3.9.1.3.3 FSAR Supplement

The applicant provided the updated FSAR Supplement, in Section 16.2.1.3 of Appendix A to the LRA, which states that ASME Section XI, Subsection IWF Inservice Inspection Program inspections identify and correct degradation of ASME Class 1, 2, and 3 component supports. The staff concludes that the updated FSAR Supplement is sufficient.

3.9.1.3.4 Conclusion

Based on the information provided by the applicant, the staff concludes that the continued examinations performed under the ASME Section XI, Subsection IWF inservice inspection program provide reasonable assurance that the aging effect of loss of material for the Class 1, 2, and 3 components and piping supports within the scope of license renewal will be managed for the period of extended operation.

3.9.1.4 ASME Section XI, Subsection IWL Inservice Inspection Program

This SER section addresses the review of Section 3.2.1.4 of Appendix B to the LRA related to Subsection IWL of the ASME Section XI inservice inspection program.

3.9.1.4.1 Summary of Technical Information in the Application

in Chapter 3 of the LRA, the applicant stated that this program is credited for aging management of post-tensioning system structural components in the containments. The applicant's description of the program addressing the seven program elements is discussed in 3.9.1.4.2.

3.9.1.4.2 Staff Evaluation

It is noted that corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and covers all SSCs subject to an aging management review. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this safety evaluation report. This program satisfies the elements of corrective actions, confirmation process and administrative controls. The remaining seven (7) elements are discussed below.

[Program Scope] The scope of the program provides for inspection of tendon wires and tendon anchorage hardware surfaces for loss of material, as well as a confirmatory program for measurement of tendons for loss of prestress.

In RAI 3.9.1.4-1, the staff stated that the applicant had credited the ASME Section XI, Subsection IWL, for aging management of the containment post-tensioning system components. However, Subsection IWL of Section XI of the ASME Code is established as a required program for the inservice inspection of concrete and post-tensioning system. The staff asked the applicant to provide a description of a program for managing the aging of containment concrete, including, inspection interval, personnel qualifications, examination method(s), acceptance criteria, and quality assurance requirements in lieu of its reference to subsection IWL, which does not contain specific acceptance criteria for examination of concrete. The staff requested the applicant to revise the discussion in Section 3.2.1.4 to incorporate specific acceptance criteria for examination of concrete in an overall ISI program to be used for aging management of the containment post-tensioning system component.

In its response, the applicant argued that there are no aging effects that could cause a loss of intended function for the containment concrete above groundwater. At the same time, the applicant recognized the existence of concrete degradations depicted in Appendix A to NUREG-1522. The applicant proposed to modify its exclusive reliance on the ASME Section XI, Subsection IWL in the description of the aging management program in Section 3.2.1.4 of Appendix B to the LRA to include aging management of containment reinforced concrete above ground water. In a letter dated April 19, 2001, the applicant stated, "the Turkey Point ASME Section XI, Subsection IWL Inservice Inspection Program was developed considering ACI 201.1R 68 (Revised 1984), 'Guide for Making a Condition Survey of Concrete in Service,' to establish degradation type and IWL-3211 acceptance criteria." As supplemented by the RAI response, the staff considers this issue to be resolved.

*[Preventive Actions]* The applicant described the presence of two mechanisms that serve as preventive actions: (1) a layer of low-strength nonstructural concrete is provided to prevent the intrusion of rainwater under the grease caps of the top anchorages of vertical tendons, and (2) all the metallic components (such as reinforcing bars, liner plate, and tendon anchorages) are interconnected to an impressed current cathodic protection system (CPS). Additionally, the applicant states that the CPS is not credited in the determination of the aging effects requiring management.

A number of components (e.g., reinforcing bars, tendon anchorage components) to which the CPS is connected are embedded or not available for direct examination. Depending upon the reliability of the continuous source for applying impressed current, the CPS may or may not be effective at certain times (power outage, low battery). Such incidents could lead to adverse effects on the protected components. Thus, if the CPS is relied upon for preventing corrosion of the protected components, its effectiveness in performing its function has to be periodically assessed. The staff requested more information regarding ensuring the effectiveness of the CPS. During the AMR inspection in August — September 2001, the inspectors reviewed the procedures and records and concluded that the applicant has adequate procedures and sufficient surveillance that the staff's concern is resolved.

*[Parameters Monitored or Inspected]* In accordance with ASME Section XI, Subsection IWL, unbonded post-tensioning system components are examined. These components consist of tendons, wires or strand, anchorage hardware and surrounding concrete, corrosion protection medium, and free water. Surface conditions are monitored through visual examinations to determine the extent of corrosion or concrete degradation around anchorage locations. Prestress forces are measured for sample tendons to determine loss of prestressing force. Tension tests are performed on wire or strand samples removed from tendons to be examined for corrosion and mechanical damage. As discussed in *{Program Scope}*, the applicant has committed to monitor the parameters associated with the degradation of concrete containment surfaces. The staff considers the program element acceptable.

*[Detection of Aging Effects]* The presence of age-related degradation is determined by visual inspection or by measurement. Tendon anchorage hardware is examined for corrosion. A select number of tendons are completely detensioned, and a sample wire from each group of tendons is examined for the presence of corrosion and tested to verify ultimate strength. Tendon anchorage hardware and concrete surfaces are examined for corrosion protection medium (grease) leakage and the tendon caps are examined for deformation. As discussed in *[Program Scope]*, the applicant has committed to monitor and detect aging effects in the above ground and below ground containment concrete surfaces. Thus, the staff finds the element acceptable.

*[Monitoring and Trending]* The applicant stated that the first period containment inspections are scheduled for completion by September 9, 2001, as required by 10 CFR 50.55a. The tendon inspections are performed as required by Subsection IWL of the ASME Section XI Code (the Code). Subsection IWL requires the evaluation of loss of material of the tendon components, and loss of prestress (the principal age-related effects on post-tensioning system components). Thus, these aging effects will be monitored and trended. As described, the staff finds this program element is acceptable.

*[Acceptance Criteria]* The results of inspections (performed in accordance with the requirements of Subsection IWL of the Code) are evaluated against the acceptance standards in the IWL. As discussed in *[Program Scope]*, the applicant has committed to implement the acceptance criteria IWL-3211 for concrete examination during the extended period of operation. As described, the staff finds this element description acceptable.

*[Operating Experience and Demonstration]* The applicant describes its operating experience related to the post-tensioning tendon system as follows:

The measured lift-off forces for a number of randomly selected surveillance tendons were below the predicted lower limit. Condition Reports and a Licensee Event Report were issued. In accordance with the Technical Specifications, engineering evaluations were prepared and concluded that the lower than expected tendon lift-off forces were caused by greater than expected tendon wire relaxation losses due to average tendon temperatures higher than originally considered.

To accommodate the increased prestress losses, a license amendment was submitted and approved to reduce the containment design pressure from 59 psig to 55 psig, and a containment reanalysis was performed to determine the new minimum required prestress forces to maintain Turkey Point licensing-basis requirements. The results of the reanalysis are provided in the UFSAR, Section 5.1.3. The ASME Section XI, Subsection IWL Inservice inspection program incorporates all of the inspection criteria and guidelines of the previous tendon inspection program attributes, and is implemented using existing plant procedures.

Based on the inspections performed prior to the implementation of Subsection IWL as part of the operating experience, RAI 3.9.1.4-3 asked the applicant to provide a summary of significant events related to the following causative agents:

- containment concrete (e.g., dome delamination, wide-spread scaling)

- containment prestressing force (unusual systematic losses) (closed based on the information provided in the UFSAR supplement)
- corrosion of post-tensioning system hardware (breakage of wires or anchor-head components)
- grease leakage through concrete

The applicant was also asked to include the corrective actions taken and procedures modified to alleviate such events in the future and to provide a description of the condition of tendon gallery environment and measures implemented to control it to alleviate the corrosion of vertical tendon anchorage hardware.

