

4. TIME-LIMITED AGING ANALYSIS

4.1 Identification of Time-Limited Aging Analyses

In the license renewal application (LRA), Section 4.1, the applicant identified the time-limited aging analyses (TLAAs) applicable to Turkey Point Units 3 and 4. The NRC staff reviewed the information in the LRA to determine whether the applicant provided adequate information to meet the requirements stated in 10 CFR 54.21(c)(1).

4.1.1 Summary of Technical Information in the Application

In the LRA, Table 4.1-1, the applicant identified the calculations and evaluations that meet all six criteria of 10 CFR 54.3 for a TLAA. The applicant identified the following as TLAA categories:

- Reactor Vessel Irradiation Embrittlement
- Metal Fatigue
- Environmental Qualification
- Containment Tendon Loss of Prestress
- Containment Liner Plate Fatigue
- Other Plant-Specific Time-Limited Aging Analyses

Each of these categories contain specific TLAAs that are discussed in Sections 4.2 through 4.7 of the LRA.

4.1.2 Staff Evaluation

In the LRA, Section 4.1, the applicant described the requirements for identifying and evaluating TLAAs and plant-specific exemptions based on TLAAs. The applicant reviewed the Turkey Point UFSAR, Technical Specifications, docketed licensing correspondence, and applicable Westinghouse WCAPs. The information provided by the applicant was reviewed by the NRC staff to determine which analyses and calculations met the six criteria defining TLAAs in 10 CFR 54.21(c)(1).

4.1.3 Conclusions

The NRC staff concludes that the applicant has provided a list of acceptable 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 Irradiation Embrittlement

The TLAAs for evaluating the effects of neutron irradiation on the ability of the reactor vessel to resist failure during a pressurized thermal shock (PTS) event, the maintenance of acceptable Charpy upper-shelf energy (USE) levels, and the analysis of P-T limits for 32 and 48 effective full-power years (EFPYs) are discussed in Section 4.2 of the LRA.

4.2.1 Summary of Technical Information in the Application

The applicant described its reactor vessel irradiation embrittlement TLAA's in Section 4.2 of the LRA. The TLAA's evaluated in Section 4.2 of the LRA include analyses and calculations performed to show compliance with 10 CFR Part 50.60, Appendix G to 10 CFR Part 50 regarding P-T limits and acceptable Charpy USE values and 10 CFR 50.61 regarding protection against PTS events. The TLAA's are reviewed by the staff in the following paragraphs.

4.2.2 Staff Evaluation

In Section 4.2 of the LRA, the applicant stated that the group of TLAA's in this section relate to the effect of irradiation embrittlement on the beltline regions of the Turkey Point Units 3 and 4 reactor vessels. The calculations discussed in this section use predictions of the cumulative effects of irradiation embrittlement on the reactor vessels. The staff has reviewed the reactor vessel integrity program in Section 3.9.13 of this SER and finds it acceptable for the period of extended operation. The three aspects of reactor vessel embrittlement are reactor vessel resistance to failure during PTS events, the maintenance of acceptable Charpy USE levels, and analysis of P-T limits. The maximum anticipated effects of PTS, USE, and P-T limits would be in the reactor vessel beltline region at the end of the period of extended operation. A discussion of the three TLAA's is provided below.

Pressurized Thermal Shock

Rules for protecting against PTS in pressurized water reactors are given in 10 CFR 50.61(b)(1). Licensees are required to perform an assessment of the reactor vessel material's projected values of PTS reference temperature, RT_{PTS} , through the end of their operating license. Upon approval of its application for an extended period of operation for Turkey Point Units 3 and 4, this period would be 48 EFPYs.

Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," describes two methods for determining RT_{PTS} for reactor vessel materials. Position 1 is for material that does not have surveillance data available, and Position 2 is for material that has surveillance data. These provisions are also incorporated in 10 CFR 50.61.

Availability of surveillance data is not the only measure of whether Position 2 may be used. The data must also meet the credibility criteria given in the PTS rule (10 CFR 50.61).

According to the terminology in 10 CFR 50.61, RT_{PTS} is the sum of the initial (unirradiated) reference temperature, $RT_{NDT(u)}$, the shift in reference temperature caused by neutron irradiation (ΔRT_{NDT}), and a margin term (M) to account for uncertainties.

$RT_{NDT(u)}$ is determined using the method of Section III of the ASME Boiler and Pressure Vessel Code. That is, $RT_{NDT(u)}$ is the greater of the drop weight nil-ductility transition temperature or the temperature that is 60 °F below that at which the material exhibits Charpy test values of 50 ft-lb and 35 mils lateral expansion. For a material for which test data are unavailable, generic values may be used if there are sufficient test results for that class of material.

For Position 1 materials (surveillance data not available), ΔRT_{NDT} is defined as the product of the chemistry factor and the fluence factor. The chemistry factor is a function of the material's copper and nickel content expressed as weight %. Although not explicitly discussed by the applicant, the "best estimate" copper and nickel contents will normally be the mean of the measured values for a plate or forging. For a weld, the best estimate values will normally be the mean of the measured values from weld deposits made using the same weld wire heat number as the limiting weld. For Turkey Point Units 3 and 4, best estimate values were obtained from BAW-2325, "Reactor Vessel Working Group, Response to RAI Regarding Reactor Pressure Vessel Integrity." The value of the chemistry factor is directly obtained from tables in 10 CFR 50.61. The fluence factor is calculated using end-of-license peak fluence at the clad-to-base-metal interface for the material's location. Fluence values were obtained by extrapolation to 48 EFPYs from 32 EFPY values.

For Position 2 materials (surveillance data available), the discussion above for Position 1 applies except for determination of the chemistry factor, which in this instance is a material-specific value calculated as follows:

- multiply each ΔRT_{NDT} value by its corresponding fluence factor
- sum these products
- divide this sum by the sum of the squares of the fluence factors

The applicant did not discuss the ratio procedure in 10 CFR 50.61. If surveillance data are being used and there is clear evidence that the copper and nickel content of the surveillance weld differ from the vessel weld (i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld), the measured values of ΔRT_{NDT} must be adjusted for differences in copper and nickel by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld. The applicant did not apply the ratio procedure (Position 2.1) to the calculations for the Turkey Point Units 3 and 4 limiting circumferential weld (weld wire heat number 71249) but opted to obtain the chemistry factor directly from the tables in 10 CFR 50.61 (Position 1). By letter dated October 30, 2000, the staff concluded that the chemistry factor ratio adjustment described in Position 2.1 should be performed on the data for analysis purposes, but the resulting chemistry factor should be calculated using the 10 CFR 50.61 tables in accordance with Position 1. More information on the staff's findings is provided in the P-T limits discussion at the end of this section.

The margin term (M) is generally determined as follows:

$$M = 2 (\sigma_l^2 + \sigma_{\Delta}^2)^{0.5}$$

where σ_l is the standard deviation for $RT_{NDT(u)}$

and σ_{Δ} is the standard deviation for ΔRT_{NDT}

For determining M , $\sigma_l = 0$ if a measured value is used. If a generic value is used, σ_l is the standard deviation of the set of values used to obtain the mean value. For ΔRT_{NDT} , $\sigma_\Delta = 28^\circ\text{F}$ for welds and 17°F for base metal (plate and forging), except that σ_Δ need not exceed one-half of the mean value of ΔRT_{NDT} . Note that when Position 2 is applied as the method for calculating the chemistry factor using credible surveillance data, the same method for determining the σ values is used except that σ_Δ values may be halved (14°F for welds and 8.5°F for base metal).

In accordance with 10 CFR 50.61(b)(2), the screening criteria for RT_{PTS} is 270°F for plates, forgings, and axial welds, and 300°F for circumferential welds. The values of RT_{PTS} at 48 EFPYs for Turkey Point Units 3 and 4 are listed in Section 4.2.1 of the LRA. The inputs for the calculation and the resulting RT_{PTS} values are displayed in the table below.

In RAI 4.2.1-1 (letter dated February 1, 2001), the staff requested additional information on the PTS evaluation for the limiting materials in the Turkey Point Units 3 and 4 reactor vessel beltline. The applicant provided the requested information by letter dated April 19, 2001. The limiting material for Turkey Point Units 3 and 4 at the end of the license renewal period (48 EFPYs) is projected to be circumferential weld SA-1101 (weld wire heat number 71249). As mentioned previously, the RT_{PTS} value was calculated using Position 1 in 10 CFR 50.61.

The 48 EFPY fluence projections for the SA-1101 circumferential welds are $4.12 \times 10^{19} \text{ n/cm}^2$ and $4.07 \times 10^{19} \text{ n/cm}^2$ for Turkey Point Units 3 and 4, respectively. For conservatism, the applicant used a value of $4.5 \times 10^{19} \text{ n/cm}^2$ in the PTS analysis. The best estimate chemistry content values are 0.23% copper and 0.59% nickel for both units.

The inputs for the RT_{PTS} calculation are provided below:

Unit	Circumferential Weld Material (weld heat number)	Inner Surface Fluence x 10^{19} n/cm^2	Initial RT_{NDT} $^\circ\text{F}$	Margin $^\circ\text{F}$	Chemistry Factor (CF)	Inside Surface fluence factor (ff)	ff x CF	RT_{PTS} $^\circ\text{F}$
Units 3 & 4	SA1101 (71249)	4.5	10	56	167.55	1.38	231.4	297.4

The limiting projected RT_{PTS} value for Turkey Point Units 3 and 4 is projected to be below the screening criterion at the end of the license renewal period. It has a projected RT_{PTS} value at 48 EFPYs of 297.4°F (the screening criterion is 300°F for circumferential welds). Therefore, the staff finds that, with respect to PTS events, the Turkey Point Units 3 and 4 reactor vessels have sufficient margin to perform their intended functions over the period of extended operation.

The analysis associated with PTS 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).

Charpy Upper-Shelf Energy

Although not discussed by the applicant, Appendix G to 10 CFR Part 50 requires that reactor vessel beltline materials have Charpy USE levels in the transverse direction for the base metal and along the weld for the weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially, and must maintain Charpy USE levels throughout the life of the vessel of no less than 50 ft-lb (68 J). However, Charpy USE levels below these criteria may be acceptable if it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that the lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. In Section 4.2.2, of the LRA the applicant notes that 10 CFR Part 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the USE of any of the reactor vessel material is predicted to drop below 50 ft-lb, as measured by Charpy V-notch specimen testing.

RG 1.99, Rev. 2, provides two positions for determining Charpy upper-shelf energy (C_vUSE). Position 1 is for material that does not have surveillance data available and Position 2 is for material that has surveillance data. For Position 1, the %-drop in C_vUSE for a stated copper hyphenate %-drop content and neutron fluence is determined by reference to Figure 2 of RG 1.99, Rev. 2. This %-drop is then applied to the initial C_vUSE to obtain the adjusted C_vUSE . For Position 2, the %-drop in C_vUSE is determined by plotting the available surveillance data on Figure 2 of RG 1.99, Rev. 2, and fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points. Again, the percent drop is determined and used to adjust the initial C_vUSE value.

The Turkey Point circumferential weld material previously fell below the 10 CFR Part 50, Appendix G, requirement of 50 ft-lb. At that time, a fracture mechanics evaluation was performed to demonstrate acceptable equivalent margins of safety against fracture. The NRC reviewed and approved these evaluations, as documented in letters dated October 19, 1993, and May 9, 1994. These evaluations approved plant operation through the current license term (32 EFPYs).

On April 23, 2001, the staff received BAW-2312, Rev. 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of Turkey Point Units 3 and 4 for Extended Life Through 48 Effective Full Power Years." The staff required additional information to support the review of the topical report, which was referenced in the applicant's response to RAI 4.2.2-1 submitted by letter dated April 19, 2001. During a conference call on May 7, 2001, the applicant provided additional information regarding the details of the low upper shelf toughness fracture mechanics analysis. The applicant docketed this information by letter dated May 29, 2001. The applicant performed the fracture mechanics analysis in order to evaluate the SA-1101 circumferential reactor vessel welds at Turkey Point Units 3 and 4. The analysis was performed for ASME Levels A, B, C, and D service loadings based on the acceptance criteria and evaluation procedures of ASME Section XI, Appendix K, 1995 Edition with addenda through 1996. A detailed description of the methodology is provided in Section 4 of BAW-2312, Rev. 1.

