify any pipe break postulations based on CUF at Unit 1 and describe the TLAA performed for these locations.

The applicant's October 10, 2002, response indicated that pipe breaks had been postulated in Class 1 piping at locations where the CUF exceeds 0.1 at Unit 2. The applicant also indicated that it did not expect the number of design transients assumed in these CUF calculations to be exceeded in 60 years of plant operation. Therefore, the CUF calculations which form the basis for the Unit 2 pipe break postulations remain valid for the period of extended operation. The applicant's evaluation provides an acceptable TLAA for Unit 2 in accordance with the requirements of 10 CFR 54.21(c). The applicant indicated that Unit 1 does not use CUF values from the fatigue analysis to determine postulated pipe break locations, and, therefore, the Unit 1 pipe break criteria do not meet the definition of a TLAA, as provided in 10 CFR 54.3. The staff agrees with the applicant's conclusion.

Table 4.1-1 of the LRA did not identify fatigue of the reactor coolant pump (RCP) flywheel as a TLAA. In RAI 4.1-1, the staff asked the applicant to indicate whether fatigue crack growth calculations were performed for the Unit 1 and 2 RCP flywheels.

The applicant's October 10, 2002, response indicated that a reference to RCP flywheel crack growth calculations was found in Section 5.5.5.3 of the Unit 1 UFSAR. According to the applicant, RCP flywheel crack growth calculations indicate that the number of starting cycles required to cause a reasonably small crack to grow to critical size is more than 100,000. The applicant indicated that the number of starting cycles required to cause a crack to grow to critical size is far greater than the number of expected RCP pump starts for the period of extended operation. Therefore, the crack growth evaluation remains valid for the period of extended operation. The staff finds the applicant's flywheel crack growth evaluation meets the definition of a TLAA, as provided in 10 CFR 54.3. The applicant's evaluation, described above, provides an acceptable TLAA for the Unit 1 RCP flywheel crack growth calculation in accordance with the requirements of 10 CFR 54.21(c). The applicant indicated that a review of the Unit 2 licensing basis documentation did not identify or reference fatigue crack calculations for the flywheels. Therefore, there are no TLAAs associated with the Unit 2 RCP flywheels.

4.1.3 Conclusions

The staff has reviewed the information provided in Section 4.1 of the LRA. The staff concludes that, with the inclusion of the pipe break criteria for Unit 2 and the RCP flywheel crack growth analysis for Unit 1, the applicant has provided an acceptable list of TLAAs as defined in 10 CFR 54.3, and that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA, as defined in 10 CFR 54.3.

4.2 Reactor Vessel Neutron Embrittlement

The application includes three TLAAs for evaluation of the reactor vessel (RV) beltline materials, including: (1) calculation of the end-of-extended-life Charpy USE value (C_vUSE values) for each beltline material, (2) calculation of the end-of-extended-life PTS reference temperature (RT) value (i.e., RT_{PTS} values) for each beltline material, and (3) a calculation of P-T limits. Each analysis has been updated to consider 20 years of additional plant operation at power. The TLAAs take into account the effects of the additional extended-operating-period neutron irradiation on the previous calculated end-of-life C_vUSE, the RT_{PTS}, and P-T limit values for the Units 1 and 2 RVs and conservatively base the evaluations through 54 effective full power years (EFPY) of power operation.

4.2.1 Upper-Shelf Energy

Appendix G to 10 CFR Part 50 requires that RV beltline materials have C_vUSE 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 foot-pounds (ft-lb) (102 J) initially, and must maintain C_vUSE values throughout the life of the vessel of no less than 50 ft-lb (68 J). However, C_vUSE values below these criteria may be acceptable if it is demonstrated, in a manner approved by the Director of the Office of Nuclear Reactor Regulation, that the lower values of C_vUSE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of C_vUSE values and describes two methods for determining C_vUSE values for RV beltline materials, depending on whether a given RV beltline material is represented in the plant's Reactor Vessel Material Surveillance Program (i.e., 10 CFR Part 50, Appendix H program).

4.2.1.1 Summary of Technical Information in the Application

Section 4.2.1 of the application addressed the requirement that RV beltline materials must maintain a C_vUSE value of not less than 50 ft-lbs throughout the life of the vessel, unless it is demonstrated, in a manner approved by the Director of the Office of Nuclear Reactor Regulation, that lower values of C_vUSE will provide margins of safety against fracture that are equivalent to those required by Appendix G of Section XI of the ASME Code. The applicant stated that the C_vUSE values have been calculated through the period of extended operation, using guidance from Regulatory Guide 1.99, Revision 2. A value of 54 EFPY was used as the end-of-life criterion for the RV. The application contains the information derived from the C_vUSE analysis. It includes a list of all beltline materials, the weight percent copper in the steel, the end-of-life fluence for the RV located one-quarter from the vessel's inside surface (i.e., 1/4T thickness of the vessel), and the initial and final C_vUSE values. The applicant concludes that the end-of-life C_vUSE results are above the screening criterion of 50 ft-lb (68 J). The applicant states that the calculations have been projected through the period of extended operation and shown to meet the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.1.2 Staff Evaluation

The applicant summarized the end-of-extended operating period USE analyses for the Units 1 and 2 RV beltline materials in Tables 4.2-1 and 4.2-2, respectively, of the LRA. Since all of the C_vUSE values are above the 50 ft-lb (68 J) screening criterion, the staff finds that, with respect to C_vUSE, the Florida Power and Light Company (FPL) RVs have sufficient margin to perform their intended function through the end of the period of extended operation.

The staff performed an independent calculation of the end-of-extended life C_vUSE values for the beltline materials used to fabricate the St. Lucie RVs. For those RV beltline materials that were not represented in the applicant's RV material surveillance program, the staff applied Regulatory Position 1.2 of Regulatory Guide 1.99, Revision 2, to estimate the percent loss of C_vUSE as a function of copper content and neutron fluence for the beltline materials, as evaluated using the 54 EFPY end-of-extended life fluence. For RV materials represented in the applicant's RV material surveillance program, the staff applied Regulatory Position 2.2 as its bases for estimating the percentage drop in C_vUSE. The staff confirmed that all RV beltline materials will continue to satisfy the C_vUSE value requirements of 10 CFR Part 50, Appendix G, through the end-of-extended operating lives for the St. Lucie reactor units. The staff, therefore, concludes that the applicant's TLAA for calculating the C_vUSE values of the RV beltline materials is acceptable because it meets the requirements of 10 CFR 54.21(c)(1)(ii) and will ensure that the RV materials will have adequate USE levels and fracture toughness through the end-of-extended period of operation.

4.2.2 Pressurized Thermal Shock

Section 50.61 of 10 CFR Part 50 provides the fracture toughness requirements protecting the RVs of pressurized-water reactors (PWRs) against the consequences of PTS. Licensees are required to perform an assessment of the RV materials' projected values of the PTS reference temperature, RT_{PTS}, through the end of their operating license. If approved for license renewal, this would include TLAAs for PTS up through the end-of-extended operating terms for the St. Lucie units. Upon approval of its application for an period of extended operation for St. Lucie Units 1 and 2, this period would be 54 EFPY. 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., @ RT_{NDT} value), and an additional margin value to account for uncertainties (i.e., M value). Section 50.61 of 10 CFR Part 50 also provides screening criteria against which the calculated RT_{PTS} values are to be evaluated. For RV 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 RV 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 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," provides an expanded discussion regarding the calculations of RT_{PTS} values and describes two methods for determining RT_{PTS} for RV materials, depending on whether a given RV beltline material is represented in the plant's RV material surveillance program (i.e., 10 CFR Part 50, Appendix H program).

4.2.2.1 Summary of Technical Information in the Application

Section 4.2.2 of the LRA addresses the 10 CFR 50.61 requirement that the RV be protected against PTS. The applicant states that the screening criteria in 10 CFR 50.61 are 270 °F for plates, forgings, and axial welds and 300 °F for circumferential welds. According to the regulation, if the calculated RT_{PTS} values for the beltline materials are less than the screening criteria, then the RV is acceptable with respect to risk of failure during postulated thermal shock transients. In this part of the application, the applicant describes the projected values of RT_{PTS} over the period of extended operation (54 EFPY) to demonstrate that the screening criteria are not violated. The applicant states that this analysis has been carried out and that the results do not exceed the screening criteria. The applicant states that the calculations have been projected through the period of extended operation and shown to meet the requirements of 10 CFR 54.21(c)(1)(ii).

4.2.2.2 Staff Evaluation

The applicant provided its end-of-extended operating PTS assessments for the Units 1 and 2 beltline RV materials in Tables 4.2-3 and 4.2-4, respectively, of the LRA. The staff performed an independent calculation of the RT_{PTS} values for the Units 1 and 2 beltline RV materials, based on the projected end-of-extended operating term (54 EFPY) neutron fluences for the materials. In reviewing the applicant's description of the PTS analysis, the staff examined the data and results of the analysis, as summarized in Tables 4.2-3 and 4.2-4 of the LRA. The staff's calculated RT_{PTS} values for the RV beltline materials were within 2 degrees of the applicant's calculated RT_{PTS} values. Both the staff's and the applicant's PTS analyses confirm that the RT_{PTS} values for the St. Lucie Units 1 and 2 beltline materials will remain under the PTS screening criteria of 10 CFR 50.61 through the period of extended operating periods for the units.

For the Unit 1 RV, the staff determined that the lower shell axial welds 3-203 A, B, and C are the most limiting materials and calculated the end-of-extended-operating-term RT_{PTS} value for these materials to be 240 °F. For the St. Lucie Unit 2 RV, the staff determined that intermediate shell plate M-605-2 is the most limiting material and calculated the end-of-extended operating term RT_{PTS} value for this material to be 174 °F. All of these materials meet the 10 CFR 50.61 screening criteria for longitudinal weld and base metal materials of 270 °F. Based on these considerations, the staff finds the applicant's TLAAs for protecting the Units 1 and 2 vessels against PTS to be acceptable because the staff confirmed that the RT_{PTS} values for all Units 1 and 2 RV beltline materials remain below the screening criteria of 10 CFR 50.61. The staff therefore concludes that the applicant's TLAA for calculating the RT_{PTS} values for the Units 1 and 2 RV beltline materials is acceptable because it meets the requirements of 10 CFR 54.21(c)(1)(ii) and will ensure that the RV materials will have sufficient protection against PTS events through the end-of-period of extended operations.

4.2.3 Pressure-Temperature Limits

The requirements in 10 CFR Part 50, Appendix G, are designed to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on NRC regulations and guidance. Appendix G to 10 CFR Part 50 requires that P-T limit curves be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P-T limit curves. SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The critical locations in the RV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

Operation of the reactor cooling system (RCS) is also limited by the net positive suction curves for the RCPs. These curves specify the minimum pressure required to operate the RCPs. Therefore, in order to heat up and cool down, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G P-T limits and the net positive suction curves of the RCPs.

4.2.3.1 Summary of Technical Information in the Application

In Section 4.2.3 of the LRA, the applicant addresses the requirement in 10 CFR Part 50, Appendix G, that normal operations—including heatup, cooldown, and transient operating conditions—and pressure-test operations of the RV be accomplished within established P-T limits. These limits are established by calculations that utilize the materials and fluence data obtained through the unit-specific reactor surveillance capsule program.

4.2.3.2 Staff Evaluation

The P-T limits are established by calculations that utilize the materials and fluence data obtained through the unit-specific reactor surveillance capsule program.

Normally, the P-T limits are calculated for several years into the future and remain valid for an established period of time, not to exceed the current operating license expiration. The current P-T limit curves for St. Lucie Unit 1 are acceptable through 23.6 EFPY of power operation. The current P-T limit curves for St. Lucie Unit 2 are acceptable though 21.7 EFPY of power operation. Part 50.90 of 10 CFR Part 50 requires licensees to submit new P-T limit curves for operating reactors for review and have the curves approved and implemented into the TS for the reactor units prior to the expiration of the most current P-T limits curves approved in the TS. The applicant will be required to submit the extended-period-of-operation P-T limit curves for the Units 1 and 2 RVs, and have the curves approved against the criteria of 10 CFR Part 50,

Appendix G, and implemented into the TS prior to operation of the reactors during the extended operating terms for the units.

The staff will evaluate the extended-period-of-operation P-T limit curves for the Units 1 and 2 RVs prior to expiration of the current-operating-term P-T limit curves for the units. The staff's review of the extended-period-of-operation P-T limit curves, when submitted, will ensure that the operation of the units will be done in a manner that ensures the integrity of the RCS during the period of extended operations.