The applicant described the operating experience related to the above items in its response to the RAI. Based on the response, the staff finds that the applicant has adequately considered the plant-specific, as well as industry-wide experience in evaluating the aging management program for the extended period of operation. Therefore, RAI item 3.9.1.4-3 is closed and this program element is acceptable.

3.9.1.4.3 FSAR Supplement

A review of the UFSAR Supplement indicates that the response to RAI 3.9.1.4-3 is fully described in Section 5.1.3, and in Appendix 5H of the supplement Section 16.2.1.4 provides a sufficient summary of the program.

3.9.1.4.4 Conclusion

Based on its review, the staff concludes that this aging management program provides reasonable assurance that the aging of the concrete containment components (i.e., concrete and post-tensioning system components) of the primary containment structures at Turkey Point, Units 3 and 4, will be adequately managed during the period of extended operation.

3.9.2 Boraflex Surveillance Program

The applicant described the Boraflex surveillance program in Section 3.2.2, "Boraflex Surveillance Program," of Appendix B to the LRA. The application of this program is provided in descriptions found in Section 3.6.2.2, "Steel-in-Fluid Structural Components," of the LRA. The staff reviewed the application to determine whether the applicant has demonstrated that the effects of aging covered under the Boraflex surveillance program will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.9.2.1 Summary of Technical Information in the Application

The Boraflex surveillance program is credited for managing the aging effect of material changes in the Boraflex poison material found in the spent fuel storage racks. Currently, this program includes blackness testing and tracking of SFP silica levels as qualitative indicators of Boraflex degradation. The applicant states that prior to the end of the initial operating terms for Turkey Point, Units 3 and 4, this program will be enhanced to provide for density testing (or other approved testing methods). In response to the staff's RAI, the applicant stated that the enhancement to this program is the performance of density testing on the racks in lieu of blackness testing. This program enhancement is discussed in the staff's safety evaluation to amendment No. 206 to facility operating license No. DPR-31 and amendment No. 200 to facility operating license No. DPR-41 transmitted by NRC letter dated July 19, 2000.

3.9.2.2 Staff Evaluation

The staff evaluation of the Boraflex surveillance program focused on how the activities managed aging effects through the effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

*[Program Scope]* The Boraflex surveillance program is applied to the boron-impregnated polymer, Boraflex, found in the SFP storage racks. The staff agrees that it is appropriate to include this material component within the scope of the Boraflex surveillance program.

*[Preventive or Mitigative Actions]* The Boraflex surveillance program has no associated preventive or mitigative actions. There are no known methods of preventing the loss of boron carbide and the eventual release of silica since the Boraflex polymer matrix breaks down over time due to the convective aqueous environment of the SFP. The staff agrees that there are no preventive or mitigative actions to prevent the further break down of the polymer matrix and eventual release of boron carbide into the SFP. However, based on the known mechanism governing the polymer matrix breakdown, Boraflex degradation can be retarded by limiting disturbances to the SFP and maintaining silica equilibrium between the panel and the surrounding water. In response to the staff's RAI, the applicant stated that the SFP purification system has a low turnover rate, a low propensity to remove soluble boron, and no special measures are taken to reduce silica concentration. On the basis of this response, the staff concludes that the applicant's current program adequately accounts for the mechanism of Boraflex degradation.

*[Parameters Monitored or Inspected]* The application describes the current program consisting of blackness testing which confirms the inservice Boraflex panel performance data in terms of gap formation, gap distribution, and gap size. In addition, trending of the SFP silica levels is conducted to give a qualitative indication of boron carbide loss from the panels. The enhanced Boraflex surveillance program will include checking the density (or other approved methods) to ascertain the physical loss of boron carbide.

The staff agrees that blackness testing will provide information regarding gap formation consistent with the description of the change in material properties, due to irradiation, given in Section 3.6.2.2.2 of the LRA. However, the staff requested the applicant to justify the non-inclusion of the change in material properties due to both irradiation and convective forces in the SFP (i.e., a change in material properties due to dissolution of the Boraflex panel). In response to the staff's concerns, the applicant responded that the enhancement of this program evaluates changes in material properties due to dissolution of the panel through the determination of boron areal density. The applicant further specified that results from this determination will be compared with the required minimum boron areal density to indicate the panels' condition. On the basis of this response, the staff concludes that the parameters inspected and monitored under this program are appropriate and adequate to determine degradation of the Boraflex panels in the spent fuel racks.

*[Detection of Aging Effects]* The application states that the presence of silica in the SFP water, which is periodically monitored, is a physical sign of the aging effect occurring in the Boraflex material. In addition, the application states that the enhanced Boraflex surveillance program will determine the amount of degradation of the Boraflex material.

Although the applicant discusses blackness testing in the introduction of this AMP, blackness testing is not discussed as a means of detecting the aging effect of gap formation in the Boraflex panels. In addition, the applicant stated that trending of silica concentration in the SFP gives an indication of Boraflex degradation; however, this indication does not provide the degree to which the Boraflex has degraded. In response to the staff's concerns, the applicant responded with further details regarding the enhancement of this AMP. The applicant stated that this program will be enhanced to include areal density testing of the panels which will be completed in lieu of blackness testing. The staff finds that this method of testing the panels, in conjunction with silica concentration monitoring, is more effective than blackness testing alone and is adequate in detecting the aging effects associated with degradation of the Boraflex panels.

*[Monitoring and Trending]* The application states that shrinkage, gaps, and density will be monitored during scheduled Boraflex surveillance testing and that subsequent Boraflex tests will be scheduled following evaluation of the measured results. The application continues stating that trends will be established following implementation of the enhanced Boraflex surveillance program.

The staff finds that it is appropriate and prudent to monitor and trend shrinkage, gap formation, and density changes of the Boraflex panels. However, the staff requested the applicant to clarify how these parameters are currently trended and analyzed. In addition, the staff requested the applicant to provide details of how the enhanced program will affect the current analyses of these parameters. In response to the staff's concerns, the applicant stated that data from the periodic surveillances are evaluated to determine the number, size, and location of shrinkage and gaps within and among the tested panels. The data is further compared with the criticality analysis assumptions which govern the SFP to confirm that the analysis continues to bound the observed data. The enhanced program will continue to obtain data related to shrinkage and gaps but will also include data related to the density of the panels. The additional data will also be evaluated and compared with the criticality analysis assumptions. The staff finds these methods appropriate and acceptable for monitoring and trending the degradation of the Boraflex panels.

*[Acceptance Criteria]* The acceptance criteria provided in the application for Boraflex degradation are controlled by the assumptions in the criticality analysis. The applicant states that the results of each surveillance are used to ensure that 5% criticality margin will be maintained.

The staff agrees that the acceptability of Boraflex degradation should be controlled by the assumptions in the criticality analysis. However, the staff requested the applicant provide details regarding how the surveillance results ensure that the 5% subcriticality margin will be maintained. In response to the staff's concerns, the applicant stated that the data related to the enhancement to this program (i.e., areal density) will be used in conjunction with shrinkage and gap formation to evaluate the assumptions governing the 5% subcriticality margin. On the basis of this information and clarifying information provided in other responses related to staff's concerns regarding this program enhancement, the staff concludes that this enhanced program has appropriate acceptance criteria in ensuring that the Boraflex panels continue to meet their intended function.

*[Operating Experience]* The application states the current Boraflex surveillance program was initiated following installation of high density SFP racks. The results of this program have indicated that Boraflex degradation is occurring due to accumulation of silica in the SFP water. The application further discusses that the blackness testing performed once every 5 years has demonstrated that the technical specification for maintaining the subcriticality margin has been met. On the basis of this discussion, the applicant concluded that the continued implementation of the Boraflex surveillance program provides reasonable assurance that the effects of aging will be adequately managed for the period of extended operation.