With regard to transient selection, the original low upper-shelf analysis performed for B&W-designed reactor vessels (BAW-2178PA) was approved by the NRC staff by letter dated March 29, 1994. In that analysis, the licensee reviewed Level C and D transients for all participating plants, and concluded that the Turkey Point steam line break without offsite power transient was the most limiting of all Levels C and D transients, including loss-of-coolant accident transients. The new analysis in BAW-2312, Rev. 1, shows that this transient remains limiting for the period of extended operation.

The staff required additional information on the origin of K_{Iclad} , the stress intensity factor associated with the cladding. In its response, the applicant stated that the original low upper-shelf analysis considered a bounding vessel (Zion Unit 1) and the bounding transient discussed above (Turkey Point steam line break). Of all the B&W-designed reactor vessels considered in the analysis, the Zion vessel had the highest projected fluence and was as thick or thicker than any other vessel. The Turkey Point reactor vessel is 7.75 inches thick and the Zion Unit 1 reactor vessel is 8.44 inches thick. The nominal cladding thickness is 3/16 inches for both vessels. From a thermal stress perspective, it is conservative to consider the thicker vessel. It is therefore appropriate to utilize the bounding Zion value of 9 ksi \sqrt{in} as the stress intensity factor for K_{Iclad} in the Turkey Point low upper-shelf analysis reported in BAW-2312.

The applicant's evaluation concluded that the limiting weld for the Turkey Point Units 3 and 4 reactor vessels satisfies the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix K, for ductile flaw extension and tensile instability. The analysis associated with USE 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 staff concludes that the Turkey Point RPVs will have continued acceptable equivalent margins of safety against fracture through 48 EFYs.

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 RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The critical locations in the RPV 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 RCS is also limited by the net positive suction curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. 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 pressure-temperature limits and the net positive suction curves.

To address the period of extended operation, the 48 EFPY projected fluences and the Turkey Point specific reactor vessel material properties were used to determine the limiting material and calculate P-T limits for heatup and cooldown. The limiting material at all temperatures for the period of extended operation is the circumferential girth weld.

By letter dated July 7, 2000, the applicant submitted a proposed license amendment for Turkey Point Units 3 and 4 to extend the service period for the P-T limit curves to a maximum of 32 EFPYs, the end of the current license period. The proposed license amendment also included P-T limit curves and low-temperature overpressure protection (LTOP) setpoints for 48 EFPYs, the end of the period of extended operation. Florida Power and Light has not requested NRC approval of the 48 EFPY P-T limit curves and LTOP setpoints at this time. A separate license amendment specifically requesting approval of the 48 EFPY pressure-temperature limit curves and LTOP setpoints will be submitted to the NRC in the future and prior to expiration of the proposed 32 EFPY P-T limit curves.

The following description of the P-T limits evaluation applies to the 32 EFPY curves. However, when the applicant submits the 48 EFPY curves for review and approval, the treatment of the surveillance data should be consistent with the staff's method as outlined in the safety evaluation report dated October 30, 2000. For the limiting RPV material, circumferential weld heat number 71249, the applicant evaluated the four available surveillance data points for the heat. Because the surveillance weld materials had a higher copper content than the RPV welds, the applicant's analysis of the surveillance data did not incorporate a chemistry factor ratio adjustment as outlined in Position 2.1 of RG 1.99, Rev. 2. The applicant's evaluation indicated that the surveillance data did not satisfy the credibility criteria of RG 1.99, Rev. 2. Therefore, the chemistry factor for weld wire heat number 71249 was determined from Table 1 of RG 1.99, Rev. 2, and the full-margin term was used, in accordance with Position 1.1 of the regulatory guide.

In its evaluation of the surveillance data for circumferential weld heat number 71249, the staff determined, as did the licensee, that the surveillance data do not meet the credibility criteria of RG 1.99, Rev. 2. However, the staff notes that for an evaluation of the data to be consistent with the guidance of RG 1.99, Rev. 2, the chemistry factor ratio adjustment described in Position 2.1 should be performed on the data. This adjustment is necessary to ensure an accurate assessment of the data. Using the weld surveillance data with the chemistry factor ratio adjustment, the staff calculated a surveillance chemistry factor in accordance with Position 2.1 of RG 1.99, Rev. 2. The value was lower than the value determined by the applicant and lower than the chemistry factor calculated using Position 1.1 of RG 1.99, Rev. 2. As described previously, the staff confirms the licensee's finding that the surveillance data are not credible in accordance with RG 1.99, Rev. 2, and therefore the chemistry factor for the RPV weld should be calculated in accordance with Position 1.1 of RG 1.99, Rev. 2.

In a related matter, the staff notes that the NRC reactor vessel integrity database (RVID) information for the Turkey Point RPVs lists the circumferential weld (heat number 72442) between the nozzle belt and the intermediate shell as exhibiting a relatively high RT_{PTS} at end of life (EOL), although the neutron fluence ($\sim 0.3 \times 10^{19} \text{ n/cm}^2$) is an order of magnitude less than that for the materials considered by the applicant. Although this material is not the limiting material for Turkey Point Units 3 and 4, future additions to surveillance data or changes to embrittlement correlations could result in this material becoming a more significant consideration in determining the limiting material, and therefore this material should be tracked and considered by the licensee in future submittals.

As mentioned, the applicant should consider the methodology described in this section when submitting the 48 EFPY P-T limits curves for review and approval.

4.2.3 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description for the RCS TLAAAs described in the LRA, Appendix A, are acceptable. The applicant has met the requirements of 10 CFR 54.21(d). However, as discussed above in Section 4.2.2 of this SER, the applicant must apply the chemistry factor ratio adjustment described in RG 1.99, Rev. 2, Position 2.1, to the surveillance data when submitting the 48 EFPY P-T limits curves for review and approval. This adjustment is necessary to ensure an accurate assessment of the data. The staff confirms the licensee's finding that the surveillance data are not credible in accordance with RG 1.99, Rev. 2, and therefore the chemistry factor for the RPV weld should be calculated in accordance with Position 1.1 of RG 1.99, Rev. 2.

In addition, the circumferential weld (heat number 72442) between the nozzle belt and the intermediate shell exhibits a relatively high RT_{PTS} at EOL, and therefore this material should be tracked and considered by the licensee in future submittals.

By letter dated December 17, 2001, the applicant revised the updated FSAR supplement to reflect that the ratio procedure adjustment will be applied when the 48 EFPY P-T limits curves are submitted for NRC approval. The applicant also stated that it would track the circumferential weld fabricated from heat number 72442 due to its relatively high RT_{pts} at EOL. The staff finds this response to the confirmatory item 3.0-1 FSAR item 4.2-1 acceptable.

4.2.4 Conclusion

The staff has reviewed the TLAAAs regarding the ability of the reactor vessel to resist failure during a PTS event, the maintenance of acceptable Charpy USE levels, and the analysis of P-T limits for 32 and 48 EFPYs. It should be noted that the applicant submitted 48 EFPY P-T limits curves for information with a proposed license amendment for 32 EFPY curves (dated July 7, 2000). The applicant will submit a separate license amendment for approval of the 48 EFPY curves prior to the expiration of 32 EFPY curves. On the basis of the applicant's response to the confirmatory item described above, the staff concludes that the applicant's PTS, Charpy USE and P-T limits analyses satisfy the requirements of 10 CFR 54.21(c)(1)(ii).

4.3 Metal Fatigue

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity when metal fatigue initiates and propagates 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 the Turkey Point facility and, consequently, fatigue is part of the current licensing basis for these components. The applicant discussed the TLAA evaluations performed to address thermal and mechanical fatigue analyses of plant mechanical components in Section 4.3 of the LRA.

4.3.1 Summary of Technical Information in the Application

The applicant discusses the criteria used for the design of reactor coolant loop components in Section 4.3.1 of the LRA. The applicant indicates that the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. Fatigue analyses were performed for the critical locations in these components using conservative assumptions regarding the anticipated plant operational cycles. The applicant indicated that a review of the Turkey Point Units 3 and 4 operating history indicates that the number of operational cycles assumed in the design of these components bounds the number of cycles 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 Turkey Point Fatigue Monitoring Program (FMP) as a confirmatory program that assures the number of design cycle limits are not exceeded during the period of extended operation. The FMP is described in Appendix B of the LRA.

The applicant discussed the evaluation of reactor vessel underclad cracking in Section 4.3.2 of the LRA. Grain boundary separation perpendicular to the direction of the cladding weld overlay was identified in the heat-affected zone of the reactor vessel base metal in 1971. The acceptability of this condition was demonstrated by a generic fracture mechanics evaluation for the 40-year plant life. The applicant indicated that this evaluation has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

The applicant discussed the evaluation of the reactor coolant pump flywheel in Section 4.3.3 of the LRA. The flywheel has been evaluated for potential fatigue crack initiation in the keyway. The applicant indicated that the analysis was determined to remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant described the criteria used for the reactor coolant loop piping and balance-of-plant piping in Section 4.3.4 of the LRA. This piping, except for the pressurizer surge lines and the Unit 4 emergency diesel generator safety-related piping, was designed to the requirements of the ANSI B31.1, "Power Piping." The pressurizer surge lines were designed to the Class 1

requirements of the ASME Code. These lines are covered in the applicant's fatigue assessment discussed in Section 4.3.1 of the LRA. The Unit 4 emergency diesel generator safety-related piping was designed to the Class 3 requirements of the ASME Code, which are equivalent to the ANSI B31.1 requirements.

ANSI B31.1 requires a reduction factor be applied to the allowable bending stress range if the number of full-range thermal cycles exceeds 7,000. The applicant stated that its review of plant operating practices indicates that the number of thermal cycles assumed in the analysis will not be exceeded during the period of extended operation. 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).

The applicant described actions taken to address the issue of environmentally assisted fatigue in Section 4.3.5 of the LRA. The applicant described its evaluation of the following fatigue sensitive component locations:

- Reactor vessel shell and lower head
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line (including the pressurizer and hot leg nozzles)
- Reactor coolant system (RCS) piping charging nozzle
- RCS piping safety injection nozzle
- Residual heat removal system Class 1 piping

4.3.2 Staff Evaluation

As discussed in the previous section, components of the RCS were designed to the Class 1 requirements of the ASME Code. The Class 1 requirements contain explicit criteria for the fatigue analysis of components. Consequently, the applicant identified the fatigue analyses of these RCS components as TLAAs. The staff reviewed the applicant's evaluation of the RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The specific design criterion for Class 1 components involves calculating the cumulative usage factor (CUF). The fatigue damage caused by each thermal or pressure transient depends on the magnitude of the stresses caused in the component by the transient. The CUF sums the fatigue resulting from each transient. The design criterion requires that the CUF not exceed 1.0. The applicant stated that a review of the plant operating history indicates that the postulated 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.

Table 4.1-8 of the Turkey Point UFSAR contains a list of transient design conditions and associated design cycles used to evaluate RCS components. In RAI 4.3.1-1, dated February 2, 2001, the staff requested that the applicant provide the following information:

- The current number of operating cycles and a description of the method used to determine the number and severity of the design transients during the units' 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.

The applicant provided the information in its April, 19, 2001, response to RAI 4.3.1-1. The applicant obtained the current number of operating cycles from its FMP, which has been ongoing since initial plant startup. The applicant based its estimate of the number of cycles of design transients for 60 years of plant operation on the mean frequency of occurrence through June 1988 for most of the design transients. The applicant indicated that the mean frequency method is too conservative for plant heatup and cooldown transients and for reactor trip transients because of the large number of these transients in the early years of operation.