4.2.4 FSAR Supplement

The applicant's FSAR supplement for the TLAA on RV neutron embrittlement is provided in Section 18.3.1 of Appendices A1 and A2 for Units 1 and 2, respectively. The applicant's appropriate consideration of RV neutron embrittlement, including the effects of neutron irradiation on the PTS, USE, and P-T limit assessments for Units 1 and 2, constitutes the bases for the staff acceptance of the licensee's evaluation of the TLAA for the period of extended operation. On the basis of its review of the updated FSAR supplement, the staff concludes that the summary description of the applicant's actions to address RV neutron embrittlement on the Units 1 and 2 RV beltline materials for the period of extended operation is adequate.

4.2.5 Conclusions

The staff has reviewed the TLAAs regarding the maintenance of acceptable Charpy USE levels for the Units 1 and 2 RV materials and the ability of the Units 1 and 2 RVs 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 RV beltline materials as evaluated to the end-of-extended-operating periods for the units, and therefore satisfy the requirements of 10 CFR 54.21(c)(1)(ii) for 60 years of operation.

Prior to operation of the reactors during the extended period of operation, the applicant will submit the end-of-extended-operating term P-T limit curves for the reactor units. The staff's review of the extended-period-of-operation P-T limit curves, when submitted, will ensure that the operation of the RCS for the Units will be done in a manner that ensures the integrity of the RCS for the period of extended operation and that the curves will satisfy the requirements of 10 CFR Part 54.21(c)(1) for the period of extended operation. The staff also concludes that the UFSAR supplement contains an appropriate summary description of the RV neutron embrittlement TLAA evacuation for the period of extended operation.

4.3 Metal Fatigue

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity due to metal fatigue, initiating and propagating cracks in the material. The fatigue life of a component is a function of its material, its environment, and the number and magnitude of the applied cyclic loads. Fatigue was a design consideration for plant mechanical components in St. Lucie Units 1 and 2 and, consequently, fatigue is part of the CLB for these components. The applicant addresses the TLAA evaluations performed to address thermal and mechanical fatigue analyses of plant mechanical components in Section 4.3 of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has evaluated the TLAA in accordance with the requirements of 10 CFR 54.21(c)(1).

4.3.1 Summary of Technical Information in the Application

In Section 4.3.1 of the LRA, the applicant discussed the design requirements for components of the RCS at Units 1 and 2. The RVs, RV internals, pressurizers, steam generators, RCPs, and the Unit 2 reactor coolant piping were designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. The Unit 1 reactor coolant piping was designed in accordance with the requirements of ANSI B31.7, "Nuclear Power Piping." The applicant reanalyzed the Units 1 and 2 pressurizer surge lines in accordance with the requirements in Section III of the ASME Code in response to NRC Bulletin (BL) 88-11, "Pressurizer Surge Line Thermal Stratification." The applicant determined the fatigue usage factors for critical locations in the Units 1 and 2 Class 1 components using design cycles that were intended to be conservative and bounding for all foreseeable plant operations. The applicant noted that a review of Units 1 and 2 operating history indicates that the number cycles used in the design of these components bounds the number anticipated for the period of extended operation and, therefore, the analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant referenced the St. Lucie fatigue monitoring program (FMP) as a confirmatory program that assures that the design cycle limits are not exceeded during the period of extended operation. The FMP is described in Appendix B of the LRA.

In Section 4.3.2 of the LRA, the applicant discussed the design of ASME Class 2 and 3 components and ANSI B31.1 components. The requirements of these codes specify a stress reduction factor to 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 could exceed the 7000 cyclic limit during the period of extended operation, and that a further evaluation considering the projected number of cycles found that the analyses would be acceptable for the period of extended operation.

In Section 4.3.4 of the LRA, the applicant described the actions taken to address the issue of environmentally assisted fatigue. The applicant describes its evaluation of the following fatigue sensitive component locations:

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

The applicant discussed its proposed aging management program (AMP) to address pressurizer surge line fatigue at Units 1 and 2 during the period of extended operation. The applicant indicated that potential fatigue crack initiation and growth will be adequately managed during the period of extended operation by continued performance of the St. Lucie Inservice Inspection Program. The applicant indicated that several pressurizer surge line welds on both units have been examined ultrasonically with no reportable indications identified. The applicant indicated that additional inspections of the surge line welds will be performed prior to the period of extended operation, and that the results of these inspections will be used to determine the appropriate approach for addressing environmentally assisted fatigue of the surge lines.

4.3.2 Staff Evaluation

As discussed in the previous section, components of the Units 1 and 2 RCSs were designed to either the Class 1 requirements of the ASME Code or ANSI B31.7. The Class 1 requirements of both codes 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 Class 1 RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for ASME Class 1 components involves calculating the 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 noted that review of the St. Lucie plant operating history 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 data:

- the current number of operating cycles and a description of the method used to determine the number and severity of the design transient 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 UFSAR with the transients monitored by the FMP as described in Section B3.2.7 of the LRA, identifying any transients listed in the UFSAR that are not monitored by the FMP and explaining why it is not necessary to monitor these transients.

The applicant's October 10, 2002, response indicated that cycle counting has been performed since the startup of each unit. The applicant listed the UFSAR design transients for each unit in Tables 4.3-1.1 and 4.3-1.2 of the response. The applicant indicated that the design calculations were reviewed, and that design transients that result in a fatigue usage greater than 0.1 are monitored by the FMP. The applicant also indicated that transients associated with plant loading and unloading events were not monitored because Units 1 and 2 are not load-following plants and, therefore, the number of cycles used in the design is very conservative. The applicant's statement regarding the conservative number of design transients associated with plant loading and unloading events is consistent with the information presented in NUREG/CR-6260 for an older-vintage Combustion Engineering plant. The applicant provided comparisons of the number of design cycles with the number of transients projected for 60 years of plant operation at the monitored locations for each unit in Tables 4.3-1.3 and 4.3-1.4 of the response. The staff finds the applicant's criteria for selecting transients to be monitored by the FMP to be reasonable.

NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification," identified a concern regarding the potential temperature stratification and thermal striping in the pressurizer surge line. The applicant indicated that the pressurizer surge lines were analyzed in response to the bulletin. NRC BL 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," identified a concern regarding the potential for temperature stratification or temperature oscillations in unisolable sections of piping attached to the RCS. In RAI 4.3-2, the staff requested the applicant to describe the actions taken to address NRC BL 88-08 during the period of extended operation. The applicant's October 10, 2002, response indicated that no fatigue calculations had been performed to address NRC BL 88-08. Therefore, no additional actions are required to address this bulletin during the period of extended operation.

The applicant indicated that the steam generators, pressurizers, RVs, RCPs, control rod drive mechanisms, and all RCS piping have been evaluated and the results of the analyses have been determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). 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 [Nuclear Energy Institute] and EPRI [Electric Power Research Institute]), 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 plant, for 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 Design Curves of Austenitic Stainless Steels," were considered in the evaluation. In RAI 4.3-3, 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 October 10, 2002, response provides the St. Lucie Units 1 and 2 plant-specific usage factors that include environmental effects for the six components listed in NUREG/CR-6260 in Tables 4.3-3.1 and 4.3-3.2. The applicant calculated an environmental multiplier for the six components and applied that multiplier to the design CUF to obtain a CUF that accounts for environmental effects. The applicant's evaluation indicates that the CUFs, including environmental effects, are expected to be below the ASME Code limit of 1.0 at all locations except for the surge lines at both units for 60 years of plant operation.

The staff compared the results of the applicant's evaluation with the results presented in NUREG/CR-6260 for an older-vintage Combustion Engineering plant. NUREG/CR-6260 identified three locations where the CUF, including environmental effects, may be exceeded based on the number of design transient cycles. These locations include the surge line, the charging nozzle, and the safety injection nozzle. The applicant indicated that the charging and safety injection nozzles at Units 1 and 2 are carbon steel as opposed to the stainless steel listed for the charging and safety injection nozzles in NUREG/CR-6260. The environmental multiplier for carbon steel is less than the environmental multiplier for stainless steel in a low oxygen (PWR) environment. Application of carbon steel environmental multipliers for the NUREG/CR-6260 charging and safety injection nozzles would result in CUFs less than 1.0. In its November 27, 2002, supplemental response, the applicant indicated that the location of highest fatigue usage on the Unit 2 charging nozzle occurs at the piping side of the safe end which is stainless steel. The applicant's evaluation of this location, using the appropriate stainless steel environmental multiplier, indicates the safe end CUF is expected to be less than 1.0 for sixty years of plant operation. This would leave the pressurizer surge line as the only location where the CUF, including environmental effects, exceeds 1.0. On the basis of the comparison of the results of the applicant's evaluation with the results presented in NUREG/CR-6260, the staff concludes that the results of the applicant's evaluation are reasonable.

The applicant indicates that the pressurizer surge line elbows required further evaluation for environmental fatigue during the period of extended operation. The applicant further indicated that it would use an AMP to address fatigue of the surge line during the period of extended operation. The AMP would rely on the Inservice Inspection Program to manage surge line fatigue during the period of extended operation. The applicant noted that no indications have been identified as a result of the weld examinations performed to date. The applicant also indicated that additional surge line weld examinations will be performed prior to the period of extended operation. The applicant indicated that the results of the examinations would be used to develop the approach for addressing environmentally assisted fatigue of the surge lines. This approach could include one or more of the following:

- further refinement of the fatigue analysis to lower the CUF(s) to below 1.0
- repair of the affected locations
- replacement of the affected locations
- management 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 indicated that if the last option is selected, the inspection details, including scope, qualification, method, and frequency, will be provided to the NRC for review prior to the period of extended operation. The staff finds that the applicant's proposed options provide acceptable plant-specific approaches to address environmentally assisted fatigue of the St. Lucie Units 1 and 2 pressurizer surge lines during the period of extended operation in accordance with 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), these options need to be included in the FSAR supplement (Confirmatory Item 4.3.1-1).

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 applicant indicates that its review of plant operating practices found that most B31.1 systems in the scope of license renewal are subject to continuous steady-state operation and the temperature only varies as a result of plant heatup and cooldown, during plant transients, or for periodic testing. Therefore, the applicant concluded that the analyses of these piping components remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). However, the applicant indicated that the reactor coolant hot leg sample lines on both units could be subject to greater than 7,000 cycles during the period of extended operation. The applicant indicated that the sample piping and tubing were reevaluated for the number of expected cycles and found acceptable for the period of extended operation. Therefore, the applicant concluded that these analyses have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii). The staff finds the applicant's evaluation acceptable.

4.3.3 UFSAR Supplement

The applicant's Units 1 and 2 UFSAR supplements for metal fatigue are provided in Appendices A1 and A2 of the LRA, respectively. Section 18.2.7 of the Unit 1 supplement and Section 18.2.6 of the Unit 2 supplement describe the FMP. Section 18.3.2 of both UFSAR supplements describe the applicant's TLAA for metal fatigue. Section 18.3.2.3 includes a discussion of the applicant's proposed AMP for the surge line. However, the discussion does not include the applicant's commitment that this program be reviewed and approved by the staff prior to the period of extended operation. The applicant should update the UFSAR supplements to include the approach to address environmental fatigue of the surge line as discussed in the previous section of this safety evaluation. (Confirmatory Item 4.3.1-1).

4.3.4 Conclusions

On the basis of its evaluations of Units 1 and 2 components, the applicant concludes that the fatigue analysis of RCS components and piping remain valid for the period of extended operation. The applicant also has a FMP that maintains a record of the transients used in the fatigue analyses of RCS components. That process will continue during the period of extended operation.

Pending resolution of the confirmatory item identified in this SE, the staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal fatigue TLAA the analysis remains valid for the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the metal fatigue TLAA evaluation for the period of extended operation.

4.4 Environmental Qualification

The aging (or qualified life) analysis for electrical/I&C components included as part of the EQ program (required by 10 CFR 50.49) that involve time-limited assumptions (as defined by the current operating term for the St. Lucie plant, i.e., 40 years) meet the 10 CFR 54.3 definition for time-limited aging analyses (TLAAs) and are thus considered TLAAs for license renewal. The existing thermal, radiation, and wear cycle aging analyses required by 10 CFR 50.49 for plant electrical/I&C components identified as TLAAs have been evaluated by the applicant pursuant to 10 CFR 54.21(c)(1)(ii) to determine if they can be projected to the end of the period of extended operation by re-analysis or additional analysis.