The staff requested the applicant (RAIs dated February 1, 2001) provide further details supporting the adequacy of the current program in determining the effectiveness of the degraded Boraflex panels currently in the SFP. Blackness testing is an appropriate method for determining gap formation in the panels but is not indicative of the concentration of boron carbide remaining in the panel. In addition, the staff requested the applicant to discuss how the enhanced Boraflex surveillance program will support conclusions drawn from the applicant's operating experience. On the basis of the staff's concerns, the applicant provided in a letter dated April 19, 2001, clarifying details of the enhanced program which includes areal density testing of the Boraflex panels. On the basis of the details provided in the responses to various aspects of this program, the staff concludes that the applicant's enhanced program will adequately address the Boraflex degradation experience at Turkey Point, Units 3 and 4.

### 3.9.2.3 FSAR Supplement

Based on the responses provided to the staff's RAIs, the staff requests the applicant to update Chapters 14 and 16 of the UFSAR Supplement found in Appendix A to the LRA, to include a description of all applicable aging effects of Boraflex and the program enhancement discussed in the staff's SER to amendment No. 206 to facility operating license DRP-31 and amendment No. 200 to facility operating license No. DRP-41 transmitted by NRC letter dated July 19, 2000. This is confirmatory item 3.9.2-1.

3.9.2.4 Conclusion

The staff has reviewed the Boraflex surveillance program, described in the following sections of the application: Sections 3.2.2, "Boraflex Surveillance Program," and 3.6.2.2, "Steel-in-Fluid Structural Components," of Appendix B and responses to staff's RAIs. On the basis of the review, the staff concludes that there is reasonable assurance that the Boraflex surveillance program, with the stated enhancements, will adequately manage the aging effects of gap formation and dissolution of the Boraflex panels in the SFP racks in accordance with the CLB during the period of extended operation.

3.9.3 Boric Acid Wastage Surveillance Program

The applicant described the boric acid wastage surveillance program in Section 3.2.3, "Boric Acid Wastage Surveillance Program," of Appendix B to the LRA. The application of this program is credited for managing the aging effects associated with cast iron, carbon steel and low alloy steel components/commodities found in the following systems and structures: auxiliary building ventilation, chemical and volume control, CCW, containment isolation, containment post-accident monitoring and control, containment spray, electrical/I&C components, emergency containment cooling, emergency containment filtration, feedwater and blowdown, fire protection, instrument air, intake cooling water, main steam and turbine generators, normal containment and control rod drive mechanism cooling, primary water makeup, reactor coolant, residual heat removal, safety injection, sample, SFP cooling, waste disposal, auxiliary building, containments, spent fuel storage and handling, and yard structures. The staff reviewed the application to determine whether the applicant has demonstrated that the aging effects covered by this activity will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.9.3.1 Summary of Technical Information in the Application

The boric acid wastage surveillance program manages the effects of loss of material and loss of mechanical closure integrity due to aggressive chemical attack of cast iron, carbon steel, and low alloy steel components and structural components including bolting. The program encompasses mechanical closures (e.g., bolted connections, valve packing, pump seals) and electrical structural components (e.g., enclosures, cable trays, conduits). This program will be enhanced to include some systems outside containment (i.e., SFP cooling and portions of the waste disposal associated with containment integrity) currently inspected under other existing programs. The enhancement does not reflect additional inspection activities but a modified grouping to include inspections currently completed in other activities.

3.9.3.2 Staff Evaluation

The staff requested additional information dated February 2, 2001, from the applicant with respect to the enhancement of this program. Specifically, the staff requested the applicant provide details discussing how the systems outside containment, currently inspected under other existing programs, will continue to be inspected under the enhanced boric acid wastage surveillance program. In a response, dated April 19, 2001, the applicant stated that this program will be enhanced to include the SFP cooling and waste disposal system which is currently inspected under the systems and structures monitoring program described in Section

3.2.15, "Systems and Structures Monitoring Program," of Appendix B to the LRA. The applicant further stated that applicable procedures will be enhanced to provide additional guidance for evaluating potential effects of boric acid leakage (i.e., boric acid corrosion) on adjacent components and structural components. In addition, the procedures currently require leakage testing to be corrected or evaluated but do not explicitly address the potential for corrosion of adjacent components subjected to borated water. The staff has reviewed this information and has determined that this enhanced program will continue to inspect these additional systems in a manner similar to the current inspection program.

The staff evaluation of the boric acid wastage surveillance program focused on how the activities managed aging effects through the effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this safety evaluation report. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are evaluated below.

*[Program Scope]* The boric acid wastage surveillance program is applied to various cast iron, carbon steel, and low alloy steel components and structural components including bolting found in various systems and structures exposed to borated water. The program includes systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not compromise the pressure boundary or structural integrity of components, supports or structures. In addition, this program includes electrical structural components in proximity to borated water systems, and will be enhanced to include inspections currently completed under other existing programs.

The staff agrees that it is appropriate and prudent to include components constructed from cast iron, carbon steel and low alloy steel. However, the external surfaces of other materials are also susceptible to corrosion from exposure to concentrated boric acid. The staff requested the applicant to discuss the non-inclusion of components constructed from aluminum, brass, bronze, carbon, and galvanized steel which may also be exposed to the corrosive boric acid environment. The applicant responded that other metals such as copper, copper alloys, nickel, nickel alloys, and aluminum are resistant to boric acid corrosion and therefore, loss of material due to aggressive chemical attack does not require management for these materials. The staff has reviewed this information and has concluded that the severity of the chemical attack on surrounding components is dependent on the concentration of boric acid. However, the staff notes that the methods in this program for monitoring and preventing the aging effects associated with boric acid are appropriate and adequate in controlling boric acid wastage on surrounding components.

*[Preventive or Mitigative Actions]* The applicant states that preventive actions included in the boric acid wastage surveillance program are removal of concentrated boric acid and boric acid residue and the elimination of boric acid leakage.

The staff agrees that these actions are applicable and prudent in mitigating corrosion by minimizing the exposure of susceptible material to the corrosive environment.

*[Parameters Monitored or Inspected]* The applicant states that this program monitors the effects of boric acid corrosion on the intended function of the component by detection of coolant leakage discussed in NRC Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," including guidelines for locating small leaks, conducting examinations, and performing evaluations. The applicant further states that crystal buildup and evidence of moisture are conditions that lead to boric acid corrosion.

The staff finds that the detection of coolant leakage through evidence of crystal buildup and moisture is acceptable because these are conditions which are directly related to the degradation of components exposed to boric acid.

*[Detection of Aging Effects]* The applicant states that degradation of components cannot occur without leakage of coolant containing boric acid. Visual inspections resulting from discovery of crystal buildup and evidence of moisture are performed on external surfaces in accordance with plant procedures and are used to indicate leakage of coolant containing boric acid.

The staff finds that the discovery of crystal buildup and/or moisture is an appropriate and acceptable method of determining coolant leakage which will eventually lead to corrosion of the material. However, in the case of electrical cables or insulated piping, discoloration of the insulation is also indicative of a coolant leakage. The staff requested the applicant to provide additional information related to the adequacy of this program in identifying aging effects prior to the loss of component intended function. The applicant, in its response, stated that if insulated piping or electrical cables show signs of boric acid leakage (e.g., boric acid residue), the source of the leakage is determined and corrected. In addition, the applicant stated that the commitments to GL 88-05 have been aggressively implemented and that a review of plant history shows minor leaks which have been corrected. The applicant noted that none of the identified leaks resulted in significant component/system degradation or loss of intended function. On the basis of the information provided, the staff finds that the AMP includes appropriate and adequate methods for detecting boric acid leaks prior to affected component loss of function.

*[Monitoring and Trending]* The applicant states that leakage calculations are performed each shift. If identified or unidentified RCS leakage is greater than 0.5 gpm, an RCS leakage investigation is initiated to identify and address the source of the leakage. In addition, during each refueling, inspections of systems inside containment are performed. Every 18 months, inspections of borated water systems outside containment are performed.

The staff finds these frequencies acceptable and appropriate given the description of the applicant's operating history and industry practice of inspecting systems inside containment every refueling outage.