The applicant gave more weight to recent operating history of the plant to estimate the number of cycles of these transients for 60 years of plant operation. The staff considers the method described by the applicant to estimate the number of transient cycles for 60-years of plant operation reasonable. The applicant's FMP will continue to track the number of these cycles during the period of extended operation. The staff review of the FMP is contained in Section 3.9.7 of this SER.

Flaws in ASME Class 1 components that exceed the size of allowable flaws defined in IWB-3500 of Section XI of the ASME Code need not be repaired if they are analytically evaluated to the criteria in IWB-3600 of the ASME Code. The analytic evaluation requires the licensee to project the amount of flaw growth due to fatigue and stress corrosion mechanisms, or both, where applicable, during a specified evaluation period. In RAI 4.3.1-2, dated February 2, 2001, the staff requested that the applicant identify all Class 1 components that have flaws exceeding the allowable flaw limits defined in IWB-3500 and have been analytically evaluated to IWB-3600 of the ASME Code. The staff also requested that the applicant provide the results of the analyses that indicate whether the flaws will satisfy the criteria in IWB-3600 for the period of extended operation. In an April 19, 2001, response to RAI 4.3.1-2, the applicant indicated that there are no currently identified flaws in Class 1 components that exceed the allowable flaw limits defined in IWB-3500. Therefore, there are no TLAAs associated with flaw evaluations, and the RAI item 4.3.1-2 is therefore resolved.

NRC Bulletin (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. 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. In RAI 4.3.1-3, dated February 2, 2001, the staff requested that the applicant describe the actions taken to address these bulletins during the period of extended operation. In an April 19, 2001, response to RAI 4.3.1-3, the applicant indicated that no calculations that meet the definition of a TLAA were performed in response to NRC BL 88-08. The applicant further indicated that fatigue analyses of the Turkey Point Units 3 and 4 surge lines were performed in response to NRC Bulletin 88-11 and these were evaluated as TLAAs for the period of extended operation. The applicant's TLAA of the surge lines is discussed later in the staff evaluation. This RAI item is therefore closed.

The Westinghouse Owners Group has issued generic topical report WCAP-14574 to address aging management of pressurizers. In Sections 2.3.1.4 and Section 3.2.3 of the LRA, the applicant stated that WCAP-14574 was not incorporated by reference in the LRA. However, in Section 2.3.1.4 of the LRA, the applicant stated that the component intended functions for the Turkey Point pressurizers are consistent with the intended functions identified in WCAP-14574. In Section 3.2.3 of the LRA, the applicant further stated that the Turkey Point pressurizers are bounded by the description contained in WCAP-14574 with regard to design criteria and features, modes of operation, intended functions, and exposure to specific environments. Table 2-10 of WCAP-14574 indicates that the ASME Section III Class 1 fatigue CUF criterion could be exceeded at several pressurizer subcomponent locations during the period of extended operation. WCAP-14574 also identified recent unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses. In RAI 4.3.1-4, dated February 2, 2001, the staff requested that the applicant provide the following information:

- (1) Show the ASME Section III Class 1 CLB CUFs for the applicable subcomponents of Turkey Point Units 3 and 4 pressurizers specified in Table 2-10 of WCAP-14574, including consideration of environmental effects on the fatigue curves and the corresponding CUFs for the extended period of operation.
- (2) WCAP-14574, Section 3.8.3, lists other off-normal and additional transients. WCAP-14574, Section 3.8.4, described recently discovered surge line inflow/outflow thermal transients. These thermal cyclic transients were not considered in the CLB fatigue analyses of Westinghouse pressurizers, including Turkey Point Units 3 and 4. Provide the highest CUFs, considering these transients for the following pressurizer subcomponents for the extended period of operation:
 - (a) Surge nozzle
 - (b) Lower head region
 - (c) Heater wells
 - (d) Support skirt and flange
- (3) Describe the aging management programs that will be used to manage fatigue of the Turkey Point Units 3 and 4 pressurizer subcomponents for the extended period of operation, considering the transients listed above and environmental effects on fatigue.

In an April 29, 2001, response to RAI 4.3.1-4, the applicant provided a table of the CLB CUFs for the pressurizer subcomponents. All CUFs were shown to meet the ASME Section III Class 1 CUF criterion, without environmental considerations. The applicant stated that the CUFs for critical locations on the pressurizer were originally determined using CLB design transients and frequencies that were intended to be conservative and bounding for all foreseeable plant operational conditions. As discussed previously, the applicant compared the frequencies of the actual plant transients, obtained from its FMP, with the frequencies of the design transients. Based on this comparison, the applicant concluded that the CLB design transients and their frequencies were conservative and bounding for the period of extended operation, and therefore, the CLB CUFs for the Turkey Point Units 3 and 4 pressurizers were also conservative and bounding for the period of extended operation. The applicant further

indicated that it will monitor the CLB design transients using the Turkey Point FMP to assure that the number of design transients used in the evaluation of the pressurizer is not exceeded in the period of extended operations. The applicant stated that the CLB CUFs for the surge nozzle, the lower head region, the heater wells, and the support skirt and flange given in the response to this RAI include consideration of the off-normal and additional transients discussed in Section 3.8.3 of WCAP-14754, as applicable, including specific consideration of insurge/outsurge transients described in Section 3.8.4 of the report.

The CLB CUFs did not include consideration of environmental effects on the fatigue curves. The applicant indicated that the effects of environmentally assisted fatigue on pressurizer components are addressed through three approaches: (1) screening, (2) plant-specific evaluation, or (3) aging management.

The applicant stated that, based on evaluations reported in EPRI Report TR-107515, a conservative estimate of the environmental effect on the CUF for stainless steel is a factor of four. Therefore, the applicant evaluated the effects of the environment on the fatigue usage factor for components with a CUF > 0.25. These components included the surge nozzle, the spray nozzle, the lower head and heater well, and the upper head and shell. As indicated in the applicant's response, the environmental effect on the CUF for stainless steel components can be greater than a factor of 4. Even though the environmental effect could be greater than a factor of 4, the staff considers the applicant's screening criteria an acceptable method to obtain a sample of high fatigue usage pressurizer components for further evaluation.

The applicant's plant-specific evaluation consisted of a combination of quantitative evaluations and qualitative discussions of the conservatism in the fatigue analyses of the spray nozzle, the lower head and heater well, and the upper head and shell. The applicant used these evaluations to argue that the plant-specific CUFs would not exceed the screening criteria of CUF > 0.25 if conservative assumptions were removed from the analysis.

In its assessment of the pressurizer spray nozzle, the applicant used the number of cycles of inadvertent auxiliary spray operation projected and the number of cycles of normal spray operation during plant loading and unloading that are projected for 60 years of plant operation. The applicant also relied on a qualitative discussion of margins in the analysis. In its response, the applicant committed to either (1) modify the Turkey Point FMP to limit transient accumulations to the values used in the spray nozzle evaluation, (2) perform a more refined evaluation for the spray nozzle to show an acceptable CUF for 60 years, or (3) track CUF values in addition to cycle counts to ensure that CUF values remain acceptable. However, in accordance with 10 CFR 54.21(d), this information needs to be added to the FSAR supplement. This was part of confirmatory item 3.0-1 FSAR item 4.3-1. By letter dated December 17, 2001, the applicant provided this information in Section 16.3.2.5 of the updated FSAR supplement. The staff finds this response to the confirmatory item acceptable.

The applicant's evaluation of the lower shell consisted primarily of qualitative assessment of margins in the analysis with a reliance on the Section XI visual inservice inspections. The applicant's evaluation of the upper head and shell relied on the results of a 1989 Westinghouse study to argue that the pressurizer spray transient does not impinge directly on the upper shell as assumed in the fatigue analysis. The applicant indicated that the fatigue usage is negligible with direct impingement.

The applicant stated that the pressurizer surge nozzle is considered part of the pressurizer surge line. The applicant has committed to monitoring the surge line during the period of extended operation as discussed later in this section. The staff considers the surge line a bounding example to represent the effects of the environment on the fatigue life of pressurizer components during the period of extended operation.

The staff considers the applicant's evaluations a satisfactory method of identifying the most limiting pressurizer component, the surge line nozzle, for monitoring during the period of extended operation. If monitoring of the surge line nozzle identifies the need for additional actions for the period of extended operation, then the applicant should reassess the fatigue evaluation of the pressurizer components as part of its corrective action program. This reassessment should quantify the conservatism in the analyses as discussed above. RAI 4.3.1-4 is therefore closed.

The applicant indicated, based on its review of the Turkey Point operating history, that the ASME Code fatigue analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i). The applicant further indicated that the Turkey Point FMP will be continued in the period of extended operation. The applicant's FMP tracks transients and cycles of RCS components that have explicit design transient cycles to assure that these components stay within their design basis. Generic Safety Issue (GSI) 166, "Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of the RCS components. Although GSI-166 was resolved for the current 40-year design life of operating components, the staff identified GSI-190, "Fatigue Evaluation of Metal Components for 60-year Plant Life," to address license renewal. The NRC closed GSI-190 in December 1999, concluding:

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

The applicant evaluated the component locations listed in NUREG/CR-6260 that are applicable to an older vintage Westinghouse plant for the effect of the environment on the fatigue life of the components. The applicant indicated that the results reported in NUREG/CR-6260 were used to scale up the Turkey Point plant-specific usage factor, for the same locations to account for environmental effects. The applicant also indicated that the later environmental fatigue correlations in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," were considered in the evaluation. In RAI 4.3.5-1, dated February 2, 2001, 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 staff also requested that the applicant discuss how the factors used to scale up the Turkey Point plant-specific usage factors were derived. The staff requested that the applicant also discuss how the later environmental data provided in NUREG/CR-6583 and NUREG/CR-5704 were factored in the evaluations.

In an April 19, 2001, response to the RAI, the applicant indicated that the older vintage Westinghouse plant evaluated in NUREG/CR-6260 matched Turkey Point in terms of design codes and analytical techniques. The staff agrees that the NUREG/CR-6260 component evaluations of the older vintage Westinghouse plant are applicable to Turkey Point. The applicant compared the design-basis usage factors calculated for Turkey Point to the corresponding values reported in NUREG/CR-6260. This comparison is summarized in Table 1 of the response. The applicant indicated that the Turkey Point fatigue usage factors were different from the NUREG/CR-6260 usage factors because the Turkey Point usage factors accounted for the results of the power uprate evaluation performed in 1995. The power uprate had not been considered in NUREG/CR-6260. The final column of Table 1 contains the NUREG/CR-6260 usage factors with environmental fatigue effects factored into the assessment. The applicant described how the NUREG/CR-6260 usage factors that consider environmental effects are scaled to obtain Turkey Point plant-specific usage factors that account for environmental effects.

The applicant assessed the impact of the later data provided in NUREG/CR-6583 for carbon and low alloy steels on the usage factors calculated in NUREG/CR-6260. The applicant concluded that use of the later data would not have a significant impact on the calculated usage factors. The staff agrees with this conclusion. However, the applicant's plant-specific usage factors for the vessel and vessel nozzle are higher than those reported in NUREG/CR-6260 because of the power uprate. The applicant demonstrated acceptable plant-specific usage factors at these locations, accounting for environmental effects, by considering the number of transient cycles expected to occur during the 60 years of plant operation. In its response, the applicant committed to either (1) modify the Turkey Point FMP to limit transient accumulations to those used in the above evaluations, (2) perform a more refined evaluation for the RPV outlet nozzle and RPV shell at the core support pads to show acceptable CUF values for 60 years, or (3) track CUF values in addition to cycle counts to ensure CUF values remain acceptable. The staff considers these options acceptable methods of demonstrating that environmental fatigue effects will be adequately managed during the period of extended operation. However, in accordance with 10 CFR 54.21(d), this information needs to be added to the FSAR supplement. This was part of confirmatory item 3.0-1 FSAR item 4.3-1. By letter dated December 17, 2001, the applicant provided this information in Section 16.3.2.5 of the updated FSAR supplement. The staff finds this response to the confirmatory item acceptable.