The staff reviewed Section 4.4, “Environmental Qualification of Electric Equipment” of the LRA to determine whether there continues to be reasonable assurance that electrical/I&C components (after re-analysis for a 60 year qualified life) will be capable of performing their required safety function pursuant with 10 CFR 54.21(c)(1)(ii).

4.4.1 Summary of Technical Information in the Application

In Section 4.4 of the LRA, the applicant describes its process (which is encompassed as part of the existing 10 CFR 50.49 EQ program) for analysis (and also for re-analysis) of electrical/I&C component’s qualified life. In addition, the applicant provides the results of its re-analysis to project the current 40 year qualified life to 60 years.

The applicant describes its process for re-analysis of qualified life of electrical/I&C components using the environmental service conditions that are applicable to the components. The environmental service conditions are divided into normal and accident service conditions. 10 CFR 50.49 requires that all significant aging effects from normal service conditions be considered as part of the qualified life analysis. Significant aging effects include the expected thermal aging effects from normal temperature exposure, any radiation effects during normal plant operation, and mechanical cycle effects as applicable. 10 CFR 50.49 also requires evaluation of the effects of any harsh environments the electrical/I&C components could be exposed to under accident conditions.

The description provided by the applicant of its re-analysis of qualified life based on normal service conditions for 60 years is as follows:

- Thermal-Aging Considerations - The specific analyses for thermal aging have been reviewed by the applicant to confirm that the existing qualified life calculations remain valid for the extended period of operation or a re-calculation projects the component’s qualified life to encompass the extended period of operation.

- Radiation-Aging Considerations - The St. Lucie EQ Program has established bounding radiation dose qualification values for all EQ components. These bounding radiation dose values were determined through testing. To verify that these bounding radiation test values are acceptable for the period of extended operation, the total integrated dose values for the 60 year period were determined and then compared to these bounding radiation test values. The total integrated dose for the 60-year period is determined by adding 60-year normal operating dose (i.e., 1.5 times the 40-year normal operating dose) to the established accident dose for the component.

- Mechanical-Cycle Aging Considerations - The expected wear cycles to which electro-mechanical components will be subject to over a 60 year period were found (with margin) to be less than the wear cycles to which components were subjected to prior to the performance of design basis accident testing.

In summary, the applicant credits the EQ program as part of the screening process for ensuring the qualified life of electrical/I&C components within the scope of 10 CFR 50.49 is maintained. The EQ program establishes the aging limit (qualified life) for each installed environmentally qualified component. The EQ program qualified life analysis is considered to be TLAA for St. Lucie Units 1 and 2. Pursuant to 10 CFR 54.21(c)(1)(ii), a re-analyses was performed to demonstrate that the qualified life for electrical/I&C components has been projected to 60 years (i.e., the end of the period of extended operation). This re-analysis demonstrates that there is reasonable assurance that electrical/I&C components will be capable of performing their required safety function for 60 years and thus for the period of extended operation.

4.4.2 Staff Evaluation

4.4.2.1 Radiation Aging

As part of the original type test for components to demonstrate their EQ for 40 years of operation, conservative (or bounding) radiation test values were selected (consistent with industry practice) to encompass the possibility for higher than normally expected radiation dose values if they were to occur due to plant modifications and events. Conservative radiation test values provide, if needed, the option for re-analysis (versus equipment replacement) to demonstrate continued EQ in accordance with the requirements of 10 CFR 50.49(e)(4). 10 CFR 50.49(e)(4) requires that the radiation environment be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

To extend EQ from 40 to 60 years, the conservative (or bounding) radiation test values (included as part of the original type test of components to demonstrate their EQ) were utilized. To verify that the original radiation test values are acceptable for the period of extended operation, the total integrated dose values for the 60 year period were determined and then compared to the original radiation test values. The total integrated dose for the 60-year period is determined by adding 60-year normal operating dose (i.e., 1.5 times the 40-year normal operating dose) to the established accident dose for the component.

At St. Lucie to establish the normal operating dose, the maximum operating value for radiation was used as part of an EQ re-analysis for establishing a 60 year qualified life. The maximum operating value is based on an area radiation dose rate values for continuous operation assuming 1% failed fuel. The total integrated dose is determined by adding the 60-year normal operating dose to the appropriate accident dose for the specific location of the component. If the new total integrated dose for the 60 year period is less than the original radiation test values, components are considered acceptably qualified for 60 years (i.e., the extended period for license renewal).

The expected radiation dose to which components will be exposed over a 60 year period plus the accident radiation dose (i.e., the new total integrated radiation dose was found (with margin) to be less than the radiation dose to which components were exposed prior to design basis accident testing. Thus, there continues to be reasonable confidence that components will be capable of performing their required safety function if needed for 60 years. The staff concluded that the radiation aging for extending qualified life of components is acceptable, since it meets the requirements of 10 CFR 54.22(c)(ii).

4.4.2.2 Temperature Aging

As part of the original type test for components to demonstrate their EQ for 40 years of operation, conservative temperature test values were selected (consistent with industry practice) to represent normal operating temperatures. Conservative temperature test values provide, if needed, the option for re-analysis based on the Arrhenius method (versus equipment replacement) to demonstrate continued EQ in accordance with the requirements of 10 CFR 50.49(e)(5). 10 CFR 50.49(e)(5) requires that components qualified by test must be preconditioned by natural or artificial (accelerated) aging to their end-of-installed life condition. To meet this requirement, re-analysis must show that when the conservatism included to account for normal operating temperatures is reduced or eliminated, the component can be shown to have been aged (i.e., preconditioned by artificial (accelerated) aging to its end-of-life condition) to the equivalent of 60 years.

In Section 4.4 of the LRA, the applicant indicates that environmental qualification (EQ) acceptance criteria for temperature aging is the component's maximum required operating temperature. If the maximum operating temperature is equal to or less then the temperature to which the component was qualified by test, the component is considered qualified.

Each component's qualification temperature used for aging to a qualified life of 40 years was re-calculated for 60 years using the Arrhenius method. The St. Lucie Units 1 and 2 Technical Specifications temperature limit for inside each unit's containment is 120 °F. By plant procedure, the temperature is limited to 115 °F on both Units. Normally the 120 °F temperature is used for the in-containment aging calculations, however, the plant procedures limit of 115 °F is used for some components in Unit 2. Because the aging calculation for Unit 1 assumes a continuous temperature of 120 °F (which exceeds the component's maximum required operating temperature of 115 °F by 5 °F), takes into account the component's self heating, and does not credit seasonal and shutdown temperature reductions, significant margin exists to ensure that the qualified life of EQ components inside containment is not exceeded. For components in Unit 2 where the 115 °F temperature is used as the qualification temperature, significant margin also exists to ensure that the qualified life of EQ components inside containment is not exceeded. Significant margin exists because (a) the aging calculation assumes a continuous temperature of 115 °F (which is equal to the component's maximum required operating temperature), (b) components are located in containment at an elevation that is lower than the temperature detectors used to establish the 115 °F operating limit and thus components will be subject to an actual temperature that is less than 115 °F, and (c) the aging calculation takes into account the component's self heating and does not credit seasonal and shutdown temperature reductions. For areas outside containment, the aging calculations are based on a temperature of 104 °F. Because the aging calculation assumes a continuous temperature of 104 °F which is significantly higher than the average temperatures that would normally be expected to exist outside containment, significant margin exists to ensure that the qualified life of EQ components outside containment is not exceeded. In addition, no change of a component's activation energy (determined and utilized as part of the original aging calculation for 40 years) was used in the re-calculation for 60 years.

For those circumstances in which a component's maximum required operating temperature is equal to the temperature to which it had been tested to demonstrate EQ, the staff was concerned that there may be no margin to account for the uncertainties of the Arrhenius method. The applicant by letter dated October 10, 2002 (in response to a July 1, 2002 request for additional information) indicated the following: The maximum operating temperatures referred to in the LRA are the 104 °F design ambient for outside the Containments, and the 120 °F design ambient (Unit 1) and 115 °F design ambient (Unit 2) inside the Containments used to calculate the qualified life of EQ components. Section 4.4 also indicates that EQ components are assumed to be exposed to continuous design ambient temperatures (104 °F, 120 °F, or 115 °F, as appropriate), and that the evaluation does not credit lower temperatures due to seasonal/daily temperature changes or temperature changes associated with unit shutdown. These seasonal and shutdown reductions in temperature are more than adequate to account for the uncertainties of the Arrhenius Methodology when considering that the EQ components are exposed to a higher continuous design ambient temperature conditions. As an additional conservatism, continuous self-heating is also added to the design ambient temperatures.

The staff agrees that the average operating temperature of components due to seasonal/daily temperature changes or temperature changes associated with unit shutdown over a 60 year period will be less than the maximum required operating temperature to which Arrhenius method was applied. The difference between the average operating temperature and the maximum continuous design temperature to which components are qualified can therefore be considered sufficient to account for the uncertainties of the Arrhenius Methodology. The applicant's EQ acceptance criteria for establishing temperature aging (i.e., if the maximum operating temperature is equal to or less then the temperature to which the component was qualified by test, the component is considered qualified) is therefore considered acceptable.

The expected temperature to which components will be exposed over a 60 year period was found (with margin) to be less than the equivalent temperature (determined by the Arrhenius Methodology) to which components were exposed prior to design basis accident testing. In addition, no change of a component's activation energy (determined and utilized as part of the original aging calculation for 40 years as determined by the Arrhenius Methodology) was used in the re-calculation for 60 years. Thus, there continues to be reasonable assurance that components will be capable of performing their required safety function if needed for 60 years. The staff concludes that the temperature aging for extending qualified life of components is acceptable since it meets the requirements of 10 CFR 21(c)(ii).

4.4.2.3 Wear Cycle Aging

Wear cycle aging mechanically ages the electro-mechanical components to the end of their qualified lives prior to performing design basis accident testing. The EQ components at St. Lucie Units 1 and 2 where wear is a consideration are motors and solenoid valves.

EQ motors are either normally energized or in a standby mode during normal operation.

Standby components are tested once a month with preventive maintenance every 18 months. This results in less than 2000 cycles for valve operators and less than 1000 cycles for other motors over a 60-year life. This is less than the 2000 cycles that was performed during valve operator EQ testing. The motors considered continuous duty in the Environmental Qualification Program are the Units 1 and 2 containment fan cooler motors, the Units 1 and 2 charging pump motors, and certain Unit 2 ventilation fan motors. The qualification of the electro-mechanical components of these motors is maintained through a combination of maintenance required by the conditions in the test report (e.g., periodic replacement of seals that were only aged for ten years prior to qualification testing), and maintenance recommended by the vendor (e.g., overhaul a motor after 25,000 hours of operation or every 5 years whichever comes first). The frequency of maintenance for these components is normally governed by the maintenance requirements of the vendor rather than by any restrictions that are required by the EQ test report.

Depending on the application, solenoid valves can be cycled significantly more often than motors. The solenoid valve vendors, ASCO, Target Rock, and Valcor, cycled their valves from 18,000 to 50,000 times during their EQ testing. Of these three solenoid valves used in EQ applications at St. Lucie, only ASCO solenoid valves are used in high cycle applications. ASCO solenoid valves that experience a high cycle rate are classified as normally energized. As identified in the EQ evaluations, normally energized solenoid valves reach the end of their thermal qualified lives prior to 40 years. Therefore, they will be replaced periodically when they reach the end of their qualified lives. Thus, their qualification for life cycles is not considered to be a TLAA. Normally de-energized solenoid valves are operated the same as any other standby component, thereby establishing acceptability for 60 years.

The expected wear cycles to which electro-mechanical components will be subject to over a 60 year period was found (with margin) to be less than the wear cycles to which components were subjected to prior to the performance of design basis accident testing. Thus, there continues to be reasonable assurance that electro-mechanical components will be capable of performing their required safety function for 60 years. The staff concluded that the wear cycle aging for extending qualified life of electro-mechanical components is acceptable since it meets the requirements in 10 CFR 54.21(c)(i).

4.4.3 FSAR Supplements

The staff reviewed Section 18.3.3, “Environmental Qualification,” of Appendix A1 and A2 to the St. Lucie Units 1 and 2 LRA and found descriptions of the above described EQ program for electrical/I&C component TLAA evaluations. These FSAR supplement descriptions provide a summary of the programs and activities for the evaluation of TLAA for electrical/I&C components, meet the requirements of 10 CFR 54.21(d), and are considered acceptable.