*[Acceptance Criteria]* The applicant states that all cases of boric acid leakage are either corrected or evaluated. The staff requested the applicant to provide details regarding the evaluation of a boric acid leakage discovery including specific evaluation criteria and the bases for such criteria. In response to the staff's request, the applicant stated that this AMP implements commitments made through the applicant's response to GL 88-05 including guidelines for locating small leaks, conducting examinations and performing evaluations. In addition, leakage evaluations are performed under the applicant's corrective actions program and consider the location and characteristics of the leak, the component's function, other systems affected by the leak, operability requirements, technical specifications, and the UFSAR. On the basis of the information provided, the staff finds that appropriate and adequate acceptance criteria for detecting and correcting boric acid leaks are implemented through this AMP.

*[Operating Experience]* The applicant states that this program was originally implemented as a result of boric acid leaks experienced at Turkey Point and NRC GL 88-05. The program addresses the generic letter requirements including: (1) detection of principal location where coolant leaks are smaller than allowable TS limits, (2) methods for conducting examinations which are integrated into ASME Code VT-2 inspections; and (3) corrective actions to prevent recurrences of this type of leakage. Since establishing the program, the applicant asserts that there have been no instances of boric acid corrosion that have impacted license renewal system intended functions.

The staff finds that the applicant has demonstrated the boric acid wastage surveillance program has been effective in preventing damage to components due to exposure to concentrated boric acid.

#### 3.9.3.3 FSAR Supplement

In Section 3.9.3.3 of the SER with open items, the staff requested that the applicant update the UFSAR Supplement with a summary description of the Boric Acid Waste Surveillance program. By letter dated November 1, 2001, the applicant provide the requested information in Section 16.2.3 of Appendix A to the LRA. The staff finds the summary description acceptable, and therefore confirmatory item 3.9.2-1 is closed.

#### 3.9.3.4 Conclusions

The staff has reviewed the boric acid wastage surveillance program described in Section 3.2.3, "Boric Acid Wastage Surveillance Program," of Appendix B and various sections of the LRA and responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that there is reasonable assurance that the boric acid wastage surveillance program will adequately manage the aging effects of various components susceptible to the corrosive element of boric acid in accordance with the CLB during the period of extended operation.

### 3.9.4 Chemistry Control Program

This program is covered in Section 3.1.1 of this safety evaluation report.

### 3.9.5 Containment Spray System Piping Inspection Program

#### 3.9.5.1 Summary of Technical Information in the Application

Containment spray is designed to remove sufficient heat to maintain the containment below its design pressure and temperature during a loss-of-coolant accident or main steam line break. Containment spray is composed of two motor-driven horizontal centrifugal pumps, each discharging to two spray lateral headers located near the top of the containment structure. The system also utilizes the residual heat removal pumps and heat exchangers for the long-term recirculating phase of containment spray, as described in section 2.3.2.5 of the LRA. Additionally, containment spray provides a source of water for the emergency containment filtration spray (see Subsection 2.2.2.6 of the LRA). Components associated with this function are included in the scope of emergency containment filtration. Containment spray is described in UFSAR Section 6.4

The flow diagrams listed in Table 2.3-4 of the LRA show the evaluation boundaries for the portions of containment spray that are within the scope of license renewal.

Containment spray is within the scope of license renewal because it contains structures and components that are safety-related and are relied upon to remain functional during and following design-basis events and structures and components that are a part of the environmental qualification program.

Containment spray components subject to an aging management review include the pumps and valves (pressure boundary only), heat exchangers, cyclone separators, piping, tubing, fittings, orifices, and spray nozzles. The intended functions for containment spray components subject to an aging management review include pressure boundary integrity, spray, throttling, filtration, and heat transfer. A complete list of containment spray components requiring an aging management review and the component intended functions is provided in Table 3.3-2 of Section 3.3 of the LRA. The aging management review for containment spray is discussed in Section 3.3 of the LRA.

#### 3.9.5.2 Staff Evaluation

As identified in Table 3.3-2, of the LRA, the containment spray system piping inspection program is credited for aging management of selected valves, piping and fittings in containment spray. The applicant has identified loss of material to be the aging effect requiring management for the stainless steel pressure boundary components in a treated water environment.

The staff evaluation of the containment spray system piping inspection program focused on how the program manages the aging effect through the effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The corrective actions, confirmation process and administrative controls for license renewal are in accordance with the site controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to an aging management review. The staff's evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

*[Program Scope]* The applicant stated that the containment spray system piping inspection program manages the aging effect of loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel piping/fittings and valves wetted by boric acid in the containment spray headers. In RAI 3.9.5-4, the staff requested the applicant to discuss the differences in design, construction or operation of this system at Turkey Point that explain why the scope of their program is limited to loss of material for carbon steel components.

In its response the applicant stated that for austenitic stainless steels in treated water, the relevant conditions required for stress corrosion cracking (SCC) are the presence of halogens in excess of 150 ppb or sulfates in excess of 100 ppb, and elevated temperature. For Turkey Point treated water environments, a temperature criterion of greater than 140 °F is utilized for susceptibility of austenitic stainless steels to SCC. Containment spray (CS) operates at a temperature less than 140 °F. Therefore, cracking due to SCC is not an aging effect requiring management for CS components. This conclusion is supported by plant operating and maintenance experience.

NRC Bulletin 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants," Information Notice 79-19, "Pipe Cracks in Stagnant Water Systems at PWR Plants," and IE Circular 76-06, "Stress Corrosion Cracks in Stagnant, Low-Pressure Stainless Piping Containing Boric Acid Solution at PWRs," describe several instances of throughwall cracking in stainless steel piping in stagnant borated water systems. NRC Bulletin 79-17 required licensees to review safety-related systems that contain stagnant, oxygenated, borated water. For these identified systems, licensees were requested to review preservice NDE, inservice NDE results, and chemistry controls. Also, ultrasonic and visual examinations of representative samples of circumferential welds were performed. The results of these reviews and inspections for Turkey Point, which included the containment spray system, identified no anomalies in chemistry or indications of SCC at welds. All of the instances of SCC in the nuclear industry have identified the presence of halogens, such as chlorides in the failed component. These occurrences most likely resulted from the inadvertent introduction of contaminants into the system. SCC can occur in stainless steel at ambient temperature if exposed to a harsh environment (i.e., with significant contamination). However, these conditions are considered to be event-driven, resulting from a breakdown of quality controls for water chemistry. Based on the above discussion, cracking due to SCC was determined not to be an aging effect requiring management for containment spray. The staff concurs with the applicant and considers the RAI issue resolved. The staff finds the overall scope of the program acceptable.

*[Preventive Actions]* The applicant states that as a preventive action, the surveillance procedures require the closure of a second isolation valve in the containment spray headers when the pumps are started for testing. In RAI 3.9.5-1 the staff requested the applicant to clarify the effectiveness of the preventive action. In its response the applicant stated that the

containment spray pumps surveillance testing procedures require closure of the second isolation valve in the containment spray headers. This preventive measure minimizes the possibility of water entering the spray headers, however, it is not credited for managing any aging effect. The aging management review assumed that the isolation valves leak and that the containment spray header is exposed to a borated water environment. The staff is satisfied with this response because the applicant's action is based on a conservative assumption and the RAI issue is considered resolved. Therefore, the staff finds the applicant's prevention actions adequate and acceptable.

*[Parameters Monitored or Inspected]* The applicant stated that the program monitors the wall thickness of selected piping/fittings in the spray headers within the containments. The staff finds the parameters identified for monitoring will permit timely detection of aging effects and are therefore acceptable.

*[Detection of Aging Effects]* The applicant stated that ultrasonic thickness measurement is utilized for this examination. The aging effect of concern, loss of material would be evident by the reduced wall thickness in the piping/fittings being examined. The staff concurs with the applicant's determination that the ultrasonic thickness measurements are effective in detecting the aging effects and, therefore, finds the detection method acceptable.