The applicant assessed the impact of the data provided in NUREG/CR-5704 stainless steels on the usage factors calculated in NUREG/CR-6260. The applicant used the results of the analyses presented in NUREG/CR-6260 for the charging nozzle, safety injection nozzle, and residual heat removal system tee to represent the Turkey Point components. The applicant used these analyses because the design code for the Turkey Point piping did not require explicit fatigue analyses. The staff agrees that the fatigue analyses presented in NUREG/CR-6260 are representative of the Turkey Point components. The applicant multiplied the usage factors presented in NUREG/CR-6260 (based on revised interim stainless steel curves for 40 years) by a factor of two to account for the later fatigue data for stainless steel components provided in NUREG/CR-5704. The applicant concluded that the usage factors would remain below 1.0 if the number of cycles assumed in the design for 40 years were not exceeded during the period of extended operation. The staff agrees with the applicant's assessment that multiplying the usage factors in NUREG/CR-6260 by a factor of two bounds the impact of data provided in NUREG/CR-5704 for the charging nozzle, safety injection nozzle, and residual heat removal tee. Therefore, monitoring the number of design transients to assure that the number assumed in the design is not exceeded during the period of extended operation adequately addresses these components, and resolves the issue in RAI 4.3.5-1.

The applicant indicated that the pressurizer surge line required further evaluation for environmental fatigue during the period of extended operation. The applicant further indicated that it would use an aging management program to address fatigue of the surge line during the period of extended operation. The aging management program would rely on ASME Section XI inspections to address surge line fatigue during the period of extended operation. As indicated in the draft safety evaluation on Westinghouse Owners Group generic technical report WCAP -14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," the NRC has not endorsed a procedure on a generic basis which allows for ASME Section XI inspections in lieu of meeting the fatigue usage criteria. In RAI 4.3.5-2, dated February 2, 2001, the staff requested that the applicant provide a detailed technical evaluation which demonstrates the proposed inspections provide an adequate technical basis for detecting fatigue cracking before such cracking leads to through-wall cracking or pipe failure. The detailed technical evaluation should be sufficiently conservative to address all uncertainties associated with the technical evaluation (e.g., fatigue crack initiation and detection, fatigue crack size, and fatigue crack growth rate considering environmental factors). As an alternative to the detailed technical evaluation, the staff requested that the applicant provide a commitment to monitor the fatigue usage, including environmental effects, during the period of extended operation, and to take corrective actions, as approved by the staff, if the usage is projected to exceed 1.0.

In an April 19, 2001, response to RAI 4.3.5-2, the applicant discussed the results of its ultrasonic inspections of the surge line welds. The applicant stated that no reportable indications were identified. The applicant further stated that it plans to inspect all surge line welds prior to the period of extended operation. In addition to these inspections, the applicant has committed to address the concern of environmentally assisted fatigue using one or more of the following approaches:

- further refinement of the fatigue analysis to lower the CUFs to below 1.0, or
- repair of the affected locations, or

- replacement of the affected locations, or
- management of the effects of fatigue by an inspection program that has been reviewed and approved by the NRC.

The applicant commits to provide the NRC with the inspection details of the aging management program (AMP) requiring staff approval prior to the period of extended operation if the last option is selected. As indicated by the applicant, the use of an AMP to manage fatigue will require prior staff review and approval. The staff finds that the applicant's proposed program is an acceptable plant-specific approach to address environmentally assisted fatigue during the period of extended operation in accordance with 10 CFR 54.21(c)(1) and adequately addresses the issue in RAI 4.3.5.2. However, in accordance with 10 CFR 54.21(d), this information needs to be added to the FSAR supplement. This was part of confirmatory item 3.0-1 FSAR item 4.3-1. By letter dated December 17, 2001, the applicant provided this information in Section 16.3.2.5 of the updated FSAR supplement. The staff finds this response to the confirmatory item acceptable.

Components of the reactor coolant loop piping and balance-of-plant piping were designed to the requirements of the ANSI B31.1 power piping code, with two exceptions. The pressurizer surge lines were designed to the Class 1 requirements of the ASME Code. The staff evaluation of the surge lines is discussed above. The Unit 4 emergency diesel generator safety-related piping was designed to the Class 3 requirements of the ASME Code, which are equivalent to the ANSI B31.1 requirements. Both ANSI B31.1 and ASME Class 3 require a reduction in the range of allowable bending stresses caused by thermal loads if the number of full-range cycles exceeds 7,000. The applicant indicates that to obtain 7,000 full range cycles in 60 years a piping system would have to be cycled approximately once every 3 days. The applicant indicated that the piping systems subject to license renewal are only occasionally subjected to cyclic operation. Therefore, the applicant concluded that the analysis associated with B31.1 piping remains valid for the period of extended operation in accordance with Section 54.21(c)(1)(i). The staff agrees with the applicant's conclusion.

Turkey Point has two 3-loop RPVs. The method and materials used in the fabrication of the RPVs resulted in underclad cracks in the RPV forgings. In accordance with 10 CFR 54.21(c), the applicant must perform a time-limited aging analysis to determine the impact of 60 years of operation on the underclad cracks.

The applicant indicates that a generic evaluation of underclad cracks had been extended to 60 years using fracture mechanics evaluations based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service. In RAI 4.3.2-1, dated February 2, 2001, the staff requested that the applicant either reference a previous staff review of the generic analysis or provide the analysis for staff review. The staff also requested that the applicant compare the transients in the 60-year generic evaluation to the Turkey Point design transients and explain why the crack growth projected in the 60-year generic evaluation will bound the crack growth projected for Turkey Point in 60 years of operation.

By letter dated March 1, 2001, the WOG submitted for staff review topical report WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants (MUHP-6110)." This report describes the fracture mechanics analysis that evaluates the impact of 60 years of operation on reactor vessel underclad crack growth and reactor vessel integrity. In an RAI dated April 12, 2001, the staff identified areas where additional information was needed to complete its review of WCAP-15338. The WOG responded to the staff's RAI in letters dated June 15, 2001 and July 31, 2001. The staff review of this topical report is contained in a letter to Roger A. Newton, dated October 15, 2001. The staff concluded that upon completion of the renewal applicant action items, the WCAP-15338 report provides an acceptable evaluation of a TLAA for the RPV components with underclad cracks for Westinghouse Owners Group (WOG) plants. The staff's safety evaluation identifies two license renewal applicant action items to be addressed in the plant-specific license renewal application when incorporating the WCAP-15338 report in a renewal application.

Renewal Application Item (1):

The license renewal applicant is to verify that its plant is bounded by the WCAP-15338 report. Specifically, the renewal applicant with a 3-loop RPV is to indicate whether the number of design cycles and transients assumed in the WCAP-15338 analysis bounds the number of cycles for 60 years of operation of its RPV. The renewal applicant with a 2-loop or 4-loop RPV needs to demonstrate that the transients for normal, upset, emergency, faulted, and PTS conditions used in WCAP-15338 report bound their plant-specific transients for these conditions. Otherwise, they need to perform similar Section XI flaw evaluations using their plant-specific transients to demonstrate that their RPVs with underclad cracks are acceptable for 60 years of operation.

In an April 19, 2001, response to RAI 4.3.2-1, the applicant indicated that the number of design cycles and transients assumed in the WCAP-15338, analysis bounds the Turkey Point Units 3 and 4 design transients identified in UFSAR Table 4.1-8 and provided in Appendix A of the LRA. Therefore, the conclusions in the WCAP are applicable to Turkey Point reactor vessels.

Renewal Application Item (2):

10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAA for the period of extended operation. Those applicants for license renewal referencing the WCAP-15338 report for the RPV components shall ensure that the evaluation of the TLAA is summarily described in the FSAR supplement.

In a letter dated November 1, 2001, the applicant indicates that this TLAA is summarily described in the FSAR supplement. The TLAA summary is provided in Subsection 16.3.2.2 of Appendix A of the Turkey Point LRA.

Based on the fracture mechanics analysis documented in WCAP-15338, the applicant's responses to the staff RAI and the summary description of this TLAA described in the FSAR supplement, the applicant has provided an acceptable evaluation of the TLAA of the underclad cracks in the Turkey Point RPVs. Therefore, Open Item 4.3-1 is resolved.

The applicant indicates that an evaluation of the probability of reactor coolant pump flywheel failure was performed for the period of extended operation. The evaluation involved the potential fatigue crack initiation and growth in the flywheel bore keyway. The applicant indicated that the evaluation demonstrates that the flywheel design would have negligible crack growth over a 60-year service life. The applicant, therefore, concluded that the analysis remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The staff agrees with the applicant's assessment.

4.3.3 FSAR Supplement

The applicant's FSAR supplement for metal fatigue is provided in Appendix A, Section 16.3.2, of the LRA. The applicant described the TLAA evaluations and the transient cycle logging program. The staff requested that the applicant should update the FSAR supplement to provide a more detailed discussion of its proposed program to address environmental fatigue effects. The applicant provided additional discussion of its program to address environmental fatigue effects in Section 16.3.2.5 of the updated FSAR supplement. On the basis of its review of the updated FSAR supplement, the staff concludes the summary description of the applicant's actions to address metal fatigue for the period of extended operations is adequate.

4.3.4 Conclusions

On the basis of its projection of the number of expected transients, the applicant concluded that the fatigue analysis of RCS components and the RCP flywheel and B31.1 piping remain valid for the period of extended operation. In addition, the applicant has projected the reactor vessel underclad cracking analysis to a 60-year period of operation. The applicant also has an FMP to maintain a record of the transients used in the fatigue analyses of RCS components, and that process will continue during the period of extended operation. In the draft SER, the staff concluded the applicant's actions and commitments satisfy the requirements of 10 CFR 54.21(c)(1) after satisfactory resolution of the open item, identified in the draft SER. As discussed above, the open item has been adequately resolved. Therefore, the staff concludes that the applicant's actions and commitments satisfy the requirements of 10 CFR 54.21(c)(1).

4.4 Environmental Qualification

The Turkey Point Units 3 and 4 10 CFR 50.49 environmental qualification (EQ) program has been identified as a TLAA for the purposes of license renewal. The TLAA of EQ components includes all long-lived, passive and active electrical and instrumentation and control (I&C) components and commodities that are located in a harsh environment and are important to safety, including safety-related equipment, non-safety-related equipment whose failure could prevent satisfactory accomplishment of any safety-related function, and the necessary post-accident monitoring equipment.

The staff has reviewed Section 4.4, "Environmental Qualification," of the Turkey Point Units 3 and 4 LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1) for evaluating the EQ TLAA. The staff also reviewed Section 4.4.2, "GSI-168, 'Environmental Qualification of Electrical Components'," of the LRA.

On the basis of this review, the staff requested additional information in a letter to the applicant dated January 17, 2001. The applicant responded to this RAI in a letter to the staff dated March 30, 2001. The applicant provided a supplemental response to RAI B-3.2.6-1 on May 11, 2001, and to RAI 4.4.1-1 on May 29, 2001. In addition, the staff met with the applicant on October 31, 2000, to review related EQ calculations. The results of this meeting are documented in letter from the staff to the applicant dated December 22, 2000.

4.4.1 Summary of Technical Information in the Application

In Section 4.4 of the LRA, the applicant described its TLAA evaluation methodology and the results of its evaluations to demonstrate that (i) the analyses remain valid for the period of extended operation and (ii) analyses have been projected to the end of the period of extended operation. The following is a summary description of the Turkey Point Units 3 and 4 methodology used to evaluate the EQ TLAA.