4.4.4 Conclusions

The staff has reviewed the information in Sections 4.4, 4.4.1, and 4.4.2 of the LRA. On the basis of this review, the staff concludes that the applicant (for electrical/I&C components that meet the definition for TLAA as defined in 10 CFR 54.3) has projected the TLAA (i.e., the 10 CFR 50.49 radiation, temperature, and wear cycle aging analyses) from the current 40 years to 60 years (i.e., to the end of the period of extended operation) as provided in 10 CFR

54.21(c)(1)(ii). In addition, the staff concludes that the FSAR supplements contain a summary description of the programs and activities for the evaluation of TLAA as required by 10 CFR 54.21(d).

4.5 Metal Containment and Penetration Fatigue

4.5.1 Metal Containment Fatigue

4.5.1.1 Summary of Technical Information in the Application

The applicant states that no TLAAs exist for the St. Lucie Unit 1 and 2 containment vessels. These vessels are fabricated from welded steel plates. The criteria that are applied in the design of these vessels assure that the specified leak rate is not exceeded under the design basis accident conditions. The containment vessels are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III. No fatigue analysis was required for these applicable design codes. The applicant concludes that fatigue of the Units 1 and 2 containment vessels are not TLAAs.

4.5.1.2 Staff Evaluation

In RAI 4.5-1, the staff requested that the applicant indicate how the design criteria for the containment penetrations provide assurance that the specified leak rate for the containment vessels will not be exceeded. In a letter dated October 10, 2002, the applicant states that the Unit 1 containment vessel was designed to meet the requirements of ASME Section III 1968, Article 4, Subsection N-415, "Analysis for Cyclic Operation." The Unit 2 containment vessel was designed to meet the requirements of ASME Section III, Article 4, Subsection NB-3222.4, "Analysis for Cyclic Operation." These sections specify conditions for which analysis of cyclic service is not required. Meeting design requirements precludes cyclic fatigue cracking that may result in leakage. The applicant, therefore, did not perform fatigue analyses or TLAAs for these vessels. However, compliance with leakage design criteria is verified through periodic testing in accordance with ASME Section XI, Subsection IWE, "Inservice Inspection Program," as described in LRA Appendix B, Subsection 3.2.2.2. Compliance with the testing requirements assures containment integrity. Therefore, the staff finds the applicant's response acceptable.

4.5.2 Penetration Fatigue

4.5.2.1 Summary of Technical Information in the Application

The applicant states that the containment penetration bellows at Units 1 and 2 are specified to withstand a lifetime total of 7,000 cycles of expansion and compression as a result of maximum operating thermal expansion, and 200 cycles of seismic motion and differential settlement.

The containment penetrations are categorized into five types, depending on the operating conditions. The designs of penetration bellows, which must accommodate considerable or moderate thermal movements, are bounded by the thermal design limits of the associated piping systems. The other bellows do not require a thermal fatigue analysis because they are associated with cold penetrations, penetrations used for post-accident scenarios, or penetrations that are not subject to high temperatures. For these bellows, the applicant stated that the 200 cycles of differential settlement and seismic motion are also bounding for the period of extended operation.

The applicant states that the analyses associated with containment penetration bellows fatigue have been evaluated and determined to remain valid for the period of extended operation.

4.5.2.2 Staff Evaluation

The applicant stated that containment penetration bellows were specified to withstand a lifetime total of 7000 cycles of thermal expansion and compression, and 200 cycles due to other effects. In RAI 4.5-2, the staff requested that the applicant show that the specified cycles bound the period of extended operation.

In a letter dated October 10, 2002, responding to RAI 4.5-2, the applicant states that the piping systems associated with hot penetration bellows were evaluated in LRA Subsections 4.3.1 and 4.3.2 and found to be acceptable for the period of extended operation. The applicant also states that the methods used to confirm that the existing design cycles for Class 1 components are conservative and bounding for extended operation are described. Four St. Lucie Unit 1 containment penetrations associated with safety injection piping are designed to ASME Section III Class 1 requirements. The cycles that these piping components are subjected to are monitored as part of the FMP. Table 4.3-1.3 of the response to RAI 4.3.1 shows that the 7000 thermal expansion cycles bound the total number of thermal cycles assumed for the Class 1 safety injection piping during 60 years of operation.

The applicant states that the remainder of the Units 1 and 2 containment penetrations are associated with piping designed to ASME Section III, Class 2 requirements. In Subsection 4.3.2 of the LRA, the applicant indicates that these piping systems, as well as the containment penetrations associated with these piping systems, were originally designed for 7000 full temperature thermal cycles. The applicant performed an evaluation of these piping systems, reviewed plant operating procedures and practices, and concluded that these piping systems will not exceed 7000 equivalent full temperature thermal cycles during 60 years of operation. A review of plant operations to date also concluded that 200 cycles bound the expected number of seismic and differential settlement cycles that could occur during 60 years of operation. The staff finds this justification reasonable and acceptable because the current fatigue analyses limits will not be exceed during the period of extended operation because the designed number of cycles will not be exceeded.

In RAI 4.5.2, the staff also requested that the applicant describe the methods used to provide assurance that hot penetration bellows will withstand the cycles specified in the LRA under the corresponding thermal expansion loads and other loads for the period of extended operation. In its response, the applicant stated that the methods used to provide assurance that the penetration bellows will withstand the specified cycles include the FMP. Additional information regarding the design of the penetration bellows was also provided in Appendix 3G of the Unit 1 UFSAR. This information is also applicable to Unit 2. The staff finds that the applicant's response is acceptable because the margin in the design of the containment penetration bellows as compared to actual plant operations will be maintained for the period of extended operation.

In RAI 4.5-3, the staff asked if the containment penetration bellows are included within the scope of the St. Lucie FMP, or to provide justification for the exclusion if they are not. In a letter dated November 27, 2002, responding to RAI 4.5-3, the applicant states that the scope of the FMP, as described in LRA Appendix B, comprises RCS Class 1 components. The only Class 1 piping containment penetrations and associated bellows at Units 1 and 2 that are required to accommodate thermal expansion are those associated with Unit 1 safety injection piping. These penetrations are included in the scope of the FMP. Penetrations such as those associated with the Class 1 hot leg sample lines are not required to accommodate thermal expansion and are therefore not included in the FMP.

The containment penetrations and associated bellows for Class 2 piping systems at Units 1 and 2 were originally designed to accommodate 7000 equivalent full thermal cycles. The applicant stated that these piping systems will not exceed 7000 full thermal cycles during 60 years of operation. On this basis, the applicant stated that there is no need to monitor the thermal cycles of these penetrations and, therefore, the penetrations associated with Class 2 piping systems are not included in the scope of the FMP. The staff finds the applicant's response acceptable because the applicant demonstrated that the margin in the design of the penetrations will be maintained for the period of extended operation.

The staff has reviewed the UFSAR supplement, Section 18.3.4, for each unit, which provides a description of the containment penetration TLAA. The staff finds the description of the containment penetration fatigue evaluation sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.5.3 Conclusions

The staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for the metal containment and penetrations fatigue TLAA, the analyses remain valid and have been projected to the end of the period of extended operation. The staff also concludes that the USFAR supplements contain an appropriate summary description of the containment penetrations fatigue TLAA evaluation for the period of extended operation.

4.6 Plant-Specific Time-Limited Aging Analyses

In Section 4.6 of the LRA, the applicant provides its evaluation of St Lucie plant-specific time-limited aging analyses (TLAAs). The TLAA evaluated include:

- Leak-before-break for reactor coolant system piping
- Crane load cycle limit
- Unit 1 core support barrel repair
- Alloy 600 instrument nozzle repairs

The staff reviewed the site-specific TLAAs to verify the applicant's evaluations meet the requirements contained in 10 CFR 54.21(c)(1).

4.6.1 Leak-Before-Break

4.6.1.1 Summary of Technical Information in the Application

The applicant describes its LBB analysis in Section 4.6.1 of the LRA. The staff reviewed this section to determine whether the applicant provided adequate information to meet the requirements contained in 10 CFR 54.21(c) related to the TLAA for LBB for Units 1 and 2.

A successful application of LBB to the RCS primary loop piping is described in CEN-367-A, "Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems." This report provides the technical basis for evaluating two distinct postulated flaws in the main RCS piping using the two essential elements of the LBB methodology (1) the determination of the leakage flaw size under the normal loading condition and (2) the determination of the allowable flaw size under the faulted loading condition.

The applicant states that there are two considerations for the LBB analysis. The first analysis consideration is that the material properties of the cast austenitic stainless steel can change over time. Cast austenitic stainless steels used in the RCS are subject to thermal aging during service. This thermal aging causes an elevation in the yield strength of the material and a degradation of the fracture toughness, the degree of degradation being a function of the level of ferrite in the material. Thermal aging in these stainless steels will continue until a saturation or fully aged point is reached.

CEN-367-A used the fracture toughness values of the SA515 Grade 70 carbon steel weld in the LBB analysis, which are the lowest among all base and weld materials in the primary loop piping system. The staff compared the fracture toughness values in CEN-367-A with the more recent information in NUREG-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," and found that the CEN-367-A toughness data are more conservative than the NUREG-6177 lower-bound curve. Therefore, because the original analysis supporting LBB relied on fully aged stainless steel material properties, the analysis does not have a material property time dependency that requires further evaluation for license renewal.

The second analysis consideration is the accumulation of actual fatigue transient cycles over time that could invalidate the fatigue flaw growth analysis that was done as part of the original LBB analysis. A review of the accumulation of the applicable fatigue transient cycles is performed to meet the TLAA definition. This review was done within the scope of the FMP. The applicant stated that the continued implementation of the FMP provides reasonable assurance that thermal fatigue will be managed for the Class I, components such that they will continue to perform their intended function(s) for the period of extended operation.

4.6.1.2 Staff Evaluation

In the LRA regarding LBB, the applicant intended to demonstrate through qualitative assessment that the plant-specific FMP is capable of programmatically managing the assumptions, including the fatigue cycles, in the existing LBB analyses for the period of extended operation. The staff confirmed that the LBB applications for the primary loop piping were approved generically for Combustion Engineering Owners Group (CEOG) plants by the NRC on October 30, 1990, and specifically for St. Lucie Units 1 and 2, on March 5, 1993. The LBB analyses, which provided technical bases for these approved LBB applications, considered the thermal aging of the cast austenitic stainless steel material of the piping and assumed 40 years of operation. Since the primary loop piping contains cast stainless steel material, the LBB application is a TLAA for both plants.

The thermal aging of the cast stainless steel material has been identified as an issue to be reevaluated. The applicant's reevaluation revealed that the original LBB analyses had employed the thermal aging properties which are more conservative than the lower-bound curve documented in NUREG-6177, and therefore bounded the aging material data for St. Lucie. The staff performed a comparison of the material aging information in CEN-367-A with the information in NUREG-6177, and agreed with the applicant's conclusion that fully aged, lower bounding material property was used in the original LBB analyses. Hence, the properties for the cast stainless steel piping material are acceptable because they will not degrade below the fully aged properties in the period of extended operation.

For the remaining primary loop piping materials, instead of revising the original analyses by taking into account the fatigue transient cycles for the period of extended operation, the applicant relies on the plant-specific FMP to ensure that the accumulation of the applicable fatigue transient cycles over time will not invalidate the fatigue flaw growth analysis that was performed as part of the original LBB analyses. With this program in place, which calls for constant review of the accumulation of applicable fatigue transient cycles, the applicant concluded that the continued implementation of the Fatigue Monitoring Program will provide reasonable assurance that the Reactor Coolant Systems components within the scope of license renewal will continue to perform their intended functions consistent with the CLBs for the period of extended operation. The staff reviewed the FMP and determined that the program is adequate to monitor the applicable set of transients and their limits, and to count the actual thermal cycle transients to ensure that it is within the allowable limits of the defined transients. In the event that 80 percent of a design cycle limit assumed in the original LBB analyses is reached, the applicant will review the FMP and determine appropriate actions.