*[Monitoring and Trending]* The applicant stated that the examination is initially performed each refueling outage. The piping/fittings thickness measurements permit calculation of a corrosion rate. Inspection frequency may be adjusted based on corrosion rate to ensure that minimum wall thickness requirements for the pipe are maintained. If evaluation of the inspection results indicates that loss of material due to corrosion is not occurring, the containment spray system piping inspection program may be discontinued. Also, in RAI 3.9.5-3 the staff requested that the applicant describe the methods for monitoring and evaluating the aging effects for piping/fitting joints that may be inaccessible. In its response, the applicant stated that all piping/fittings required to be examined are accessible to perform ultrasonic thickness measurements. The RAI issue is, therefore, considered resolved. The staff finds the applicant's methodology will be effective in monitoring and trending the aging effects and is therefore acceptable.

*[Acceptance Criteria]* The applicant stated that the wall thickness measurements greater than minimum wall thickness values for the component design of record are acceptable. Wall thickness measurements less than the minimum required values are entered into the corrective action program. The staff finds this acceptable.

In RAI 3.9.5-2, the staff requested the applicant to indicate whether or not the required minimum wall thickness of the piping/fittings and valves has been evaluated to withstand damage due to fatigue resulting from flow-induced vibrations. In its response, the applicant stated that flow-induced vibration is not a design consideration for the containment spray system because the fluid flowing through the system is water (single phase) and there is no flow geometry (e.g., cross flow through tubes, etc.) that would induce flow vibrations. The minimum wall thickness is based on design pressure, dead weight, thermal, and seismic loads in accordance with the requirements of ANSI B31.1. The staff finds the response reasonable and acceptable. The RAI issue is, therefore, considered resolved. The staff finds the acceptance criteria for evaluating component damage will be able to determine the progress of damage due to aging effects and specify the time when appropriate corrective action needs to be taken and are therefore acceptable.

*[Operating Experience and Demonstration]* Ultrasonic thickness measurements have been performed for several years. The technique has proven to be successful at determining the wall thickness of piping/fittings and other components.

This is an existing program at Turkey Point that uses a technique with demonstrated capability and a proven industry record to measure wall thickness. This examination is performed utilizing approved procedures and qualified personnel. The ultrasonic thickness measurement technique has been previously used to measure the wall thickness in the containment spray system spray headers and other plant systems. The results of these examinations have detected some areas of localized corrosion in the headers, and the results are documented.

Based on the operating experience at Turkey Point, the staff considers the continued implementation of the containment spray system piping inspection program provides reasonable assurance that loss of material will be managed such that components from the containment spray system piping and therefore is acceptable.

#### 3.9.5.3 FSAR Supplements

The staff has reviewed the UFSAR Section 16.2.5 and has confirmed that it contains the appropriate elements of the program.

#### 3.9.5.4 Conclusion

In conclusion, based on the information discussed above, the staff finds the implementation of the containment spray system piping inspection program will provide reasonable assurance that loss of material will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the CLB throughout the period of extended operation.

### 3.9.6 Environmental Qualification Program

#### 3.9.6.1 Summary of Technical Information in the Application

The environmental qualification program is created for ensuring the qualified life of electrical and I&C components within the scope of 10 CFR 50.49. The thermal, radiation, and wear cycle aging analyses of plant electrical and I&C components required to meet 10 CFR 50.49 have been identified as time-limited aging analyses for Turkey Point, Units 3 and 4.

### 3.9.6.2 Staff Evaluation

The staff reviewed the EQ program to determine whether it will ensure that the electrical and I&C components covered under this program will continue to perform their intended function consistent with the current licensing basis for the period of extended operation. The staff's evaluation of the component qualification focused on how the program manages the aging effect through effective incorporation of the following 10 elements: program scope, preventive action, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The LRA indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site controlled corrective actions program pursuant to Appendix B to 10 CFR Part 50, and cover all components that are subject to AMR. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

*[Program Scope]* The scope of the program includes the environmentally qualified devices that are within the scope of 10 CFR 50.49 including 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
- the necessary post-accident monitoring equipment

The staff considers the scope of the program acceptable.

*[Preventive Actions]* The program includes preventive actions required to maintain the qualification time period for the environmentally qualified devices.

*[Parameter Monitored or Inspected]* The program establishes an aging limit (qualified life) for each installed device. The installed life of each device is monitored, and appropriate actions (replacement, refurbishment, or requalified) are taken before the aging limit is exceeded. The staff considers this monitoring appropriate because the program objective is to ensure the qualified life of devices established is not exceeded.

*[Detection of Aging Effects]* The program does not require the detection of aging effects for equipment while in service since effects are maintained within established acceptable limits by the EQ program actions. When the qualified life is less than the plant license period, the program requires replacement, refurbishment, or requalification of the component prior to the end of its qualified life. When unexpected adverse effects are identified during operation or maintenance activities, they are evaluated to determine the root cause and significance in accordance with the approved procedures. The staff considers the applicant's program to replace, refurbish, or requalify the component prior to the end of its qualified life acceptable.

*[Monitoring and Trending]* The installed life of each environmentally qualified device is monitored, and appropriate actions (replacement, refurbishment, or requalified) are taken before the aging limit is exceeded. The program does not require monitoring or trending of condition or performance parameters of equipment while in service to manage the effects of aging. The staff considers this is acceptable since 10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of inservice components to manage the effects of aging.

*[Acceptance Criteria]* The program requires replacement, refurbishment, or requalification before exceeding the life limit (qualified life) of each installed device. The staff considers this is acceptable since it is consistent with 10 CFR 50.49 requirements of refurbishment, replacement, or requalification before exceeding the qualified life of each installed device.

*[Operating Experience]* The EQ program is an ongoing program at Turkey Point that considers the best practices of industry organizations, vendors, and utilities. The program provides assurance that the environments to which installed devices are exposed will not exceed the qualified lives associated with the devices. This is accomplished through effective monitoring of key parameters (temperature, radiation) at established frequencies with well-defined acceptance criteria. The EQ program is governed by the quality control program to ensure accurate results.

The overall effectiveness of the EQ program is supported by the excellent operating experience for systems, structures, and components that are influenced by the program. No environmental qualification related degradation has resulted in loss of component intended functions on any systems. The program has been subject to periodic internal and external assessment activities that help to maintain highly effective control and facilitate continuous improvement. The staff finds that the applicant has adequately addressed operating experience.

#### 3.9.6.3 FSAR Supplement

The summary description of the EQ program provided in Section 16.2.6 of Appendix A to the LRA is sufficient.

#### 3.9.6.4 Conclusion

The applicant stated that its EQ program is an effective program for managing the effects of aging to ensure that the components within the scope of license renewal will continue to perform their intended function consistent with the current licensing basis for the period of extended operation. The staff considers the applicant's program which monitors key parameters (temperature and radiation) at established frequencies with well-defined acceptance criteria, provides assurance that the environments to which installed devices are exposed will not exceed the qualified lives associated the devices. Thus, the equipment will continue to perform its intended function consistent with the CLB throughout the period of extended operation.

The staff concludes that the EQ program will adequately manage the qualified life of components for the period of extended operation as required by 10 CFR 54.21(a)(3).

### 3.9.7 Fatigue Monitoring Program

#### 3.9.7.1 Summary of Technical Information in the Application

In Section 3.2.7 of Appendix B to the LRA, the applicant describes an existing aging management program, the FMP, that is designed to track cyclic and transient occurrences to ensure that reactor coolant pressure boundary components remain within ASME Code Section III fatigue limits. The applicant refers to the FMP as a confirmatory program (rather than an actual aging management program) because the program only monitors the number of significant plant transients to ensure that number of transients assumed in the design fatigue analyses are not exceeded.