Scope of EQ Equipment

The qualification requirements for electrical and I&C equipment installed at Turkey Point, Units 3 and 4 are based on NRC IE Bulletin 79-01B, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," which is now referred to as the Division of Operating Reactors (DOR) Guidelines. The Turkey Point Units 3 and 4 EQ program complies with the scope of 10 CFR 50.49 requirements and was "grandfathered" by 10 CFR 50.49, allowing qualification in accordance with the DOR Guidelines. Therefore, the DOR Guidelines document is the current licensing basis for the Turkey Point Units 3 and 4 EQ program.

The EQ program at Turkey Point Units 3 and 4 is a centralized plant support program administered by the design engineering group to maintain compliance with 10 CFR 50.49. The scope of the EQ program includes the following categories of electrical equipment located in a harsh environment:

- safety-related equipment,
- non-safety-related equipment whose failure could adversely affect safety-related equipment, and
- the necessary post-accident monitoring equipment.

The identification of equipment is procedurally controlled and the component database is utilized to maintain an EQ equipment master list.

EQ Process

The EQ program has three main elements:

- establishing and controlling a list of equipment and service conditions
- establishing and controlling equipment documentation
- maintaining qualification through preventive maintenance, the procurement process, and corrective actions

First, an EQ master list of equipment and the service conditions for the harsh environment plant areas is established and controlled. Next, the qualification documents are established and controlled, including vendor test reports, vendor correspondence, calculations, evaluations of equipment tested conditions as compared to plant required conditions, and determinations of configuration and maintenance requirements. Finally, required processes are established to maintain the qualification, including:

- a preventive maintenance process for replacing parts and equipment at required intervals
- a design control process to ensure changes to the plant are evaluated for impact on the EQ program
- a procurement process to ensure new and replacement equipment is purchased to applicable EQ requirements
- a corrective action process to identify and correct problems

Replacement of Equipment

As a normal part of the Turkey Point Units 3 and 4 EQ process, when the EQ documentation process establishes that equipment or parts thereof have a limited life, the preventive maintenance process ensures that the equipment or parts are replaced prior to the expiration of the qualified life. The Turkey Point Units 3 and 4 EQ program ensures that replacement equipment is purchased to applicable EQ requirements.

Analysis of the Qualified Life

The applicant evaluated the age-related service conditions that are applicable to environmentally qualified components (i.e., 60 years of exposure versus 40 years) for the period of extended operation to verify that the current environmental qualification analyses were bounding. The temperature and radiation values assumed for service conditions in the environmental qualification analyses are the maximum design operating values for Turkey Point. The thermal, radiation, and wear cycle aging effects were evaluated as follows:

- Thermal Considerations

The component qualification temperatures were calculated for 60 years using the Arrhenius method, as described in EPRI NP-1558, "A Review of Equipment Aging Theory and Technology." The Turkey Point EQ program utilizes ambient temperatures of 50°C for inside containment and 40°C for areas outside containment. For conservatism, a temperature rise of 10°C was added to the maximum ambient temperature for power cables to account for ohmic heating. The applicant stated that no power cables in the EQ program are normally energized, making the consideration of continuous ohmic heating a very conservative assumption for cable aging. This results in maximum design operating temperatures of 60°C inside containment and 50°C outside containment for these power cables and penetrations. If the component qualification temperature bounded the maximum design operating temperatures, then no additional evaluation was required. Additionally, the Technical Specification (TS) containment temperature limit is 48.9°C. The integrated maximum temperature profile for inside containment over Turkey Point's history has been below the TS limit of 48.9°C.

In 1991, new environmentally qualified Patel/EGS conformal splices and Patel/EGS Grayboot connectors that will not experience 60 years of thermal aging by the end of the license renewal period were installed at Turkey Point. Credits may be taken for less than 60 years of aging for these components.

- Radiation Considerations

The Turkey Point EQ program has established bounding radiation dose qualification values for all environmentally qualified components. These bounding radiation dose values were determined by component vendors through testing. To verify that the

bounding radiation values are acceptable for the period of extended operation, 60-year total integrated dose (TID) values were determined and then compared to the bounding values. The TID for the 60-year period is determined by adding the established accident dose to the 60-year normal operating dose for the component. The 60-year normal operating dose is obtained by multiplying the current 40-year normal operating dose by 1.5. The established post-accident dose is large when compared to the change in normal operating dose from 40 to 60 years and the original 40-year inside containment TID was rounded up. The current 40-year inside containment TID bounds the 60-year TID.

- Wear Cycle Considerations

The wear cycle aging effect is only applicable to ASCO solenoid valves at Turkey Point. ASCO has established a wear cycle limit of 40,000 cycles for these valves. The cycles for these valves were projected for 60 years and then compared to the limit provided by the vendor to establish acceptability for the period of extended operation.

The applicant used the margin values identified in Section 6.3.1.5 of IEEE 323-1974 in the EQ program. The only regular exception to the IEEE 323-1974 margins was for radiation. Additional margin need not be added to the radiation parameters if the methods identified in Appendix D of NUREG-0588 are utilized. The methods used to determine the Turkey Point radiation parameters are consistent with the Appendix D methodology. Accordingly, margin is adequately addressed in the Turkey Point EQ program.

Refurbishment of EQ Electrical Equipment

Refurbishment is an option at Turkey Point Units 3 and 4. EQ equipment that is in need of refurbishment is refurbished in place or is replaced with new equipment or previously refurbished equipment taken out of storage before the end of its qualified life. Refurbishment preserves the qualification status of equipment and is typically accomplished by replacing items such as gaskets, seals, and wires that are the limiting components or subcomponents for the qualified life. The EQ documentation identifies limited-life replacement parts for specific equipment, manufacturers, and models. The replacement option discussed for several types of equipment would effectively involve refurbishment. The Turkey Point EQ program and the procedures and administrative controls related to the Turkey Point EQ program are implemented in accordance with the requirements of Appendix B to 10 CFR Part 50 and will continue to be adequate for the period of extended operation. Replacement and refurbishment of EQ components is a part of the EQ program and its procedures.

The EQ program relies on specific equipment configurations, operational limitations, and bounding environmental limits. The program requires specific preventive or corrective actions to address the effects of aging (e.g., periodic part replacement) and restoration of configurations and conditions. The program also requires appropriate verification of these actions (e.g., documented completion of required maintenance activities). The documentation required by the EQ program, including the TLAAs, for each qualified component is maintained in an auditable form in accordance with the FPL quality assurance program.

Turkey Point maintenance and administrative procedures provide specific directions to maintenance personnel on what equipment to replace, when the equipment needs replacing, how to replace the equipment, and what post-maintenance testing needs to be performed to demonstrate that the item has been replaced correctly. Such procedures also provide forms required to document that the required maintenance actions have been completed, and the forms are maintained as quality assurance program records.

Ongoing Qualification or Retesting

For EQ equipment with a qualified life less than the required design life of the plant, “ongoing qualification” is a method of long-term qualification involving additional testing. Ongoing qualification or retesting, as described in IEEE Standard 323-1974, Section 6.6, “Ongoing Qualification,” paragraphs (1) and (2), is not currently considered a viable option by the applicant, and the applicant has no plans to implement it. If this option becomes viable in the future, ongoing qualification or retesting will be performed in accordance with accepted industry and regulatory standards.

Procurement of EQ Equipment

The EQ program has procurement processes to ensure that new and replacement equipment is purchased to applicable EQ requirements.

Plant Environmental Changes

Controls used to monitor changes in plant environmental conditions involve operating temperature and radiation monitoring in the containment. Containment temperature is monitored continuously by three temperature monitors at the 58 foot elevation of the containment to meet Technical Specification 3/4.6.1.5 (120 °F). Control room personnel record and log the values on every shift under all plant conditions. To ensure the monitored temperatures are bounding for the service environment of EQ equipment, the monitors are located at the highest level of EQ equipment inside containment. Since the qualified life calculations take into account increases in temperature due to self-heating and are done at a continuous temperature 2°F higher than the maximum continuous temperature allowed by the technical specifications, these monitors ensure that the qualified life of EQ equipment inside containment will not be exceeded. Containment area radiation levels are monitored continuously by three radiation monitors in various locations throughout each containment. (Note that these monitors are in addition to the safety-related high-range radiation, particulate, and gas monitors.) Turkey Point UFSAR Chapter 11.2 describes the area radiation monitoring system. High radiation activity from any of these areas is indicated, recorded, and alarmed in the control room. To ensure that the monitored radiation levels are bounding for the service environment for EQ equipment, the high alarm setpoint of the monitors is much lower than the values used for normal containment dose rates in EQ calculations.

Outside containment, the qualified life calculations are based on a continuous maximum design temperature of 104 °F. The only defined harsh temperature areas in the EQ program outside of containment are outdoors (e.g., main steam platforms). EQ list equipment in the auxiliary building is required to be qualified only for harsh radiation environments. Per Table 2.6-1 in the Turkey Point UFSAR, the actual average yearly temperature is between 74 °F and 76.2 °F. This 28 °F (15 °C) difference in temperature indicates that the qualified life based on the actual average temperature is more than double the life used by the Turkey Point analyses. Additionally, the area radiation monitoring system (14 monitors located throughout the auxiliary building that are indicated, recorded, and alarmed in the control room), daily operator walkdowns, health physics radiation monitoring, and maintenance and system engineering personnel provide feedback to engineering through FPL's corrective action program when the plant environment or EQ equipment changes. Because of the significant difference between the average temperature and the temperature used for qualified life calculations, the applicant would readily identify any change in temperature that could adversely affect qualification. The same applies to radiation. The dose calculations assume over 10 times the fuel leakage that has ever been experienced at Turkey Point. Turkey Point plant procedures govern the frequency of surveillances, radiation surveys, and plant walkdowns. The frequencies range from shiftily to annual, and the activities are performed during all modes of plant operation.

Containment temperature and radiation are logged at least daily, and operators walkdown all other EQ areas at least daily while the plant is operating. The temperature and radiation data obtained is representative of the service conditions of EQ equipment and any change in temperature or radiation that could adversely affect qualification would be readily identified.

EQ Generic Safety Issue (GSI)

GSI-168, "Environmental Qualification of Electrical Equipment," was developed to address environmental qualification of electrical equipment. The staff guidance to the industry (letter dated June 2, 1998, from NRC (Grimes) to NEI (Walters)) states:

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

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

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

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

For addressing issues associated with GSI-168, “Environmental Qualification of Electrical Components,” the applicant has chosen the second option. The applicant will continue to manage the effects of aging in accordance with the CLB and considers the evaluation of the EQ TLAA in Section 4.4 of the LRA to be the technical rationale that demonstrates that the CLB will be maintained during the period of extended operation.

4.4.2 Staff Evaluation

The staff reviewed Section 4.4 of the Turkey Point, Units 3 and 4 LRA to determine whether the applicant submitted adequate information to meet the requirements of 10 CFR 54.21(c)(1). In addition, the staff met with the applicant to obtain clarifications and to review specific EQ calculations, and reviewed the applicant’s response to the staff’s request for additional information.

The staff verified that applicant is using standard, approved EQ methodologies and acceptance criteria applicable to EQ as defined by NRC Bulletin 79-01B (the DOR Guidelines), including Supplements 1, 2, and 3; NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” Rev. 1; 10 CFR 50.49, “Environmental Qualification for Electric Equipment Important to Safety for Nuclear Power Plants”; Regulatory Guide 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Rev. 1; various NRC generic letters and information notices; and NRC safety evaluation reports on EQ. The current Turkey Point Units 3 and 4 actions for short-lived EQ equipment are also acceptable for long-lived EQ equipment.