Based on the above evaluation, the staff agrees with the applicant's conclusion that the continued implementation of the FMP provides reasonable assurance that thermal fatigue will be managed for the primary loop piping and components, and that therefore the analyses for this TLAA remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

Since the V.C. Summer main coolant loop weld cracking event involving Alloy 182/82 weld material, the staff has considered the effect of primary water stress corrosion cracking (PWSCC) on Alloy 182/82 piping welds as an operating plant issue affecting all piping with or without approved LBB applications. To resolve this issue, the industry has taken the initiative to (1) develop overall inspection and evaluation guidance, (2) assess the current inspection technology, and (3) assess the current repair and mitigation technology. An interim industry report, "PWR Materials Reliability Project Interim Alloy 600 Safety Assessment for US PWR Plants (MRP-44), Part 1: Alloy 182/82 Pipe Butt Welds," was published in April 2001 to justify the continued operation of PWRs while the industry completes the development of the final report. The staff accepted this interim report in an SE dated June 14, 2001, with the following statement, "Should the industry not be timely in resolving inspection capabilities to identify PWSCC in Alloy 600 welds, regulatory action may result." The final industry report on this issue has not yet been published, and the staff is resolving it under 10 CFR Part 50, pending receipt of this final report and additional ultrasonic testing inspection data from piping involving Alloy 182/82 weld material from the industry.

4.6.1.3 UFSAR Supplement

The applicant's UFSAR supplement for LBB for RCS piping is provided in Section 18.3.5 of Appendices A1 and A2 for Units 1 and 2, respectively. The plant design cycles used in the applicant's LBB analysis are consistent with those utilized in the fatigue crack growth analysis and bound the period of extended operation. In addition, the applicant's appropriate consideration of thermal aging of the cast austenitic stainless steel material constitutes the basis for the staff acceptance of the licensee's evaluation of the LBB TLAA for the period of extended operation. On the basis of its review of the updated UFSAR supplements, the staff concludes that the summary description of the applicant's actions to address LBB for the period of extended operation is adequate.

4.6.1.4 Conclusions

The staff concludes that the applicant has provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1), that, for leak before-break TLAA, the analyses remain valid and the effects of aging on the pressure boundary function will be adequately managed for the period of extended operation. The staff also concludes that the USFAR supplements contain an appropriate summary description of the containment penetrations fatigue TLAA evaluation for the period of extended operation.

4.6.2 Crane Load Cycle Limit

4.6.2.1 Summary of Technical Information in Application

In Section 4.6.2 of the LRA, the applicant identified the crane load cycle limit as a TLAA for the cranes within the scope of license renewal. The cranes include the reactor building polar cranes, refueling machine and hoist (Unit 2 only), reactor containment building auxiliary telescoping jib cranes, fuel transfer machine (Unit 2 only), spent fuel handling machine (Unit 2 only), refueling canal bulkhead monorail (Unit 2 only), cask storage pool bulkhead monorail (Unit 2 only) and intake structure bridge cranes. The applicant stated that these cranes are designed in accordance with the criteria of the Crane Manufacturers Association of America (CMAA) Specification No. 70, "Specifications for Electric Overhead Traveling Cranes," and are acceptable for at least 20,000 to 200,000 load cycles. The applicant also stated that these cranes are used primarily during refueling outages. Occasionally, cranes make lifts at or near their rated capacity. However, most crane lifts are substantially less than their rated capacity. The St. Lucie Unit 2 spent fuel handling machine is bounding for the other cranes within the renewal scope.

The applicant states that the spent fuel handling machine is used primarily to move fuel assemblies during refueling cycles and is subject to the most loading cycles at or near its rated capacity. Considering a 3-batch fuel management scheme, which assumes one-third of the core is replaced at each refueling (every 18 months), and a full core off load every 10 years, the number of lifts performed in 60 years is projected to be less than 7100. Since the spent fuel handling machine load cycle analysis bounds the other cranes within the license renewal scope, all the cranes considered in this evaluation are adequate for expected load cycles over the period of extended operation. In addition, because crane gearing and shafting fatigue design per CMAA-70 are related to load lifts, the crane gearing and shafting are also adequate for the period of extended operation. Therefore, the applicant concluded that the crane analyses associated with crane design, including fatigue, remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.6.2.2 Staff Evaluation

The staff reviewed Section 4.6.2 of the LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1). On the basis of the staff's review of the information described above, the staff finds the applicant's analysis demonstrated that the actual usage of the cranes over the projected life through the period of extended operation will be far less than the analyzed load cycles per the design specification, and all the cranes within the LRA will continue to perform their intended function throughout the period of extended operation. Therefore, the applicant's TLAA analysis concerning the crane load cycle limit meets the requirements of 10 CFR 54.21(c)(1).

The applicant provides a summary description of the evaluation of the crane load cycle limit in Section 18.3.6 of Appendix A1 and Section 18.3.6 of Appendix A2, for Units 1 and 2, respectively. The applicant stated that the load cycles for these cranes were evaluated for the period of extended operation. On the basis of staff's review, the staff concludes that the applicant's description is sufficient to satisfy the requirements of 10 CFR 54.21(d).

4.6.2.3 Conclusions

The staff concludes that the applicant has provided an acceptable demonstration pursuant to 10 CFR 54.21(c)(1) that, for the crane load cycle limits TLAA, the analyses have been projected to the end of the period of extended operation. The staff also concludes that the UFSAR supplements contain an appropriate summary description of this TLAA evaluation for the period of extended operation.

4.6.3 Unit 1 Core Support Barrel Repair

4.6.3.1 Summary of Technical Information in the Application

In Section 4.6.3 of the LRA, the applicant states that during the 1983 St. Lucie Unit 1 refueling outage, the CSB and thermal shield assembly were observed to be damaged. The thermal shield was permanently removed. Four lugs were found to have separated from the CSB, and through-wall cracks were found adjacent to the lug areas. The CSB was repaired at the thermal shield support lug locations. Through-wall cracks were arrested with crack-arrestor holes and non-through-wall cracks were machined out. The lug tear-out areas were machined out and patched. The crack arrestor holes were sealed by inserting expandable plugs. The nuclear steam supply system supplier performed an analysis of the CSB repair method that demonstrated that the repair patches and expandable plug designs were acceptable for the remaining (40-year) life of the plant, consistent with ASME Code allowable stresses.

In 1984, a post-repair inspection of the CSB lug area repairs was performed to verify proper installation of the plugs and to provide a baseline for comparison of data from subsequent inspections. A visual and mechanical inspection was performed in 1986, after one cycle of operation. The inspection report concluded that no changes had occurred with respect to the baseline inspection. The applicant determined that the CSB was acceptable for long-term operation, and only visual inspections at 10-year intervals were necessary. A 10-year inservice visual inspection of the lug repair areas was performed during the 1996 refueling outage. On the basis of comparisons between the 1984 and 1986 inspection results, no abnormal changes were observed in the repaired lug areas.

The analyses and followup inspection reports for the repaired CSB and the expandable plugs were screened against the six TLAA criteria. The applicant determined that two specific elements of the repair qualify as TLAAs—(1) the fatigue analysis of the CSB middle cylinder and (2) the acceptance criteria for the CSB expandable plugs' preload based on irradiation-induced stress relaxation. In Section 4.3.1 of the LRA, the applicant states that the design cycles for 40-year operation bound the period of extended operation. The applicant evaluated the CSB analysis and determined that the analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The CSB repair plugs are of an expandable design that allows the plugs to be preloaded against the CSB wall. The preload is required to provide proper seating of the plugs and patches and to prevent movement of the plugs due to hydraulic drag loads.

The applicant stated that the original plug preload analysis was sufficient to accommodate normal operating hydraulic loads and thermal deflections for the original operating life of the plant. This preload analysis was revised for increased 60-year end-of-life fluence and for irradiation-induced relaxation input. The analysis concluded that all the repair plug flange deflection measurement readings are sufficient to meet the minimum required values and maintain the plugs preloaded. The applicant concluded that the CSB repair plugs will perform their intended function for the period of extended plant operation. The CSB plug preload relaxation analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

The applicant indicated in Subsection 4.3.1 of the LRA that the design cycles for 40-year operation bound the period of extended operation. The staff evaluated the CSB analysis and determined that it remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.6.3.2 Staff Evaluation

The staff reviewed the information provided by the applicant in the LRA and concluded that additional information was needed before the safety of the CSB for the period of extended operation could be evaluated. In RAI 4.6.3-1, the staff requested that the applicant provide a detailed description of the fatigue analysis of the CSB middle cylinder with the expandable plugs, and confirm that the fatigue evaluation meets the ASME Section III Class 1 limit fatigue criterion for the period of extended operation.

The applicant responded to RAI 4.6.3-1 in a letter dated October 10, 2002. In its response, the applicant states that the fatigue methodology developed for the CSB repairs employs a conservative method for combining component stresses to obtain stress intensities for the various cyclical loading conditions. The plant design transients and cycles utilized in the fatigue analysis are defined in Section 5.2.1.2 of the St. Lucie Unit 1 UFSAR. These design transients are also applicable to the RV internal components. The design limits for RV internals are specified in Section 4.2.2.1.2 of the St. Lucie Unit 1 UFSAR. For the core support structures, the allowable stress values are those given in the May 1972 drafts of ASME Section III, Subsection NG, and Appendix F, "Rules for the Evaluation of Faulted Conditions." In the fatigue evaluation of the CSB, the full 40-year design transient set was applied, without taking credit for cycles before the CSB damage in 1983. As stated in Subsection 4.3.1 of the LRA, the 40-year design cycles bound the period of extended operation. On this basis, the applicant calculated a CUF of 0.58 for the CSB middle cylinder. The staff finds the applicant's result acceptable because it does not exceed the ASME Section III Class 1 CUF limit of 1.0.

In RAI 4.6.3-2, the staff requested that the applicant provide the source and basis for the data and information that were used to assess irradiation-induced relaxation of the plug preload, which is expected to occur in the CSB expandable plugs at the end of 60 years of reactor operation. In RAI 4.6.3-3, the staff requested that the applicant provide a detailed description of the CSB plug preload analysis, which is based on irradiation-induced stress relaxation, showing that the expandable plugs will continue to perform their function given the predicted fluence, operating temperature, operating hydraulic loads, and thermal deflections for the period of extended operation.

The applicant responded to RAIs 4.6.3-2 and 4.6.3-3 in letters dated October 10, 2002, and November 27, 2002, respectively. In its responses, the applicant states that the preload acceptance criteria for the expandable plugs that were used in the repair of the St. Lucie Unit 1 CSB depend on irradiation-induced stress relaxation, a process in which the stress in the material under load decreases with time. The analysis of the time varying effect of stress relaxation on the preloading of the plugs thus constitutes a TLAA under the provisions of 10 CFR 54.3.

The CSB repair plugs were installed at the end of Cycle 5, as part of the overall St. Lucie Unit 1 CSB repair effort that included removing the thermal shield assembly and repairing damage incurred following a failure of the thermal shield support system. The CSB damage consisted of through-wall cracks and thermal shield support-lug non-through-wall tear-out areas. The through-wall cracks were arrested with circular crack arrestor holes, and the through-wall tear areas were machined out and sealed with patches. The function of the repair plugs is to seal the through-wall crack arrestor holes and the tear-out holes, and to limit or prevent bypass flow leakage through the holes.

The repair plugs are of an expandable design that allows the plugs to be preloaded against the CSB wall. This preload is required to provide proper sealing of the plugs and patches, to prevent movement of the plugs due to hydraulic drag loads, and to keep the plugs tight under anticipated thermal cycling conditions.

A plug consists of a thin-wall cylinder with a preformed flange. The plug is inserted and expanded in the hole, thus bending the flange and preloading the plug. The design of the plugs allows for the preload to be quantified by measuring the deflection of the plug flange, which acts against the outside diameter of the CSB. The preload criteria are defined as the minimum deflection requirements required to maintain the plug preload over the operating life of the plant. The criteria were determined based on the applied hydraulic drag forces, relative thermal expansion effects, and irradiation-induced stress relaxation of the flange/cylinder over the life of the plant.

As part of the 1997 St. Lucie Unit 1 steam generator replacement effort, the reactor coolant flow rate was increased, which increased the hydraulic drag forces on the plugs. In support of license renewal, the applicant revised the preload analysis to recalculate the preload criteria. The re-analysis utilized the original methodology, updated fluence and irradiation-induced stress relaxation material data input, and reduced temperature and temperature gradients in the CSB.

The applicant then evaluated previously measured deflections against the revised criteria. In accordance with the original evaluation of plug flange deflection measurements, actual measured plug flange deflection must be greater than or equal to the acceptance criteria. The applicant stated that the re-analysis results demonstrate that the plugs have sufficient preload to perform their intended function over the 60-year operating life of the plant. In all cases, actual plug flange deflection measurements exceed the revised acceptance criteria. The re-analysis concludes that the CSB repair plugs will maintain the preload and perform their intended function for the period of extended operation.