#### 3.9.7.2 Staff Evaluation

The staff reviewed the FMP to determine whether it will ensure that the fatigue design limits are not exceeded during the period of extended operation. The staff's evaluation of the component cyclic and transient limit program focused on how the program manages the aging effect through effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The LRA indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled corrective actions program pursuant to Appendix B to 10 CFR Part 50, and cover all structures and components that are subject to aging management review. The staff evaluation of the applicant's corrective actions program is provided separately in Section 3.1.2 of this safety evaluation report. The corrective actions program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

*[Program Scope]* The scope of the program includes the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines. The program tracks the number of design cycles to ensure that these components remain within their design limits. The staff considers the scope of the program, which includes the RCS components qualified in accordance with ASME Code fatigue analyses acceptable.

*[Preventive and Mitigative Actions]* The applicant identified the cycle counting procedure as the preventative action for this program. The staff considers counting of design cycles to be an acceptable preventive action.

*[Parameters Inspected or Monitored]* The parameters monitored are the cycles of design transients used in the Class 1 design analyses. The staff considers this monitoring appropriate because the program objective is to ensure the number of cycles assumed in the design analyses are not exceeded.

*[Detection of Aging Effects]* The program monitors design transients used in the fatigue analysis of components and the information is used to ensure that the fatigue design limits are not exceeded. This provides assurance that the fatigue analyses of record remain valid during the period of extended operation. The staff considers this monitoring appropriate.

*[Monitoring and Trending]* The applicant uses administrative procedures for logging design cycles. As stated previously, the program monitors the design transients used in the fatigue analysis of the components to ensure that the fatigue analyses of record remain valid during the period of extended operation. The staff finds this program element acceptable.

*[Acceptance Criteria]* The applicant specifies the maximum number of design cycles in the plant administrative procedures. The applicant indicated that the plant procedures require administrative action should the actual cycle count reach 80% of any design cycle limit. The staff considers this criterion acceptable.

*[Operating Experience]* The applicant's program involves tracking transients used in the design of these components. The applicant indicates that an independent assessment of the program was performed. According to the applicant the assessment concluded that the administrative procedure accurately identifies and classifies plant design cycles. The staff finds that the applicant has adequately addressed operating experience.

#### 3.9.7.3 FSAR Supplement

The summary description of the FMP provided in Section 16.2.7 of Appendix A to the LRA is sufficient.

#### 3.9.7.4 Conclusion

The applicant references the FMP in its discussion of the fatigue TLAAs as a confirmatory program to ensure that design fatigue limits are not exceeded during the period of extended operation. The staff considers the applicant's program, which monitors the number of plant transients that were assumed in the fatigue design an acceptable method to manage the fatigue usage of the RCS components within the scope of the program.

The staff concludes that the FMP will adequately manage thermal fatigue of RCS components for the period of extended operation as required by 10 CFR 54.21(a)(3).

### 3.9.8 Fire Protection Program

#### 3.9.8.1 Summary of the Technical Information in the Application

The fire protection program is designed to protect plant equipment in the event of a fire, to ensure safe plant shutdown, and minimize the risk of a radioactive release to the environment. The program relies on fire water supply including sprinklers, Halon suppression, fire dampers, RCP oil collection, alternate shutdown, safe shutdown, and fire detection and protection. Individual components that constitute alternate shutdown and safe shutdown were screened with their respective systems. The screening for fire detection and protection electrical and

Instrumentation and Controls is discussed in Section 2.5 of the LRA. Fire protection is described in UFSAR Appendix 9.6A. The majority of fire protection is common to Units 3 and 4.

The flow diagrams listed in Table 2.3-5 of the LRA show the evaluation boundaries for the portions of fire protection that are within the scope of license renewal.

Fire protection is in the scope of license renewal because it contains structures and components that are safety-related and are relied upon to remain functional during and following design-basis events, structures and components that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions, and structures and components that are relied on during postulated fires.

Fire protection components that are subject to an aging management review include the raw water tanks, pumps and valves (pressure boundary only), tanks, heat exchangers, hose stations, flame arrestors, sprinklers, strainers, orifices, piping, tubing, and fittings. The intended functions for Fire protection components that are subject to an aging management review are pressure boundary integrity, heat transfer, filtration, throttling, fire spread prevention, and spray. A complete list of the fire protection components that require aging management review and the component intended functions, appears in Tables 3.4-14 and 3.6-12 of the Application. The aging management reviews for fire protection are discussed in Sections 3.4 and 3.6.2 of the LRA. Fire extinguishers, fire hoses, and air packs are not subject to an aging management review because they are replaced based on conditions in accordance with National Fire Protection Association (NFPA) standards and plant surveillance procedures for fire protection equipment. This position is consistent with the NRC staff's guidance on consumables provided in the NRC's letter to the applicant dated March 10, 2000.

The Fire Protection Program manages the aging effects of loss of material, cracking, and fouling for the components/piping of the fire Protection System and Fire Rated Assemblies. Additionally, this program manages the aging effects of loss of material, loss of seal, cracking, and erosion for structures and structural components associated with fire protection. Appendix 9.6A contains a detailed discussion of the Fire Protection Program.

As stated earlier, the scope of the Fire Protection Program will be enhanced to include inspection of additional components prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4.

#### 3.9.8.2 Staff Evaluation

As identified in Table 3.4-14 of the LRA, the fire protection program is credited for aging management of specific component/commodity groups associated with the fire protection and fire rated assemblies.

The following specific component/commodity groups are identified:

- carbon steel raw water tanks in air/gas, outdoor, and raw water environments with an aging effect requiring management of loss of material

- cast iron electric and diesel fire pumps and heat exchanger shell in treated water, indoor-not air-conditioned, and outdoor environments with aging effects requiring management of loss of material and fouling

- copper alloy diesel fire pump heat exchanger tubes and cover in a raw water environment with aging effects requiring management of loss of material and fouling

- cast iron basket strainers in raw water and outdoor environments with an aging effect requiring management of loss of material

- carbon steel, stainless steel, cast iron, and copper alloy valves, piping, tubing, fittings, sprinklers, flexible hoses, flame arrestors, and flow restriction orifices in air/gas, raw water, and outdoor environments with an aging effect requiring management of loss of material

- rubber expansion joints in an indoor-not air-conditioned environment with an aging effect requiring management of cracking

The staff's evaluation of the fire protection program focused on how the program manages the aging effect through the effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored of inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components that are subject to an aging management review. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this safety evaluation. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

[*Program Scope]* The fire protection program manages the aging effects of loss of material, cracking, and fouling for the components/piping of the Fire Protection System and Fire Rated Assemblies. Additionally, this program manages the aging effects of loss of material, loss of seal, cracking, and erosion for structures and structural components associated with fire protection.

The applicant states that the scope of the fire protection program will be enhanced to include inspection of additional components. Commitment dates associated with the enhancement of this program are contained in Appendix A to the LRA.

In RAI 3.9.8-1, the staff requested the applicant to provide the basis and guidelines which are to be used for the selection of the additional components in the enhanced program. In its response, the applicant stated that cracking of rubber, neoprene, or coated canvas materials due to embrittlement is an aging effect evaluated in the aging management review process. The aging management review of fire protection components identified rubber expansion joints on the suction and discharge of the diesel fire pump. As a result, the fire protection program, described in Section 3.2.8 of Appendix B to the LRA will be enhanced to include inspection of the rubber expansion joints on the suction and discharge of the diesel fire pump engine piping for evidence of cracking or drying. All other components subjected to aging effects requiring management under the fire protection program are currently included within the scope of this program. The staff finds the applicant's response reasonable and acceptable. The RAI issue is, therefore, considered resolved. With the resolution of the staff's concerns as discussed above, the staff finds the scope of the fire protection program adequate and acceptable.

*[Preventive Actions]* The applicant states that many fire protection components are provided with a protective coating to minimize the potential for external corrosion. Coating minimizes corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management. This is acceptable to the staff because coatings provide an added measure of protection.