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(i)

For the following list of electrical equipment identified in Section 4.4.1 of the LRA, the applicant uses 10 CFR 54.21(c)(1)(i) in its TLAA evaluation to demonstrate that the analyses remain valid for the period of extended operation:

- 4.4.1.1 Anaconda Cables
- 4.4.1.2 AIW Cables
- 4.4.1.3 ASCO Solenoid Valves
- 4.4.1.4 Brand Rex Coaxial Cables
- 4.4.1.5 Brand Rex Instrument Cables
- 4.4.1.6 Conax Conduit Seals

- 4.4.1.7 Conax Penetrations
 - 4.4.1.8 Conax Unitized Resistance Temperature Detectors
 - 4.4.1.9 Champlain Cables
 - 4.4.1.10 Crouse Hinds Penetrations
 - 4.4.1.11 General Atomic Radiation Monitors
 - 4.4.1.12 General Cables
 - 4.4.1.13 General Electric Cables
 - 4.4.1.14 General Electric Terminal Blocks
 - 4.4.1.15 Joy Emergency Containment Cooler and Emergency Containment Filtration Fan Motors
 - 4.4.1.16 Limitorque Valve Operators With Reliance Motors for Use Inside Containment
 - 4.4.1.17 Limitorque Valve Operators With Reliance Motors With Class H(RH) Insulation for Use Inside Containment
 - 4.4.1.18 Limitorque Valve Operators With Reliance Motors for Use Outside Containment
 - 4.4.1.19 Limitorque Valve Operators With Peerless Motors for Use Outside Containment
 - 4.4.1.20 Okonite Cables
 - 4.4.1.21 Raychem Heat Shrink Sleeving
 - 4.4.1.22 Raychem Cables
 - 4.4.1.23 Macworth Rees Pushbutton Stations
 - 4.4.1.24 Rockbestos Cables
 - 4.4.1.26 3M Insulating Tape and Scotchfil
 - 4.4.1.27 Westinghouse Residual Heat Removal Pump Motors
 - 4.4.1.28 Westinghouse Containment Spray Pump Motors
 - 4.4.1.29 Westinghouse Safety Injection Pump Motors
 - 4.4.1.30 Combustion Engineering Mineral-Insulated Cables and Connectors
 - 4.4.1.31 Kerite HTK/FR Cables
 - 4.4.1.32 Kerite FR2/FR Cables
 - 4.4.1.33 Kerite FR/FR Cables
 - 4.4.1.34 Kerite HTK/FR Power Cables
 - 4.4.1.35 Teledyne Thermatics Cables
 - 4.4.1.36 Weed Resistance Temperature Detectors
 - 4.4.1.37 Amertace NQB Terminal Blocks
 - 4.4.1.38 Patel/EGS Conformal Splices
 - 4.4.1.39 Patel/EGS Grayboot Connectors
- In response to the staff's concern regarding the aging effect of energizing the ASCO solenoid valves during testing, the applicant stated that the FPL calculation assumed that these are normally deenergized and the energization time during testing of the valves (which could be 1000 times over their lifetimes) was considered to be insignificant because it takes 2 hours for an ASCO solenoid valve to reach thermal

equilibrium once it is energized. In addition, 60 years is 29% of the calculated deenergized life of 207 years (EQ Documentation Package 3.0, Rev. 6). This leaves 71% of the calculated life as margin. Multiplying the calculated inside containment energized life of 4.6 years by 71% leaves 3.25 years that the normally deenergized solenoid valves could be energized over the 60-year qualified life. This would allow each of the 1000 cycles to remain energized for over 28 hours (3.25 years x 365.25 days x 24 hours/1000 cycles) plus the additional time to reach thermal equilibrium.

For example, 12 solenoid valves associated with the component cooling water to the emergency containment coolers energize to allow flow to the coolers whenever they are operated. EQ Document Package 16.0 indicates that the coolers undergo a 1-hour test once a month, two 1-hour maintenance tests per year, and 8 hours of other incidental operations per year. Therefore, the solenoid valves would not reach an equilibrium temperature and would operate less than 24 hours per year. Hence, aging due to energization time is insignificant. The staff concludes that this applicant addressed the staff's concern adequately.

- In response to the staff's concern regarding major plant modifications or events of sufficient duration to change the temperature and radiation values that were assumed in the EQ calculations, the licensee stated that there have been no major plant modifications or events at Turkey Point Units 3 and 4 that have changed the temperature and radiation values used in the EQ analyses. The postulated normal operating dose rates are based on the assumption of 1% failed fuel, which is 10 times the amount of fuel leakage that has been recorded at Turkey Point. The postulated accident doses are based on the conservative assumptions and methodologies in NUREGs-0578, -0737, and -0588. Any plant modifications that could affect the qualification of a component in the EQ program are addressed and resolved in the modification package. The effect of events on the qualification is addressed and resolved by the corrective action process. The staff concludes that the applicant addressed the staff's concern adequately.
- In response to the staff's request for the basis for 10°C rise above the maximum ambient temperature for power cables, the licensee stated that the 10 °C rise is conservative based on the maximum cable temperature rise of 3.2 °C for the 4160 VAC EQ motors of the safety injection and residual heat removal pumps. The applicant performed additional screening of the cable temperature rise for the 480 VAC EQ motors inside and outside containment including the emergency containment filter, emergency containment cooler, and containment spray pump motors. For the emergency containment cooler and filter motor cable inside containment, the temperature rises are 13.31 °C and 9.72 °C, respectively, above the 50 °C ambient.

For the emergency containment cooler, filter, and containment spray pump motor cable outside containment, the temperature rises are 22.89 °C, 9.39 °C, and 18.63 °C respectively, above a 40 °C ambient. Although the actual temperature rises are greater than the 10 °C continuous temperature rise assumption, when actual operating times of the emergency containment cooler and containment spray pump motors are considered (0.25 and 0.3 years, respectively, over a 60-year period), the 10 °C continuous

temperature rise assumption is over three times as harsh for both inside and outside containment. Therefore, the 10 °C rise applied continuously for 60 years is a conservative value for ohmic heating. The staff concludes that the applicant addressed the staff's concern adequately.

- In response to the staff's concern regarding the wear cycle aging effect on motors, MOV actuators, limit switches, and electrical connectors, the applicant stated that Limitorque cycled the actuators 2,000 times as part of the environmental qualification testing. The applicant determined that worst case cycling required for the MOV actuators would not exceed 2,000 over a 60-year plant life. The limit switches have a qualified life of less than 40 years based on thermal aging. The applicant adequately addressed wear cycle aging of electrical connectors. There are no TLAAs associated with limit switches and electrical connectors in the EQ program at Turkey Point. The applicant determined that the worst case wear cycles (start/stop cycles) would not exceed 1000 for Joy and Westinghouse motors over a 60-year plant life. The applicant stated that the wear cycling is normally not the limiting factor in the qualified life of the equipment and is not discussed in the qualification package. The applicant stated that a motor should be able to withstand 35000 to 50000 starts according to Volume 6 of the EPRI Power Plant Electrical Reference Series (page 6-46). Thus, the wear cycle aging effect is considered insignificant for these motors. In a letter dated May 29, 2001, the applicant committed to revise the EQ documentation packages for Westinghouse and Joy motors to include a reference to Volume 6 of the EPRI Power Plant Electrical Reference Series (page 6-46). This was confirmatory item 4.4.2-1. The staff reviewed the revised documentation package during the AMR inspection at the plant site during August 20-24, and September 10-14, 2001, and verified that the EQ documentation package has been revised accordingly. This response to the confirmatory Item 4.4.2-1 is acceptable.

On the basis of the staff's review of the information submitted by the applicant and the review of the EQ calculations on October 31, 2000, the staff finds the applicant's demonstration to be consistent with 10 CFR 54.21(c)(1)(ii). However, the applicant classified these TLAAAs under 10 CFR 54.21(c)(i). The applicant provided the following basis: (1) the activation energies, qualification temperatures, and methodologies were unchanged; (2) the current 40-year inside containment TID is bounding for 60 years; and (3) no new qualification testing and analyses were performed. The staff finds that the applicant's classification of these TLAAAs under 10 CFR 54.21(c)(1)(i) does not affect the technical adequacy of the equipment qualification and, hence, is acceptable.

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(ii)

For Samuel Moore cables (item 25 in Section 4.4.1 of the LRA), the applicant uses 10 CFR 54.21(c)(1)(ii) in its TLAA evaluation to demonstrate that the analyses have been projected to the end of the period of extended operation.

On the basis of the staff's review of the thermal and radiation summaries for the above electrical equipment and its review of the reanalysis of the Samuel Moore cables contained in FPL Document Package No. 25, Rev. 4, the staff finds the applicant's demonstration to be consistent with 10 CFR 54.21(c)(1)(ii).

4.4.3 FSAR Supplement

The staff reviewed Appendix A, Section 16.3.3, "Environmental Qualification," to the LRA and found that the licensee's EQ program as described will provide reasonable assurance that the functionality of systems, structures, and components requiring review will be maintained in the period of extended operation. This description is sufficient to satisfy the requirement of 10 CFR Section 54.21(d).

4.4.4 Conclusions

On the basis of the review described above, the staff has determined that there is reasonable assurance that the applicant has evaluated the time-limited aging analyses for EQ of electrical equipment consistent with 10 CFR 54.21(c)(1).

4.5 Containment Tendon Loss of Prestress

In Section 4.5 of the LRA, the applicant describes its time-limited aging analysis for containment tendon loss of prestress.

4.5.1 Summary of Technical Information in the Application

In Section 4.5 of the LRA, the applicant described the design configuration of the prestressing tendons in the prestressed concrete containment structures used in Turkey Point Units 3 and 4. The applicant described the factors contributing to the loss of prestressing force, and indicated that at the time of initial licensing, the magnitude of the prestress losses throughout the life of the plant was predicted and the estimated final effective preload at the end of 40 years was calculated for each tendon type. The final effective preload was compared with the minimum required preload to confirm the adequacy of the design.

Moreover, the applicant asserted that the new upper limit curves, the lower limit curves, and the trend lines of measured prestressing forces have been established for all tendons through the period of extended operation. The predicted final effective preload at the end of 60 years exceeds the minimum required preload for all containment tendons. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation. The analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

In addition, the applicant emphasized that the containment structure post-tensioning system surveillance performed as a part of the ASME Section XI Subsection IWL Inservice Inspection Program (described in Appendix B, Section 3.2.1.4, of the application), will continue to be performed as a confirmatory program, in accordance with the requirements of Technical Specifications 4.6.1.6.1 and 4.6.1.6.2.

4.5.2 Staff Evaluation

The applicant has performed the time-limited aging analysis to monitor the time-dependent characteristics of the prestressing forces for each group of tendons at each unit using the requirement in 10 CFR 54.21(c)(1)(ii). As a general rule, TLAAs must include the following:

- Analysis of the time-dependent assumptions of prestressing losses from 40 years through the extended period of operation. If the plant-specific operating experience indicates that the losses were underestimated, the new assumptions should be used in the analysis reflecting this experience. This analysis establishes the predicted lower limit (PLL) and the minimum required value (MRV) of the prestressing force at the end of the extended period of operation.
- Analysis of the effects of aging on the measured prestressing forces determined by trending the available data of the actual measured prestressing to the end of the extended period of operation if Option (ii) of 10 CFR 54.21(c)(1) is used for demonstrating that the trend line will stay above the PLL and MRV.

The applicant has established lower limit curves (same as PLLs) and the minimum required preload (same as MRV) for each group of tendons in each unit. The applicant has also established the trend lines based on the forces measured prestressing during prior inspections. Thus, this TLAA satisfies all the requirements of Option (ii) of 10 CFR 54.21(c)(1). However, to verify the adequacy of the applicant's analysis, in RAI 4.5-1, dated February 2, 2001, the staff requested the applicant demonstrate that after considering the projected loss of tendon prestress forces, the residual prestressing forces in each direction (i.e., hoop, vertical, and dome) will remain above the minimum required prestressing forces for the extended period of operation. To assess the adequacy of the analysis the applicant was requested to provide the following information:

- Curves showing the projected measured prestressing forces (i.e., trend lines) vs. the minimum required prestressing forces in each major direction, with a short description of the method used to project the measured forces (for both units, if different).
- How the trend lines represent the large number of exempt tendons (i.e., not subjected to lift off testing because of the personnel safety consideration).