The applicant stated in previous reports that the plugs were designed to meet ASME Code Section III Class 1 requirements. The ASME Code, Section III, Subsection NB, has no provision for addressing thermal stress relaxation, since this effect becomes significant above temperatures for which ASME Code materials are specified (700–800 °F). Radiation induced stress relaxation does occur at normal operating temperatures experienced by the CSB, however, its effect is negligible except for highly stressed members such as the CSB plugs. Therefore, the ASME code has no provisions or design criteria for irradiation induced stress relaxation at these temperatures.

By letter dated October 10, 2002, the applicant provided the (proprietary) description of the methodology used in the preload analysis. The staff reviewed the methodology and the updated stress relaxation data on which the analysis is based. The staff determined that the assumptions used in the re-analysis are consistent with acceptable engineering principles, the calculation are consistent with the initial analysis, and that the measured plug deflections meet the acceptance criteria determined by the re-analysis. On the basis of its review, the staff concludes that the applicant provided a reasonable demonstration that the plugs will continue to perform their intended function during the period of extended operation.

4.6.3.3 Conclusions

The staff concludes that the applicant has provided an acceptable demonstration pursuant to 10 CFR 54.21(c)(1) that, for the Unit 1 CSB repair TLAA, analyses have been projected to the end of the period of extended operation and the effects of aging on the intended functions will be adequately managed for the period of extended operation. The staff also concludes that the Unit 1 FSAR supplement contains an appropriate summary description of this TLAA evaluation.

4.6.4 Alloy 600 Instrument Nozzle Repairs

4.6.4.1 Summary of Technical Information in the Application

In Section 4.6.4 of the LRA, the applicant summarizes the process and results of its TLAA related to half-nozzle repairs of leaking Alloy 600 instrumentation nozzles to the RCS hot leg piping or pressurizers. The staff reviewed this section to determine whether the applicant provided adequate information to meet the requirements of 10 CFR 54.21(c).

Small-diameter Alloy 600 nozzles, such as pressurizer and RCS hot leg instrumentation nozzles in Combustion Engineering-designed PWRs, have developed leaks or partial through-wall cracks as a result of PWSCC. In Section 4.6.4 of the LRA, the applicant indicates that Units 1 and 2 have experienced instances of leakage from RCS Alloy 600 instrument nozzles. The applicant states that it has used an alternative repair technique known as the "half-nozzle" weld repair as the method for repairing leaking RCS Alloy 600 instrument nozzles. The applicant indicates that four leaking pressurizer steam space instrument nozzles at Unit 2, and one leaking hot leg instrument nozzle at Unit 1, were repaired using half nozzle repair methods.

4.6.4.2 Staff Evaluation

In a half-nozzle repair technique, the leaking (cracked) Alloy 600 nozzle is cut above the partial-penetration J-groove weld that was used to join the nozzle to the RCS hot leg piping or pressurizer shell. The section of the nozzle that is proximal to the outer surface of the pressure boundary component is removed and replaced with a short Alloy 690 nozzle section. The inserted Alloy 690 nozzle section is then welded to the pressure boundary component's outside surface. The half-nozzle repair method leaves a short section of the original nozzle attached to the inside surface with the "J" weld, and exposes the ferritic (i.e., low-alloy steel or carbon steel) pressure boundary material to the borated water conditions of the reactor coolant.

In Section 4.6.4 of the LRA, the applicant indicates that a fracture mechanics analysis was submitted to the NRC to support the Unit 2 pressurizer steam space half-nozzle repairs performed in 1994. The fracture mechanics analysis justified the acceptability of indications in the "J" weld based on a postulated flaw size and flaw growth considering the applicable design cycles. Based on the results of the analysis, the applicant concluded that the postulated flaw size for the worst-case instrument nozzle was acceptable for the remaining design life of the plant (30 years, or 75 percent of the original 40-year plant design life).

The applicant also indicates that a half-nozzle repair was implemented on a Unit 1 RCS hot leg instrumentation nozzle in April 2001. In response to NRC questions regarding this repair, FPL documented that the indications in the "J" weld were bounded by the fracture mechanics analysis provided in CEOG Topical Report No. CE NPSD-1198-P, Revision 0, "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/ Replacement Programs," which was submitted on February 15, 2001, to the NRC for review and approval. The applicant also documented in that response that the CEOG topical report is applicable to the Unit 2 pressurizer steam space nozzle repairs performed in 1994.

In LRA Section 4.6.4 of the LRA, the applicant states that the CEOG report provides a bounding flaw evaluation that covers all small-diameter Alloy 600/690 nozzle repairs in accordance with ASME Section XI requirements. The flaw growth analysis included in the report assumes the total number of design cycles, consistent with the Units 1 and 2 UFSARs.

This generic analysis bounds the Class 1 fatigue design requirements of Units 1 and 2. In Section 4.3.1 of the LRA, the applicant states that its review of actual plant operation supports the conclusion that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation.

The CEOG submitted Topical Report No. CE NPSD-1198-P, Revision 0, to the NRC on February 15, 2001, to obtain generic approval of the Alloy 600/690 nozzle repair/replacement programs using mechanical nozzle seal assembly or half-nozzle repair methods. The staff provided its safety evacuation concerning Topical Report No. CE NPSD-1198-P, Revision 0, by letter to the CEOG dated February 8, 2002. The scope of the staff's safety evaluation concerning Topical Report No. CE NPSD-1198-P, Revision 0, covered both the initial 40-year operating lives and extended period of operation for existing nuclear power facilities. In the safety evaluation, the staff informed the CEOG that the topical report—

- provided an acceptable method for calculating the overall general/crevice corrosion rate for the internal surfaces of the low-alloy or carbon steel materials that will now be exposed to the reactor coolant, and for calculating the amount of time the ferritic portions of the vessel or piping would be acceptable in the event that corrosive wall thinning had occurred

- provided an acceptable method of calculating the thermal fatigue crack growth life of existing flaws in the Alloy 182/82 weld material into the ferritic portion of the piping or vessels

- provided acceptable bases and arguments for concluding that unacceptable growth of existing flaws into the ferritic portion of the piping or vessels by stress corrosion is implausible

However, in the safety evaluation, the staff also stated that there were certain limitations on how the report could be applied to plant-specific TLAA assessments on use of mechanical nozzle seal assemblies or half-nozzle repairs. The staff informed the CEOG that licensees seeking to use the methods of the report would need to perform certain plant-specific calculations given below in order to confirm that the ferritic portions of the piping or vessels within the scope of the report will be acceptable for service throughout the licensed lives of their plants. The licensed life would be either 40 years for an initial license or 60 years for a renewed license.

An overall general corrosion rate assessment for ferritic components that the half-nozzle or mechanical nozzle seal assembly will be joined to is based on the following steps:

- a calculation of the minimum acceptable wall thinning thickness for the ferritic vessel or piping that will adjoin to the mechanical nozzle seal assembly repair or half-nozzle replacement

- a calculation of the overall general corrosion rate for the ferritic materials based on the calculational methods in the report, the general corrosion rates listed in the report for normal operations, startup conditions (including hot standby and cold-shutdown conditions), and the respective design-basis capacity factors (in percentage of total plant life) for these operating conditions

- a calculation of the amount of general corrosion-based thinning for the vessels or piping over the life of the plant, as based on the overall general corrosion rate calculated in Step 2 and the thickness of the ferritic vessel or piping that will adjoin to the mechanical nozzle seal assembly repair or half-nozzle replacement

- a determination whether the vessel or piping is acceptable over the remaining life of the plant by comparing the worst-case remaining wall thickness to the minimum acceptable wall thickness for the vessel or pipe

A thermal fatigue crack growth assessment is based on the thermal fatigue crack growth methods of the report and CEOG Proprietary Evaluation A-GEN-PS-0003, Revision 00, for the bounding plant-specific mechanical nozzle seal assembly or half-nozzle designs implemented at their facilities consistent with plant-specific loading conditions and the methods of analysis in the Proprietary Evaluation. This assessment was to accomplish the following steps:

- a calculation of the maximum allowable crack length and crack depth for the worst-case crack projected to extend into the ferritic portions of the vessels or piping that will adjoin to the mechanical nozzle seal assembly repair or half-nozzle replacement

- thermal fatigue crack growth analysis of the worst-case flaw assumed to occur in the original Alloy 182/82 weld metal that is based on the calculational thermal fatigue crack growth methods in Proprietary Evaluation A-GEN-PS-0003, Revision 00, "Evaluation of Fatigue Crack Growth Associated with Small-Diameter Nozzles in CEOG Plants"

- a comparison of the maximum crack length and crack depth determined from the growth analysis to the maximum allowable crack length and crack depth to determine whether fatigue growth of the worst-case crack will be acceptable over the operating life for the facility (40 years if the normal licensing-basis plant life is used, or 60 years if the facility is expected to be approved for extension of the operating license)

- a review of the plant-specific RCS coolant chemistry histories to justify that growth of the existing flaw by stress corrosion was implausible, and to confirm that the applicant had monitored and controlled the amount of dissolved oxygen, halide ion, and sulfate ion impurity concentrations introduced into RCS coolant and had maintained the electrochemical potential for the coolant to a potential below the threshold potential for postulating growth of existing flaws by stress corrosion (i.e., to support that the electrochemical potential for the coolant was below -200 million electron volts (MeV)

Consistent with the staff's safety evaluation of February 28, 2002, any plant-specific thermal fatigue crack growth and general corrosion assessments performed by the applicant for the bounding half-nozzle repair implemented at the St. Lucie Nuclear Station will need to be consistent with the staff's criteria for performing these assessments, as stated in Sections 2.3.1 and 3.2 of the staff's safety evaluation, and will need to demonstrate that any existing flaw in the nozzle's original J-groove weld metal will be acceptable for service over the periods of extended operation for the units. In RAI 4.6.4-1, the staff requested the applicant to demonstrate that the half-nozzle designs would have acceptable structural integrity against unacceptable crack growth due to thermal fatigue and would be acceptable for service through the expiration of the extended operating licenses for Units 1 and 2.

The half-nozzle repair methods will leave the ferritic portions of the hot leg or pressurizer shells, to which the half-nozzles are welded, exposed to the borated reactor coolant. Under certain conditions, corrosion of the ferritic pressure boundary materials may occur. The CEOG submitted Topical Report No. CE NPSD-1198-P, Revision 0, on February 15, 2001, to address this issue. The applicant did not indicate that it had performed a plant-specific, overall general corrosion rate calculation for the ferritic pressure boundary components to which the half-nozzle repair designs have been welded. In RAI 4.6.4-2, the staff requested the applicant to demonstrate that the half-nozzle designs will have sufficient structural integrity against loss of material by corrosion and will meet their minimum wall thickness requirements through the expiration of the extended period of operation for the Units.

Consistent with the staff's safety evaluation of February 28, 2002, any plant-specific corrosion assessments performed by the applicant for the bounding half-nozzle repair implemented at St. Lucie Nuclear Station will need to be consistent with the staff's criteria for performing these assessments, as stated in Sections 2.3.2 and 3.3 of the staff's safety evaluation, and will need to demonstrate that growth of the existing flaw in the nozzle's original J-groove weld metal would not be plausible through the period of extended operation for the units. In these sections of the staff's safety evaluation, the staff informed applicants referencing the topical report that they could apply the report's stress corrosion cracking growth assessment if they could demonstrate that a sufficient level of hydrogen overpressure was being implemented at their facilities and that the contaminant levels for dissolved oxygen, halide ions, and sulfate ions in the RCS coolant were being maintained at concentrations below 10 Parts per billion (ppb), 150 ppb, and 150 ppb, respectively. Therefore, in RAI 4.6.4-3, the staff requested justification and validation of the CEOG's conclusion that growth of the existing flaw in the original Alloy 600 J-groove weld material by stress corrosion would not be a plausible effect during the period of extended operation for the Units.