*[Parameters Monitored or Inspected]* The applicant states that surface conditions are monitored visually to determine the extent of external material degradation. Visual examination will detect loss of material due to general, crevice, and pitting corrosion, as well as loss of seal or cracking due to embrittlement. Internal conditions are monitored via leakage, flow, and pressure testing. Internal loss of material (due to general, crevice, and pitting corrosion; microbiologically influenced corrosion; and selective leaching) and blockage due to fouling can be detected by changes in flow or pressure, leakage, or evidence of excessive corrosion products during flushing of the system. The staff finds that the parameters monitored will permit timely detection of the aging effects and are therefore acceptable.

*[Detection of Aging Effects]* The applicant stated that detection of degradation on external surfaces is determined by visual examination. Surfaces of components and structures are examined for damage, deterioration, leakage, or other forms of corrosion.

Functional testing and flushing of the system clears away internal scale, debris, and other foreign material that could lead to blockage/obstruction of the system. Flow and pressure tests verify system integrity. Visual examination of breached portions of the system also verifies unobstructed flow and integrity of the piping/components.

In RAI 3.9.8-2, the staff requested the applicant to identify the specific programs which are credited for monitoring external and internal material degradation of the fire protection system components and piping. In its response, the applicant identified the programs relevant to the fire protection system. Based on its review, the staff finds that the applicant's response is satisfactory, and the issue is considered resolved. In addition, the staff requested the applicant at a meeting held on April 12, 2001, to provide clarification regarding the inspection and testing of sprinkler systems. In its response the applicant stated that per UFSAR Appendix 9.6A, Turkey Point's current licensing basis does not include National Fire Protection Association (NFPA) 25 for testing and inspection of sprinkler heads. However, Turkey Point generally conforms to NFPA guidelines, and many tests and inspections are performed in accordance with NFPA.

Turkey Point uses potable city water (potable) as its water source for fire protection. This water was conservatively classified as "raw water" for the purpose of performing aging management reviews even though it is clean and free of contaminants compared to lake or river water used in fire protection systems at other plants. The quality of the water minimizes loss of material, as evidenced by Turkey Point's operating and maintenance experience. As identified in the above list of fire protection procedures, a fire protection system annual flush is credited for ensuring the system is clear of scale, debris and foreign material.

For closed head sprinkler systems, inspections and testing are performed on an 18-month interval, in accordance with the "Spray Sprinkler System Inspection." This procedure verifies the systems are in a state of readiness by ensuring proper operation of clapper/inlet valves, all nozzles are unobstructed, and water and supervisory air pressure are within specifications.

Testing of open head sprinkler systems is done by the "Open Head Spray/Sprinkler 3-Year Air Flow Test." This procedure requires connection of service air to the dry pipe and verification of flow path by the discharge of air at the opening of each sprinkler head/spray nozzle to ensure system functionality. Additionally, each spray nozzle is also visually inspected for obstruction.

Based on feedback from the NRC staff during a meeting on April 12, 2001, the applicant proposed to perform testing of wet pipe sprinkler heads following the guidance of NFPA commencing in the year 2022 (50 years from the issuance of the original operating license on Unit 3). This enhancement will be included with the fire protection program enhancements described in Appendix A, Section 16.2.8 (page A-37), and Appendix B, Section 3.2.8 (page B-56). The staff finds that these inspections and tests will provide a satisfactory means for detecting the aging effects in fire protection system components. Therefore, the staff finds the applicant's detection methods acceptable.

*[Monitoring and Trending]* The degradation found as a result of inspection/testing of the systems/components is addressed by the fire protection program procedures. The evaluation of the inspection/testing results may result in additional testing, monitoring, and trending. The staff finds this methodology will provide effective monitoring are trending of the aging effects and is therefore acceptable.

*[Acceptance Criteria]* The results of the inspection/testing will be evaluated in accordance with the acceptance criteria in the appropriate fire protection procedure(s). Parameters required to be monitored and controlled are listed in the applicable documents.

In RAI 3.9.8-3, the staff requested the applicant to identify the specific fire protection procedures which specify the acceptance criteria for evaluating the inspection and test results of the components/piping. Also, the applicant was requested to identify the applicable documents which list the parameters required to be monitored and controlled. In its response, the applicant identified the relevant procedures which, contain the parameter required to be monitored or controlled. The staff finds the applicant's response satisfactory and acceptable. The RAI issue is therefore closed. With the resolution of the staff's concerns, the staff finds the acceptance criteria adequate and acceptable.

*[Operating Experience and Demonstration]* The Fire Protection Program has been an ongoing program at Turkey Point. The program was enhanced by implementation of 10 CFR Part 50, Appendix R, and has evolved over many years of plant operation. The program incorporates the best practices recommended by NFPA and Nuclear Electric Insurance Limited (NEIL) and is approved by the NRC.

The overall effectiveness of the program is demonstrated by the excellent operating experience of systems, structures, and components that are included in the Fire Protection Program. The applicant states that the program has been subjected to periodic internal assessment activities. These activities, as well as other external assessments, help to maintain highly effective fire protection control and facilitate continuous improvement through monitoring industry initiatives and trends in the area of aging control. The staff finds that, based on the operating experience, the applicant will effectively maintain a Fire Protection Program during the extended period of operation.

#### 3.9.8.3 FSAR Supplements

The staff has reviewed the UFSAR Section 16.2.8 of Appendix A to the LRA and has confirmed that it contains the appropriate elements of the program.

#### 3.9.8.4 Conclusion

On the basis of its review as discussed above, the staff concludes that the continued implementation of the fire protection program by the applicant provides reasonable assurance that the aging effects (loss of seal, loss of material, cracking, and fouling) will be managed such that components/commodity groups within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### 3.9.9 Flow-Accelerated Corrosion Program

The applicant described its flow-accelerated corrosion (FAC) program in Section 3.2.9, "Flow-Accelerated Corrosion Program," of Appendix B to the LRA. The LRA also included relevant material from Section 3.5 of the LRA. These sections address aging effects of the components in the feedwater and blowdown system and the main steam and turbine generators system. The objective of the FAC program is to manage the aging effects caused by FAC. It is accomplished by controlling the environment to which the affected components are exposed, predicting the degradation of these components by FAC and taking corrective actions once degradation has been identified.

The staff reviewed the applicant's description of the program in Section 3.2.9 of Appendix B of the LRA and relevant material in the other referenced section of the LRA to determine whether the applicant has demonstrated that the program will adequately manage the effects of aging caused by FAC in the plant during the period of extended operation as required by 10 CFR 54.21(a)(3).

#### 3.9.9.1 Summary of Technical Information in the Application

In the LRA the applicant has identified the following systems which contain the components that are subjected to FAC:

- main steam and turbine generator
- feedwater and blowdown

The applicant has identified loss of material by FAC as an aging effect for carbon steel components exposed to secondary water (treated water-secondary). These components, when exposed to the environment of moving single or two-phase water with low pH, low oxygen content and relatively high temperature, corrode at higher rates than if they were in contact with a stagnant fluid. The resulting loss of material produces thinning of walls in the affected components. In order to prevent their failure, the aging effect due to FAC has to be managed. The staff finds that there is reasonable assurance that this mode of degradation is the only plausible aging effect for aging management considerations.

The applicant developed a methodology for addressing the FAC issue. The applicant's methodology was based on EPRI recommendations specified in report NSAC-202L-R2, "Recommendations for Effective Flow-Accelerated Corrosion Program." The licensee concluded that it will ensure proper management of aging effects in the components subjected to FAC and will allow them to perform their intended functions consistent with the CLB, during the period of extended operation.

3.9.9.2 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 FAC program will ensure that the effects of aging due to FAC will be adequately managed so that intended functions will be maintained consistent with CLB throughout the period of extended operation for all affected components in the systems included in the LRA. After completing the initial review, by letter dated February 1, 2001, the staff issued several requests for additional information (RAIs). By letter dated April 19, 2001, the applicant responded to the staff's RAIs.