During the staff's meeting with the applicant on April 11, 2001, the applicant discussed the requested curves. A summary of this information was provided in the RAI response dated April 19, 2001:

TENDON TYPE	TREND LINE VALUES		MINIMUM REQUIRED VALUE
	40 Years	60 Years	
Unit 3 Hoop	581 kips	572 kips	492 kips
Unit 4 Hoop	567 kips	558 kips	492 kips
Unit 3 Dome	680 kips	680 kips	531 kips
Unit 4 Dome	596 kips	588 kips	531 kips
Unit 3 Vertical	614 kips	612 kips	522 kips
Unit 4 Vertical	609 kips	601 kips	522 kips

As can be seen from the values provided in the table, the trended 60-year prestressing forces are well above the minimum required value established for the plant. Subsequent inspections may change the trend lines and the 60-year prestressing force predictions. The applicant would then be expected to address any adverse findings as they arise and take the necessary corrective actions.

In response to the staff's request for information on effects of exempt tendons (tendons excluded from sampling for the lift off testing), the applicant stated that the exempted tendons are subjected to the same environmental conditions as the tendons available for testing. Therefore, the trend lines generated from the large number of available tendons are representative of the small number of exempted tendons.

The staff concludes that the responses to the RAI are acceptable.

4.5.3 FSAR Supplement

In Section 5.1.3 of the UFSAR supplement, the applicant described the reanalysis of the containment structure because of the higher than estimated losses in the prestressing tendons in Turkey Point Units 3 and 4. The probable cause of the high losses was identified as increased wire steel relaxation caused by average tendon temperatures higher than those considered in the original design. The details of the reanalysis are provided in Appendix 5H of the UFSAR supplement. The results of the reanalysis concluded that after accounting for the increased prestressing losses, the established minimum required prestress would provide sufficient prestress force to maintain the Turkey Point licensing basis requirements through the licensed plant life (i.e., 40 years). The applicant extended the analysis related to the prestressing force to the end of the extended period of operation as discussed in Section 4.5 of the application, and evaluated by the staff in Section 4.5.2 of this SER. This description is sufficient to satisfy requirements of 10 CFR Section 54.21(d).

4.5.4 Conclusion

On the basis of its review of Section 4.5 of the application and relevant information in Section 3.2.1.2 of Appendix B and the UFSAR supplement of the application, the staff concludes that the applicant's approach in addressing this TLAA is reasonable and satisfies the requirement of Option (ii) of 10 CFR 54.21(c)(1).

4.6 Containment Liner Plate Fatigue

4.6.1 Summary of the Technical Information in the Application

In Section 4.6 of the LRA, the applicant presented the results of its TLAA for the containment liner plate and piping penetrations. The interior surface of the containments are lined with welded steel plate to provide an essentially leak-tight barrier. Design criteria are applied to the liner to assure that the specified leak rate is not exceeded under design-basis conditions.

Section 4.6 of the application lists the following design fatigue loads, as described in UFSAR Appendix 5B, Section B.2.1, that were considered in the fatigue design of the containment liner plates and piping penetrations:

- (1) 40 thermal cycles corresponding to 40 years of annual outdoor temperature variations, corresponding to the plant life of 40 years
- (2) 500 thermal cycles corresponding to containment interior temperature variations during RCS heatup and cooldown
- (3) One thermal cycle corresponding to the maximum hypothetical accident
- (4) The containment liner plate piping penetrations are isolated from the piping system thermal loads by concentric sleeves. These sleeves were designed in accordance with the 1965 Edition of the ASME Section III fatigue considerations as subject to the thermal load cycles of the piping system.

The fatigue design analysis of the containment liner plate and the piping penetrations, which considers these fatigue conditions, is considered to be a TLAA for the purposes of license renewal.

The applicant evaluated the above fatigue conditions for the period of extended operation. For item (a), the applicant stated that the increase in the number of cycles from 40 to 60 is considered to be insignificant, since the containment is designed for 500 heatup/cooldown cycles. For item (b), the applicant stated that the assumed 500 thermal cycles was evaluated based on the more limiting number of 200 heatup/cooldown design transients for the RCS. An evaluation described in Section 4.3.1 of the application determined that the originally projected number of maximum RCS design cycles is conservative enough to envelop the projected cycles for the extended period of operation, and therefore the original containment liner plate fatigue

analysis based on 500 heatup/cool-down cycles is considered valid for the period of extended operation. For item (c), the assumed value is considered to remain valid for 60 years of operation. For item (d), the applicant stated that the design of the containment penetrations meets the general requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III. The applicant identified the main steam piping, feedwater piping, blowdown piping, and letdown piping as the only piping penetrating the containment wall and the liner plate that contributes significant thermal loading on the liner plate. The applicant also stated that the projected number of actual operating cycles for these piping systems was determined to be less than the original design limits.

The applicant concluded that the assumed fatigue conditions in the containment liner plate penetrations fatigue analysis are bounding for 60 years of plant operation. Therefore, this TLAA remains valid for the period of extended operation and meets the criteria of 10 CFR 54.21(c)(1)(i).

4.6.2 Staff Evaluation

In Section 4.6 of the LRA, the applicant described three cyclic loading conditions that could affect the results of the original fatigue evaluation of the containment liner plate for the period of extended operation. The applicant concluded that extrapolation of these loads from 40 to 60 years would not have a significant effect on the fatigue of the containment liner plate and penetrations, and that the existing fatigue analysis remains valid. The staff found the information contained in Section 4.6 of the application insufficient to support this conclusion and requested additional information to permit completion of the review.

In RAI 4.6-1, dated February 2, 2001, the staff requested that the applicant provide the basis for determining that the original projected number of maximum design cycles for the containment liner plate and penetrations (500) is sufficiently conservative to envelop the projected number of cycles for the extended period of operation. In its response of April 19, 2001, the applicant stated that the containment liner plate was designed for 500 cycles of assumed RCS heatup/cool-down cycles, which is well above the design 200 heatup/cool-down cycles for the RCS. The applicant also stated that in its response to RAI 4.3.1-1, dated April 19, 2001, it had demonstrated that the total projected cycles of RCS heatup and cool-down, including the extended period of operation, are well within the original 200 cycle design limit. The staff finds the reference to the response to RAI 4.3.1-1 acceptable. The thermal loads in the containment liner are caused primarily by the heatup and cool-down of the RCS. Therefore, the 500 heatup/cool-down thermal cycles assumed for the containment liner plate also bound the expected number of cycles for the total life of the plant, including the period of extended operation. The staff concludes that the response to the RAI is acceptable.

The staff found that item (d) of Section 4.6 of the LRA contains insufficient information regarding the design of the containment penetrations to permit the conclusion that these designs meet the general requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III. In RAI 4.6-2, dated February 2, 2001, the staff requested verification that the fatigue analyses of the main steam piping, feedwater piping, blowdown piping, and letdown piping containment penetrations assemblies and welds include stresses due to

attached restrained piping system thermal expansion loads and stresses due to local thermal expansion.

In its response of April 19, 2001, to RAI 4.6-2, the applicant stated that Sections 5.1, "Containment Structure," and Appendix 5B, "Containment Structure Design Criteria," of the Turkey Point UFSAR provide descriptions of the containment penetration design qualification. The containment liner plate and penetrations have been evaluated in accordance with the rules and design criteria of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Article 4. As stated in the UFSAR, the evaluation of the penetrations considers stresses from the effects of pipe loads, pressure loads, thermal loads, dead loads, and earthquake loads, and the results meet the allowable stress criteria of Article 4, Paragraph N-414, of the Code. Article 4, Paragraphs N-412 and N-414, of the Code, require the consideration of the effects of external loads, pressure loads, and general and local thermal stresses when performing a fatigue analysis of these components. Appendix 5B of the UFSAR states the liner plate penetrations and concentric sleeves, shown in UFSAR Figure 5.1-16, are designed in accordance with the applicable fatigue requirements of the ASME Code. This figure indicates that piping thermal expansion loads were considered in the analysis of the piping penetrations. As stated in item (b), above, and Appendix 5B of the UFSAR, the containment liner was evaluated for 500 heat up/cool down cycles, which exceeds by a margin of 300 the maximum design heat up/cool down cycles of 200 for the RCS. As demonstrated in the response to RAI 4.3.1-1 above, the projected number of heatup/cooldown cycles for the RCS for the life of the plant, including the extended period of operation, is well within the original 200 cycle design limit of the RCS. On this basis, the applicant concluded that the analyses associated with the containment liner penetrations remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). The staff concurs with this assessment and considers the RAI issue resolved.

The staff noted that there is no discussion in Section 4.6 about containment pressure cycling due to integrated leak rate testing. Pressure cycling due to leak rate testing may have significant effects on the liner plate state of stress, and it wasn't evident from the discussion in Section 4.6 whether this was included in the fatigue analysis of the containment liner. In RAI 4.6-3, dated February 2, 2001, the staff requested additional information regarding this concern. In its response of April 19, 2001, to this RAI, the applicant stated that, in accordance with ASME Section III, the effects of leak rate pressure testing are included in the containment liner plate fatigue analysis. The staff finds this response acceptable to address the RAI issue.

The applicant stated that the effect of the increase in annual outdoor thermal cycles from 40 to 60 for the extended period of operation was insignificant in comparison with the assumed 500 thermal cycles of containment interior temperature variation due to RCS heatup/cooldown. Likewise, the assumed value of one thermal cycle due to the maximum hypothetical accident remains valid for the period of extended operation. The staff concurs with this assessment.

4.6.3 FSAR Supplement

The applicant updated UFSAR Chapter 16, Section 16.3.5, "Containment Liner Plate Fatigue," to reflect the change in thermal cycling due to outdoor annual temperature variation from 40 cycles to 60 cycles of plant life operation. FPL also provided a discussion showing that the

fatigue analysis of the containment liner plate and piping penetrations remains valid for the period of extended operation. The staff finds this acceptable.

4.6.4 Conclusion

On the basis of the review described above, the staff concludes that the applicant has provided adequate information and reasonable assurance to demonstrate that, pursuant to 10 CFR 54.21(c)(1)(i), the existing fatigue TLAA for the containment liner plate and piping penetrations remain valid for the period of extended operation.

4.7 Other Plant-Specific Time-Limited Aging Analyses

4.7.1 Bottom Mounted Instrumentation Thimble Tube Wear

In Section 4.7.1 of the LRA, the applicant described its TLAA on wear of incore instrumentation thimble tubes, which were mounted through the bottom of the reactor vessel. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the aging effects on the incore instrumentation thimble tubes will be adequately managed by this analysis during the period of extended operation as required by 10 CFR 54.21(a)(3).

4.7.1.1 Summary of Technical Information in the Application

The LRA stated that, in response to NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," the applicant established a program for inspection and assessment of thimble tube thinning. Two eddy current inspections of the thimble tubes for each unit were performed. The results showed that the thimble tubes were acceptable for operation and that no appreciable thinning had occurred between the two inspections. On the basis of the results of the inspections and the flaw analyses performed, only Unit 3 thimble tube N-05 will require further evaluation for the extended period of operation because it had the highest wear rate. The applicant further indicated that, in order to ensure thimble tube reliability, an inspection (one-time only) of Unit 3 thimble tube N-05 will be conducted (prior to the end of the initial operating license term) under the thimble tube inspection program described in Appendix B of the LRA.

4.7.1.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information in the LRA regarding the applicant's demonstration that the effects of aging will be adequately managed so that the intended function will be maintained consistent with the CLB for the period of extended operation for the incore instrumentation thimble tubes.

As described above, the TLAA on thimble tube thinning was based on results of two eddy current test inspections of the thimble tubes. These two inspections provided a gross status of tube thinning conditions at that time. It appears that the testing results were utilized to estimate wear rates. The wear rates were then used in the TLAA for justifying the adequacy of

performing surveillance on a single thimble tube (N-05 in Unit 3) during the one-time inspection described in the thimble tube AMP.