The applicant submitted its responses to RAIs 4.6.4-1, 4.6.4-2, and 4.6.4-3, by letter dated October 10, 2002. In its responses, the applicant summarized the results of the CE's original fatigue crack growth analysis, boric acid wastage analysis, and stress corrosion-induced crack growth analysis as provided in CE Proprietary Topical Report CE NPSD-1198-P, Revision 00. Subsequent to the staff's review of CE Proprietary Topical Report CE NPSD-1198-P, Revision 00, Westinghouse Corporation revised the topical report to address potential issues with the original boric acid wastage analysis for the half nozzle designs. These potential issues were raised as a result of the boric acid wastage event of the Davis Besse reactor vessel (RV) head, and to address a design calculation error discovered by Westinghouse in the original fatigue crack growth analysis for the half nozzle designs. The revised report is provided in Class 2 Proprietary WCAP-15973-P, "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs (November 2002)," which was submitted to the NRC for review and approval in Combustion Engineering Owners Group letter CEOG-02-243, dated November 11, 2002. The report is applicable to the St. Lucie half-nozzle designs. To supplement its response to RAI 4.6.4-1, the applicant submitted Class 2 Proprietary Calculation CN-CI-02-60, "Evaluation of Fatigue Crack Growth Associated with Small Diamater Nozzles for St. Lucie 1 & 2," as the corresponding St. Lucie-specific fatigue crack growth analysis for the St. Lucie half nozzle designs. The staff is currently reviewing the acceptability of WCAP-15973-P and Class 2 Proprietary Calculation CN-CI-02-60.

In addition, by letter dated January 8, 2003, the applicant submitted a relief request for approval of the half nozzle designs implemented at the St. Lucie Nuclear Station. In this relief request, submitted pursuant to 10 CFR 50.55a(a)(3)(ii), the requested approval of an alternative to Paragraph IWB-123.3 of the 1989 Edition of Section XI to the ASME Boiler and Pressure Vessel Code, which requires that, for a components containing a flaw, that the "component or portion of the component containing the flaw be replaced." The staff is currently in the process of reviewing the acceptability of the applicant's relief request of January 8, 2003.

The acceptability of the TLAA for the St. Lucie half nozzle designs is pending approval of the WCAP-15973-P, Class 2 Proprietary Calculation CN-CI-02-60 and the applicant's relief request of January 8, 2003. This is open item 4.6.4-1.

4.6.4.3 UFSAR Supplement

Section 18.3.8 of Appendix A1 and Section 18.3.7 of Appendix A2 to the LRA provide the following UFSAR supplement summary descriptions for the TLAAs on the Alloy 600 instrument nozzle repairs:

> Small diameter Alloy 600 nozzles, such as pressurizer and RCS hot-leg instrumentation nozzles in Combustion Engineering designed PWRs, have developed leaks or partial through-wall cracks as a result of primary water stress corrosion cracking. The residual stresses imposed by the partial-penetration "J" welds between the nozzles and the low alloy or carbon steel pressure boundary components are the driving force for crack initiation and propagation.
>
> A repair technique known as the "half-nozzle" weld repair has been used to repair selected Alloy 600 instrument nozzles. In the half-nozzle technique, the Alloy 600 nozzle is cut outboard of the partial-penetration weld and replaced with a short Alloy 690 nozzle section that is welded to the outside surface of the pressure boundary component. This repair leaves a short section of the original nozzle attached to the inside surface with the "J" weld.
>
> A half-nozzle repair was implemented on a Unit 1 RCS hot-leg instrumentation nozzle in April 2001. In response to NRC questions regarding this repair, FPL documented that the indications in the "J" weld were bounded by the fracture mechanics analysis provided in Combustion Engineering Owner's Group (CEOG) Topical Report CE NPSD-1198-P.
>
> CEOG Topical Report CE NPSD-1198-P was submitted to the NRC February 15, 2001, to obtain generic approval of the Alloy 600/690 nozzle repair/replacement programs. The CEOG report provides a bounding flaw evaluation that covers all small diameter Alloy 600/690 nozzle repairs in accordance with ASME Section XI requirements. The flaw growth analysis included in the report assumes the total number of design cycles, consistent with the Unit 1 UFSAR. This generic analysis bounds the Class 1 fatigue design requirements of Unit 1. As discussed in Section 18.3.2.1, review of actual plant operation concludes that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation.

In order to ensure that the FSAR supplements summary descriptions for this TLAA are up to date, the applicant must supplement them to include the reference to Topical Report WCAP-15973-P; Class 2 Proprietary Calculation CN-CI-02-60; and the relief request for the St. Lucie half nozzle designs, dated January 8, 2003. This is set forth in open item 4.6.4-1.

4.6.4.4 Conclusions

The staff has review TLAA 4.6.4 regarding the expected acceptable lives of half-nozzle replacement techniques used to replace leaking Alloy 600 instrumentation nozzles in the RCS hot leg piping or pressurizers. With the exception of open item 4.6.4-1, the applicant's TLAA for the St. Lucie half nozzle designs for small bore Alloy 600 nozzles is acceptable.

APPENDIX A
CHRONOLOGY

This appendix contains a chronological listing of the routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and the Florida Power and Light Company (FPL), and other correspondence regarding the NRC staff's review of the St. Lucie Nuclear Plant, Units 1 and 2 (under Docket Numbers 50-335 and 50-389), for license renewal application (LRA).

November 29, 2001	In a letter (signed by J. Stall), FPL submitted its LRA for St. Lucie Nuclear Plant, Units 1 and 2. ML013400473
November 29, 2001	In a letter (signed by D. Jernigan), FPL submitted license renewal boundary drawings. ML013480240
December 19, 2001	In a letter (signed by C. Grimes), the NRC notified FPL concerning the receipt and availability of the LRA. ML013400473
December 20, 2001	In a letter (signed by D. Jernigan), FPL submitted additional copies of the LRA. ML020160029
January 8, 2002	In a letter (signed by D. Jernigan), FPL submitted a revised page for the LRA. ML020110489
January 24, 2002	In a letter (signed by C. Grimes), the NRC notified FPL of the acceptability and sufficiency for docketing, proposed review schedule, and opportunity for a hearing regarding the LRA. ML020240333.
February 18, 2002	In a letter (signed by D. Jernigan), FPL submitted additional copies of the "Application for Renewed Operating Licenses for St. Lucie Nuclear Plant, Units 1 and 2." ML020520515
February 22, 2002	In a letter (signed by C. Grimes), the NRC informed FPL of its intent to prepare an environmental statement and to conduct scoping. ML020530588
April 15, 2002	In a letter (signed by N. Dudley), the NRC notified FPL of a revision to the schedule for the conduct of the review of the LRA. ML021050186
May 7, 2002	In a letter (signed by M. Masnik), the NRC provided FPL a summary of the scoping meeting held in support of the environmental review (RAIs) of the LRA. ML021300604
May 7, 2002	In a letter (signed by M. Masnik), NRC provided FPL requests for additional information related to the staff's review of severe accident mitigation alternatives. ML021340363
May 28, 2002	In a letter (signed by D. Jernigan), FPL submitted responses to the

	staff's RAIs related to its review of severe accident mitigation alternatives. ML021820106
May 28, 2002	In a notice (signed by S. Koenick), the NRC announced a public meeting with FPL regarding the staff's review of the LRA. ML021500580
June 3, 2002	In a letter (signed by P.T. Kuo), the NRC requested confirmation of the U.S. Department of Commerce position regarding Federally protected species that may be affected by the operation of St. Lucie Units 1 and 2. ML021570345
June 19, 2002	In a letter (signed by N. Dudley), the NRC provided a summary of the May 28 and 29, 2002, teleconferencing calls with FPL regarding potential RAIs concerning its review of the LRA. ML021780091
June 21, 2002	In a letter (signed by J. Cushing), the NRC provided a summary of the May 15-16, 2002, meeting with FPL regarding potential RAIs concerning its review of the LRA. ML021780147
June 25, 2002	In a letter (signed by D. Jernigan), FPL provided a response to the NRC concerning RAIs related to the staff's review of severe accident mitigation alternatives associated with the LRA. ML021820106
July 1, 2002	In a letter (signed by N. Dudley), the NRC provided FPL RAIs regarding its review of Sections 2.0, 3.0, 4.0, and Appendix B of the LRA. ML021830288
July 1, 2002	In a letter (signed by N. Dudley), the NRC provided FPL RAIs regarding its review of Section 3.3 of the LRA. ML021830321
July 8, 2002	In a letter (signed by M. Masnik), the NRC provided FPL the environmental scoping summary report associated with its review of the LRA. ML021920466
July 18, 2002	In a letter (signed by N. Dudley), the NRC provided FPL RAIs regarding its review of Sections 2.0, 3.0, 4.0, and Appendix B of the LRA. ML022030456
July 24, 2002	In a letter (signed by P.T. Kuo), the NRC informed the U.S. Fish and Wildlife Service of its biological assessment of 14 Federally protected species in the vicinity of the St. Lucie Nuclear Plant. ML022060314
July 29, 2002	In a letter (signed by N. Dudley), the NRC provided FPL RAIs regarding its review of Sections 2.2, 2.3, and Appendix B of the LRA. ML022110165
July 30, 2002	In a letter (signed by J. Powers) the U.S. Department of Commerce

	provided clarification to the NRC regarding the effect of the cooling water intake system on local wildlife. ML022200253
July 31, 2002	In a meeting summary (signed by N. Dudley), the NRC summarized the June 10 - 11, 2002, meeting concerning draft RAIs. ML022130182
August 26, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC with a supplemental response to RAIs associated with the environmental report of the LRA. ML022410053
September 26, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC responses to RAIs concerning the scoping and screening methodology in Section 2.1 of the LRA. ML022700567
September 26, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC responses to RAIs concerning the aging management review results in Section 3.0 of the LRA. ML022740116
September 26, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC responses to RAIs concerning aging management review results – auxiliary systems in Section 3.3 of the LRA. ML022740106
September 26, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC responses to RAIs concerning the aging management programs in Appendix B of the LRA. ML022740199
September 27, 2002	In a meeting summary (signed by N. Dudley), the NRC summarized the August 15-16 and September 4-5, 2002, meetings concerning the applicant's draft responses to RAIs. ML022700262
	The FPL draft responses discussed during the meetings were e-mailed to the NRC. The six e-mails contained responses to RAIs concerning the following LRA sections.

Scoping and Screening Methodology received 7/19/02	ML022700426
Scoping and Screening Results received 8/6/02	ML022700434
Aging Management Reviews (AMRs) received 8/6/02	ML022700446
Auxiliary Systems AMRs received 8/6/02	ML022700453
Time-Limited Aging Analyses received 8/26/02	ML022700472
Aging Management Programs received 8/26/02	ML022700477

October 2, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC responses to RAIs concerning the scoping and screening results in Section 2.0 of the LRA. ML022810608
October 7, 2002	In a letter (signed by C. Casto), the NRC announced a public meeting on October 25, 2002, concerning the results of the NRC's first

	inspection of the license renewal program. ML022800527
October 10, 2002	In a letter (signed by N. Dudley), the NRC provided FPL a revised schedule for the conduct of its review of the LRA. ML022900065
October 10, 2002	In a letter (signed by D. Jernigan), FPL provided the NRC responses to RAIs concerning the time-limited aging analyses in Section 4.0 of the LRA. ML022890457
October 19, 2002	In a memorandum (signed by N. Dudley), the NRC provided FPL a summary of an October 17, 2002, telephone call concerning responses to RAIs pertaining to the LRA. ML022940378
November 19, 2002	In a letter (signed by N. Dudley), the NRC provided FPL an exemption from the requirements regarding the schedule for submitting amendments to the LRA. ML023240285
November 27, 2002	In a memorandum (signed by N. Dudley), the NRC provided FPL a summary of meetings on November 6-7, 2002, and phone calls on November 20, 21, and 25, 2002, concerning FPL's draft supplemental responses to RAIs. ML023330412
November 27, 2002	In a letter (signed by D. Jernigan, FPL provided the NRC supplemental responses to RAIs pertaining to the LRA. ML023380251
November 27, 2002	In a letter (signed by D. Jernigan, FPL provided the NRC supplemental responses to RAIs pertaining to the LRA. ML023600436

APPENDIX B
REFERENCES

This appendix contains a listing of references used in the preparation of the Safety Evaluation Report prepared during the review of the license renewal application for St. Lucie Units 1 and 2 under Docket Numbers 50-335 and 389.

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ACI 349.3R, *Evaluation of Existing Nuclear Safety-Related Concrete Structures.*

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ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWC, *Requirements for Class 2 Components of Light-Water Cooled Power Plants*.

ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants.*

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EPRI TR-103834-P1-2, *Effects of Moisture on the Life of Power Plant Cables, Part 1: Medium-Voltage Cables, Part 2: Low-Voltage Cables, prepared by Ogden Environmental and Energy Services Company, Final Report,* August 1994.