Since the wear rates used in the TLAA and the determination to conduct surveillance only on a single thimble tube are both based on information obtained from the early 1990s thimble tube testing, the staff is concerned whether such information remains valid for the TLAA. The staff finds that the wear rate may increase with time when flow-induced thimble tube vibrations become more severe due to increased wear, and the TLAA based on previous inspection results obtained in the early 1990s may not be realistic without verification. Confirmation is needed to ensure that an evaluation was performed in the TLAA to ensure adequate margin to cover potential uncertainties in wear rates. Such a concern was also stated in the staff review on Section 3.9.16 of the LRA. Consequently, the applicant was requested by letter dated February 1, 2001, to identify the wear rates, and to describe TLAA processes and results, including assumptions and analysis results used to justify that the acceptance criterion of 70% wall loss are met for extended operation of all thimble tubes except one tube (N-05 in Unit 3) because it had the highest wear rate.

In its response dated April 19, 2001, the applicant described the methodology, assumptions, and equation used to determine wear rate and time to predicted wall thickness, based on predictive models and calculations developed by WOG program on bottom-mounted thimble tubes. The program also determined that, although a thimble tube wall loss of up to 80% is acceptable, 70% is actually used as the allowable wall loss. Eddy current testing for detecting thimble wall thinning is considered accurate to plus or minus 10%. Each thimble tube has its unique wear rate, which was found following a decreasing exponential curve. Only thimble tubes with greater than 23% wall reduction need be considered, and no wear is assumed for other than full power operation of the plant. On the basis of the calculations performed on each of the tubes with greater than 23 % through-wall loss, the Unit 3 thimble tube at location N-05 was determined to be the worst case regarding the wall thinning rate, and to have the shortest remaining time to reach 70% through-wall loss. The tube with the next shortest remaining time has nearly twice the remaining time of tube N-05. In addition, according to Section 16.2.16 of the updated FSAR supplement in Appendix A, the thimble tube inspection program requires a one-time inspection on tube N-05 prior to the end of the initial operating license term for Turkey Point Unit 3, and the data of this inspection will be evaluated to determine the need for additional inspections. The staff found that the WOG program on thimble tubes in response to Bulletin 88-09 had been reviewed by the staff and is considered acceptable, and the specific calculations for Turkey Point thimble tubes had shown considerable margin regarding remaining life of all other thimble tubes tested, when compared with the remaining life of the thimble tube N-05 in Unit 3. Thus the staff concludes that it is acceptable to use the results of eddy current testing on tube N-05 for judging the acceptance of the other thimble tubes and for determining the need of further actions during the one-time inspection as defined in the thimble tube inspection program.

4.7.1.3 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description of the TLAA for emergency containment cooler tube wear contained in Section 16.3.7 of Appendix A of the LRA is acceptable.

4.7.1.4 Conclusion

The staff has reviewed the information in Section 4.7.1 of the LRA and responses to staff's RAIs. On the basis of this review, the staff concludes that the TLAA in Section 4.7.1 of the LRA provides an acceptable technical basis to justify the thimble tube inspection program, and the program will provide reasonable assurance that the effects of aging on the thimble tubes in Turkey Point Units 3 and 4 will be managed for early detection and timely corrective measures to mitigate potential thimble tube failure.

4.7.2 Emergency Containment Cooler Tube Wear

The applicant discusses the TLAA related to emergency containment cooler tube wear in Section 4.7.2 of the LRA.

4.7.2.1 Summary of Technical Information in the Application

The applicant states that the effect of increased wear due to erosion was previously evaluated and the tube wall nominal thickness was determined to exceed the minimum required wall thickness during the existing operating period of 40 years. In order to ensure emergency containment cooler coil reliability, a one-time inspection of minimum tube wall thickness will be conducted prior to the end of the existing operating period to further assess the actual tube wall thinning. The inspection will be conducted in accordance with the emergency containment coolers inspection described in Appendix B of the LRA.

4.7.2.2 Staff Evaluation

The component cooling water flow rate through the emergency containment coolers could exceed the nominal design flow during certain plant conditions. High flow rates can produce increased wear on the inside of the emergency containment cooler coils.

The emergency containment coolers inspection is a one-time inspection that will determine the extent of loss of material due to erosion in the emergency containment cooler tubes of Units 3 and 4. A sample of tubes with the greatest susceptibility to erosion will be selected and examined based on piping geometry and flow conditions. Commitment dates associated with the implementation of this new program are contained in Appendix A of the LRA.

The results of the inspection will be evaluated by Turkey Point to verify that the minimum required wall thickness for the emergency containment cooler heat exchanger tubes will be maintained during the period of extended operation.

In addition to tube wall loss, degradation of cooler frame and structural supports can occur due to the high humidity of the environment and the possible concentration of boron. In certain PWR units, boron coming from main line leak has been noticed in the vicinity of the cooler units. The applicant will inspect the frames and supports of the cooling units to ensure their structural integrity as part of its boric acid surveillance program. This program has proven to be effective in identifying and managing this degradation and the staff finds it acceptable.

The staff concludes that the applicant has provided an acceptable basis for extending the TLAA for the emergency containment cooler tubes to cover the extended period of license renewal and meets the requirements of 10 CFR 54.21(c)(1)(iii).

4.7.2.3 Conclusion

The staff concludes that the applicant has provided an acceptable TLAA of the emergency containment cooler tubes as defined in 10 CFR 54.3 and meets 10 CFR 54.21(c)(1)(iii).

4.7.3 Leak-Before-Break (LBB) for Reactor Coolant System Piping

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. Other plant-specific TLAAs are addressed in Section 4.7 of the LRA.

4.7.3.1 Summary of Technical Information in the Application

A plant-specific LBB analysis was performed for Turkey Point Units 3 and 4 in 1994. The LBB analysis was performed to show that any potential leaks that develop in the RCS loop piping can be detected by plant monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. As documented in the June 23, 1995, NRC letter to FPL, the NRC approved the Turkey Point LBB analysis. The NRC safety evaluation concluded that the LBB analysis was consistent with the criteria in NUREG-1061, Volume 3, and the draft Standard Review Plan, Section 3.6.3; therefore, the analysis complied with 10 CFR 50, Appendix A, General Design Criterion 4.

The applicant performed a plant-specific fatigue crack growth analysis for Turkey Point Units 3 and 4 for a 60-year plant life. A design transient set that bounds the Turkey Point design transients was utilized in the fatigue crack growth analysis. Fatigue crack growth for the period of extended operation was determined to be negligible.

The RCS primary loop piping LBB 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).

4.7.3.2 Staff Evaluation

The aging effects that were addressed during the period of extended operation include thermal aging of the primary loop piping components and fatigue crack growth. Thermal aging refers to the gradual change in the microstructure and properties of a material due to its exposure to elevated temperatures for an extended period of time. The only significant thermal aging effect on the RCS loop piping is embrittlement of the duplex ferritic cast austenitic stainless steel components. This effect results in a reduction in fracture toughness of the material.

The LBB analysis for Turkey Point Units 3 and 4 was revised to address the extended period of operation utilizing criteria consistent with the requirements of NUREG-1061, Volume 3, and the draft Standard Review Plan, Section 3.6.3, that the NRC referenced in approving the original LBB analysis. Since the primary loop piping includes cast stainless steel fittings, fully aged fracture toughness properties were determined for each heat of material. Based on loading, pipe geometry, and fracture toughness considerations, enveloping critical locations were determined at which the LBB crack stability evaluations were made. Through-wall flaw sizes were postulated at the critical locations that would cause leakage at a rate 10 times the leakage detection system capability. Large margins against flaw instability including the required margin for applied loads, were demonstrated for the postulated flaw sizes.

After the Turkey Point LRA was submitted, significant cracking of Alloy 82/182 weld metal was identified in the hot leg piping at a U.S. pressurized water reactor (PWR). Table 3.2-1 (page 3.2-68) of the LRA indicates that the nozzle safe ends were fabricated using stainless steel weld buildups. Since the cracking in the hot leg was associated with Alloy 82/182 weld metal, this issue does not affect the hot leg piping and nozzle safe ends at Turkey Point, Units 3 and 4.

4.7.3.3 FSAR Supplement

The staff has reviewed UFSAR Section 16.3.8 and confirmed that it provides a sufficient summary description, to satisfy the requirements of Section 54.21(d).

4.7.3.4 Conclusion

The staff concludes that the applicant has provided an acceptable basis for extending the TLAA for the leak-before-break analysis for the RCS piping to cover the time period of license renewal and meets the requirements of 10 CFR 54.21(c)(1)(ii).

4.7.4 Crane Load Cycle Limits

4.7.4.1 Summary of Technical Information in Application

In Section 4.7.4 of the LRA, the applicant identified the crane load cycle limit as a TLAA for the cranes within the scope of license renewal. They include the polar cranes, reactor cavity manipulator cranes, spent fuel pool bridge cranes, spent fuel cask crane, turbine gantry crane, and intake structure bridge crane. The applicant stated that the spent fuel pool bridge cranes were analyzed for up to 200,000 cycles of maximum load. The other cranes in the scope of license renewal were analyzed for up to 2,000,000 cycles of maximum load based on the design codes utilized for these cranes. In addition, the applicant stated that for each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed number of cycles. The applicant further stated that the 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).

4.7.4.2 Staff Evaluation

In order to determine the adequacy of the applicant's analyses, in a letter dated February 2, 2001, the staff requested the applicant to provide the load cycles experienced thus far and the cycles estimated to occur up to the end of the extended period of operation, including the conditions and assumptions used in its analyses for the applicable cranes. Also, the applicant was requested to provide the basis of the 200,000 load cycle limit for the spent fuel pool bridge cranes. The applicant responded to this RAI in its letter dated April 19, 2001. The applicant stated that actual crane usage is far less than qualified usage over the extended life of the plant. Consequently, the applicant does not count crane load cycles. The Turkey Point cranes are used primarily during refueling outages. Occasionally, cranes make lifts at or near their rated capacity (e.g., the turbine gantry crane lifting a turbine rotor). Usually, cranes make lifts substantially less than their rated capacity. However, conservatively assuming 200 lifts at or near rated capacity per refueling outage and 40 refueling outages in 60 years, results in 8000 cycles in 60 years. Also, the applicant stated that the spent fuel bridge cranes are used primarily to move fuel in the spent fuel pool. Conservatively assuming 400 lifts each refueling cycle (i.e., loading 60 new fuel assemblies, a full-core offload of 157 fuel assemblies, a full-core reload of 157 fuel assemblies, and 24 miscellaneous fuel assembly shuffles) and 40 refueling cycles in 60 years results in 16,000 cycles in 60 years. In addition, the applicant stated that the spent fuel pool bridge cranes are analyzed for up to 200,000 cycles of maximum load based on the crane manufacturer's calculations and the Crane Manufacturers Association of America (CMAA) Specification No. 70, "Specifications for Electric Overhead Traveling Cranes." On the basis that the actual usage of the crane over the projected life through the period of extended operation will be far less than the analyzed load cycles, the staff concludes that the Turkey Point cranes will continue to perform their intended function throughout the period of extended operation. Therefore, the applicant's response is acceptable.

4.7.4.3 FSAR Supplement

In Appendix A, Section 16.3.9, of the application, the applicant provided a summary description of the evaluation of the crane load cycle limit. The applicant stated that the load cycles for these cranes were evaluated for the period of extended operation. For each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed load cycles and, therefore, all cranes in the scope of license renewal will continue to perform their intended function throughout 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 54.21(d).

4.7.4.4 Conclusion

The staff has reviewed the information in Section 4.7.4 and Appendix A, Section 16.3.9, as well as the additional information provided in the applicant's letter dated April 19, 2001. On the basis of the review provided above, the staff concludes that the applicant has provided adequate information to meet the requirements of 10 CFR 54.21(c)(1)(i) related to the TLAA for the crane load cycle limits.