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APPENDIX C
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D. Ma	Task Leader
V. Shaw	Structures
S. Tam	Structures

Appendix D: Commitments Listing

During the review of FPL's LRA by the NRC staff, the applicant made commitments to provide aging management programs to manage aging effects on structures and components prior to the expiration of its current operating license terms. The following tables list these commitments along with their implementation schedule for each unit.

Table 1 - License Renewal Commitment Listing for St. Lucie Unit 1

Item	Commitment	UFSAR Supplement Location (LRA Appendix A1)	Implementation Schedule	Source
1	Perform a visual inspection to determine the extent of loss of material due to pitting and microbiologically induced corrosion on the external surfaces of the buried pipe that connects the St. Lucie Units 1 and 2 Condensate Storage Tanks.	18.1.1, Condensate Storage Tank Cross-Connect Buried Piping Inspection	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.1
2	Perform inspections on the surfaces of piping and components to determine if galvanic corrosion is active in systems where it is not expected.	18.1.2, Galvanic Corrosion Susceptibility Inspection Program	Prior to the end of the initial operating license term, additional inspections based on results.	LRA Appendix B, Subsection 3.1.2 Response to RAI B.3.1.2-1 (FPL letter L-2002-222)
3	Perform examinations using volumetric techniques of the internal surfaces of stainless steel Auxiliary Feedwater piping downstream of the recirculation orifices.	18.1.3, Pipe Wall Thinning Inspection Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.3 Responses to RAIs B.3.1.3-1 and B.3.1.3-2 (FPL letter L-2002-166)

Item	Commitment	UFSAR Supplement Location (LRA Appendix A1)	Implementation Schedule	Source
4	Submit a report summarizing the aging effects applicable to reactor vessel internals including a description of the inspection plan.	18.1.4, Reactor Vessel Internals Inspection Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.4 Response to RAI 3.1-1 (FPL letter L-2002-157)
5	Perform a one-time inspection of the reactor vessel internals	18.1.4, Reactor Vessel Internals Inspection Program	During the period of extended operation.	LRA Appendix B, Subsection 3.1.4
6	Submit a report summarizing the inspection plan for small bore Class 1 piping prior to implementation.	18.1.5, Small Bore Class 1 Piping Inspection	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.5 Response to RAI B.3.1.5-1 (FPL letter L-2002-166)
7	Perform volumetric inspections of a sample of small bore Class 1 piping.	18.1.5, Small Bore Class 1 Piping Inspection	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.5 Response to RAI B.3.1.5-1 (FPL letter L-2002-166)
8	Implement the Thermal Aging Embrittlement of CASS Program.	18.1.6, Thermal Aging Embrittlement of CASS Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.6
9	Perform inspections and examinations of the reactor vessel head, incorporate NRC requirements, FPL responses to NRC IE Bulletins, and industry recommendations, including the EPRI Materials Reliability Project.	18.2.1, Alloy 600 Inspection Program	On-going	LRA Appendix B, Subsection 3.2.1 Response to RAI B.3.2.1-1 (FPL letter L-2002-166)

Item	Commitment	UFSAR Supplement Location (LRA Appendix A1)	Implementation Schedule	Source
10	Enhance the ASME Section XI Subsection IWB, IWC, IWD Inservice Inspection Program to: Perform VT-1 inspections of the core stabilizing lugs and core support lugs, and Evaluate pressurizer surge line flaws (if identified) with regard to environmentally assisted fatigue.	18.2.2.1, ASME Section XI Subsection IWB, IWC, IWD Inservice Inspection Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.2.1
11	Revise the Boraflex Surveillance Program to include areal density testing (in lieu of blackness testing) of the encapsulated Boraflex material in the spent fuel storage racks.	18.2.3, Boraflex Surveillance Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.3
12	Expand the scope of the Boric Acid Wastage Surveillance Program to include Waste Management components in the scope of license renewal.	18.2.4, Boric Acid Wastage Surveillance Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.4
13	Revise procedures to provide guidance in the event that fatigue design cycle limits are approached.	18.2.7, Fatigue Monitoring Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.7
14	Incorporate NFPA-25 testing of wet pipe sprinklers into the Fire Protection Program.	18.2.8, Fire Protection Program	Prior to 50 years from initial operating license.	Response to RAI B.3.2.8-6 (FPL letter L-2002-222)
15	Expand the scope of the Flow Accelerated Corrosion Program to include internal and external loss of material of drain lines and selected steam traps.	18.2.9, Flow Accelerated Corrosion Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.9
16	Enhance the Periodic Surveillance and Preventive Maintenance Program to include components such as filter housings, radiator fins, flexible hoses, door seals, and expansion joints.	18.2.11, Periodic Surveillance and Preventive Maintenance Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.11

Item	Commitment	UFSAR Supplement Location (LRA Appendix A1)	Implementation Schedule	Source
17	Program documentation will be enhanced to integrate all aspects of the four subprograms that makeup the Reactor Vessel Integrity Program.	18.2.12, Reactor Vessel Integrity Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.12
18	Enhance the Systems and Structures Monitoring Program to include: Monitoring of the interior surfaces of below groundwater concrete, and examination of a representative sample of below groundwater concrete, when excavated for any reason, Aging management of inaccessible concrete, inspection of insulated equipment and piping, and evaluating masonry wall degradation and uniform corrosion, and Aging management of accessible reinforced concrete and reinforced masonry block walls.	18.2.14, Systems and Structures Monitoring Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.14 Responses to RAIs 3.5-9 and 3.5-10 (FPL letter L-2002-157 Response to RAI B.3.2.14-2 (FPL letter L-2002-166) Response to RAI 3.5-12 (FPL Letter L-2002-241)
19	Establish an aging management program to address non-EQ cables and connections in the Containment. The non-EQ cables and connections managed by this program will include those associated with sensitive, low-level signal circuits (source, intermediate, and power range neutron detectors). Complete the first inspection described in the aging management program.	New section to be added	Prior to the end of the initial operating license term.	Responses to RAIs 3.6-1 and 3.6-2 (FPL letter L-2002-222)

Item	Commitment	UFSAR Supplement Location (LRA Appendix A1)	Implementation Schedule	Source
20	Address environmentally assisted fatigue of the pressurizer surge line using one or more of the following approaches: Further refinement of the fatigue analysis to lower the CUF(s) to below 1.0, or Repair of the affected locations, or Replacement of the affected locations, or Manage the effects of fatigue by an NRC approved inspection program.	18.3.2.3, Environmentally Assisted Fatigue	During the period of extended operation	LRA Subsection 4.3.3 Response to RAI 4.3-3 (FPL letter L-2002-222)

Table 2 - License Renewal Commitment Listing for St. Lucie Unit 2

Item	Commitment	UFSAR Supplement Location (LRA Appendix A2)	Frequency	Source
1	Perform inspections on the surfaces of piping and components to determine if galvanic corrosion is active in systems where it is not expected.	18.1.1, Galvanic Corrosion Susceptibility Inspection Program	Prior to the end of the initial operating license term, additional inspections based on results.	LRA Appendix B, Subsection 3.1.2 Response to RAI B.3.1.2-1 (FPL letter L-2002-222)
2	Perform examinations using volumetric techniques of the internal surfaces of stainless steel Auxiliary Feedwater piping downstream of the recirculation orifices and carbon steel Component Cooling Water piping associated with the control room air conditioning.	18.1.2, Pipe Wall Thinning Inspection Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.3 Response to RAIs B.3.1.3-1 and B.3.1.3-2 (FPL letter L-2002-166)
3	Submit a report summarizing the aging effects applicable to reactor vessel internals including a description of the inspection plan.	18.1.3, Reactor Vessel Internals Inspection Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.4 Response to RAI 3.1-1 (FPL letter L-2002-157)
4	Perform a one-time inspection of the reactor vessel internals	18.1.3, Reactor Vessel Internals Inspection Program	During the period of extended operation.	LRA Appendix B, Subsection 3.1.4
5	Submit a report summarizing the inspection plan for small bore Class 1 piping prior to implementation.	18.1.4, Small Bore Class 1 Piping Inspection	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.5 Response to RAI B.3.5-1 (FPL letter L-2002-166)

Item	Commitment	UFSAR Supplement Location (LRA Appendix A2)	Frequency	Source
6	Perform volumetric inspections of a sample of small bore Class 1 piping.	18.1.4, Small Bore Class 1 Piping Inspection	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.5 Response to RAI B.3.5-1 (FPL letter L-2002-166)
7	Implement the Thermal Aging Embrittlement of CASS Program.	18.1.5, Thermal Aging Embrittlement of CASS Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.1.6
8	Perform inspections and examinations of the reactor vessel head, incorporate NRC requirements, FPL responses to NRC IE Bulletins, and industry recommendations including EPRI Materials Reliability Project.	18.2.1, Alloy 600 Inspection Program	On-going	LRA Appendix B, Subsection 3.2.1 Response to RAI B.3.2.1-1 (FPL letter L-2002-166)

Item	Commitment	UFSAR Supplement Location (LRA Appendix A2)	Frequency	Source
9	Enhance the ASME Section XI Subsection IWB, IWC, IWD Inservice Inspection Program to: Perform VT-1 inspections of the core stabilizing lugs and core support lugs, and Evaluate pressurizer surge line flaws (if identified) with regard to environmentally assisted fatigue.	18.2.2.1, ASME Section XI Subsection IWB, IWC, IWD Inservice Inspection Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.2.1
10	Expand the scope of the Boric Acid Wastage Surveillance Program to include Waste Management components in the scope of license renewal.	18.2.3, Boric Acid Wastage Surveillance Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.4
11	Revise procedures to provide guidance in the event that fatigue design cycle limits are approached.	18.2.6, Fatigue Monitoring Program	Prior to the end of the initial operating license term.	LRA Appendix B, Subsection 3.2.7
12	Incorporate NFPA-25 testing of wet pipe sprinklers into the Fire Protection Program.	18.2.7, Fire Protection Program	Prior to 50 years from initial operating license.	Response to RAI B.3.2.8-6 (FPL letter L-2002-222)
13	Expand the scope of the Flow Accelerated Corrosion Program to include internal and external loss of material of selected steam traps.	18.2.8, Flow Accelerated Corrosion Program	Prior to the end of the initial operating license term.	LRA Appendix B Subsection 3.2.9
14	Enhance the Periodic Surveillance and Preventive Maintenance Program to include components such as filter housings, radiator fins, flexible hoses, door seals, and expansion joints.	18.2.10, Periodic Surveillance and Preventive Maintenance Program	Prior to the end of the initial operating license term.	LRA Appendix B Subsection 3.2.11

Item	Commitment	UFSAR Supplement Location (LRA Appendix A2)	Frequency	Source
15	Program documentation will be enhanced to integrate all aspects of the four subprograms that makeup Reactor Vessel Integrity Program.	18.2.11, Reactor Vessel Integrity Program	Prior to the end of the initial operating license term.	LRA Appendix B Subsection 3.2.12
16	Enhance the Systems and Structures Monitoring Program to include: Monitoring of the interior surfaces of below groundwater concrete, and examination of a representative sample of below groundwater concrete, when excavated for any reason, Aging management of inaccessible concrete, inspection of insulated equipment and piping, and evaluating masonry wall degradation and uniform corrosion, and Aging management of accessible reinforced concrete and reinforced masonry block walls.	18.2.14, Systems and Structures Monitoring Program	Prior to the end of the initial operating license term.	LRA Appendix B Subsection 3.2.14 Response to RAIs 3.5-9 and 3.5-10 (FPL letter L-2002-157 Response to RAI B.3.2.14-2 (FPL letter L-2002-166) Response to RAI 3.5-12 (FPL Letter L-2002-241)
17	Establish an aging management program to address non-EQ cables and connections in the Containment. The non-EQ cables and connections managed by this program will include those associated with sensitive, low-level signal circuits (source, intermediate, and power range neutron detectors). Complete the first inspection described in the aging management program.	New section to be added	Prior to the end of the initial operating license term.	Response to RAIs 3.6-1 and 3.6-2 (FPL letter L-2002-222)

Item	Commitment	UFSAR Supplement Location (LRA Appendix A2)	Frequency	Source
18	Address environmentally assisted fatigue of the pressurizer surge line using one or more of the following approaches: Further refinement of the fatigue analysis to lower the CUF(s) to below 1.0, or Repair of the affected locations, or Replacement of the affected locations, or Manage the effects of fatigue by an NRC approved inspection program.	18.3.2.3, Environmentally Assisted Fatigue	During the period of extended operation	LRA Subsection 4.3.3 Response to RAI 4.3-3 (FPL letter L-2002-222)