The staff's evaluation of the applicant's AMPs related to FAC focused on program elements rather than detailed plant-specific procedures. To determine whether these program elements adequately mitigate the effects of aging to maintain the intended functions consistent with the CLB throughout the period of extended operation, the staff evaluated seven elements applicable to these programs. The corrective actions, confirmation process and administrative controls for license renewal were not discussed in this section because the applicant indicated that they are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff's evaluation of the quality assurance program is provided separately in Section 3.1.2 of this SER. The remaining seven elements are evaluated below.

*[Program Scope]* The applicant stated in Section 3.2.9 of Appendix B of the LRA that the scope of this program includes managing the aging effects caused by a loss of material by FAC from the components in the systems specified in Section 3.5 of the LRA. The program predicts, detects, monitors and mitigates FAC in high energy carbon steel piping associated with the main steam and turbine generator and feedwater and blowdown systems. It includes determination of the extent of wall thinning in these components and their repair or replacement when wall thickness reaches predetermined minimum thickness. In the future, the program will be enhanced to address loss of material from steam trap lines. The staff finds this scope adequate because it will detect and manage the aging effects in the components subjected to FAC.

*[Preventive or Mitigative Actions]* The magnitude of FAC depends on the geometry, hydrodynamic characteristics, and water chemistry of the system. The first two attributes cannot be controlled, but water chemistry can be controlled by the chemistry control program. High pH and oxidizing environment will minimize FAC. However, an oxidizing environment may

be undesirable for controlling other corrosion mechanisms. This method of controlling FAC has, therefore, limited application. Another effective preventive action against component failures by FAC is early detection and timely repair or replacement of the damaged components. The staff finds that predicting and measuring wall thickness, repairing and replacing damaged components, and, to some extent, controlling water chemistry, will effectively mitigate aging effects due to FAC.

*[Parameters Monitored or Inspected]* The program monitors the effects caused by FAC by measuring wall thickness of the components subjected to FAC. The EPRI-developed analytical model, CHECWORKS, is used to predict FAC in piping systems on the basis of plant-specific data, including material of construction, chemistry, hydrodynamics, and operating conditions. Subsequently, the components suspected to be damaged by FAC are examined by the NDE methods and their wall thickness determined. The staff finds that the parameters monitored will permit timely detection of aging effects in the components exposed to FAC.

*[Detection of Aging Effects]* Wall thickness is measured by UT examination and by radiography, as specified in EPRI NSAC-202L, which are standard, well-developed, NDE techniques that produce reliable results. The staff finds that determination of wall thickness by these techniques will provide satisfactory means for detecting aging effects in the components exposed to FAC.

*[Monitoring and Trending]* Using the predictive and inspection methods, the applicant will be able to detect, monitor and trend the magnitude of component wall thinning by FAC. If degradation is detected such that the wall thickness is less than the minimum allowed by the acceptance criteria, the component will be repaired or replaced and additional examinations will be performed of the components in adjacent areas to bound the damaged component. The staff finds this methodology will provide effective monitoring and trending of aging effects caused by FAC. The program will also include monitoring and trending of material loss by general corrosion of external surfaces of the components in the steam traps. This monitoring will be performed simultaneously with the monitoring of FAC in these components.

*[Acceptance Criteria]* The criterion for component replacement is based on the allowable minimum wall thickness for a given component specified in the ANSI B31.1 code. Inspections and analytical methods monitor and trend wall thickness. If it is predicted that the component will reach its minimum allowable wall thickness before the next inspection interval, the component is repaired, replaced, or acceptability to perform its function reevaluated. The staff finds that the criteria used for evaluating component damage will be able to determine the progress of FAC damage and specify the time when the appropriate corrective actions have to be taken.

*[Operating Experience]* The applicant has implemented the FAC program in response to NRC GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning." The program applies to components subjected to FAC in the main steam and turbine generator system and the feedwater and blowdown system, containing single and two-phase fluids. These components were periodically examined by NDE methods and those which did not continue to meet the design criteria were either repaired or replaced by the components made from the same materials or from material more resistant to FAC. In the past, there has been a small number of components replaced due to FAC damage in the portions of main steam and turbine generator and feedwater and blowdown systems in the scope of the LRA. They included the

nozzle, elbow, and expander at the discharge from the feedwater pumps, the expanders/reducers associated with the feedwater regulating valves, and the pipe segment in the feedwater line in containment. The applicant stated that all damaged components were repaired or replaced in time and there were no cases of component inservice failure. The program was, therefore, successful in managing loss of material by the components exposed to FAC. The staff finds this approach acceptable.

#### 3.9.9.3 FSAR Supplement

The summary description of the FAC program provided in Section 16.2.9 of Appendix A to the LRA is sufficient.

#### 3.9.9.4 Conclusions

The staff has reviewed the information in Section 3.2.9 of Appendix B of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated there is reasonable assurance that the FAC program will adequately manage aging effects caused by FAC in accordance with the CLB throughout the period of extended operation.

### 3.9.10 Intake Cooling Water System Inspection Program

#### 3.9.10.1 Summary of the Technical Information in the Application

The Intake cooling water removes heat from component cooling water and turbine plant cooling water. The intake cooling water pumps supply salt water from the plant's intake area through two redundant piping headers to the tube side of the component cooling water and turbine plant cooling water heat exchangers. Flow is routed from the heat exchangers to the plant discharge canal. Intake cooling water is described in UFSAR Section 9.6.2.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of intake cooling water that are within the scope of license renewal. The component cooling water heat exchangers were considered to be part of component cooling Water and were screened with that system.

Intake cooling water is in the scope of license renewal because it contains structures and components that are safety-related, and are relied upon to remain functional during and following design-basis events. The scope also includes structures and components that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions, and structures and components that are relied on during postulated fires and station blackout events.

Intake cooling water components that are subject to an aging management review include pumps and valves (pressure boundary only), strainers, orifices, piping, tubing, and fittings. The intended functions for intake cooling water components that are subject to an aging management review are pressure boundary integrity, filtration, structural integrity, structural support, and throttling. A complete list of intake cooling water components that require aging management review and the component intended functions appears in Table 3.4-1 of the LRA. The aging management review for intake cooling water is discussed in Section 3.4 of the LRA.

The Intake Cooling Water System Inspection Program manages the aging effects of loss of material due to various corrosion mechanisms, stress corrosion cracking, and biological fouling for Intake Cooling Water System components. The program includes inspections, performance testing, evaluations, and corrective actions that are performed as the result of the applicant commitments to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

This program will be enhanced to improve documentation of scope and frequency of the intake cooling water piping crawl-through inspections and component cooling water heat exchanger tube integrity inspections prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4.

3.9.10.2 Staff Evaluation

As identified in Table 3.4-1, Chapter 3 of the Application, the Intake Cooling Water System Inspection Program is credited for aging management of specific component commodity groups in the Component Cooling Water and Intake Cooling Water systems. The specific component/commodity groups identified are the following:

- carbon steel basket strainers (shell) in a raw water environment with an aging effect requiring management of loss of material
- stainless steel basket strainers (screens) in a raw water environment with an aging effect requiring management of loss of material
- cast iron valves, piping/fittings in the main lines upstream of the strainers in a raw water environment with an aging effect requiring management of loss of material
- bronze valves (CCW heat exchange vents and drains) welded to the CCW heat exchanger channels in a raw water environment with an aging effect requiring management of loss of material
- copper-nickel piping/fittings (vents and drains) welded to the CCW heat exchanger channels in a raw water environment with an aging effect requiring management of loss of material
- copper-nickel CCW heat exchanger tube sheets in a raw water environment with an aging effect requiring management of loss of material
- aluminum-brass CCW heat exchanger tubes in a raw water environment with aging effects requiring management of loss of material and fouling

The staff evaluation of the intake cooling water system inspection program focused on how the program manages the aging effect through the effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

The corrective actions, confirmation process and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to an aging management review. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this safety evaluation. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

*[Program Scope]* NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," recommended the implementation of an ongoing program of surveillance and control techniques to significantly reduce flow blockage caused by biofouling, corrosion, erosion, protective coating failures, stress corrosion cracking, and silting problems in systems and components supplied by the intake cooling water system. The intake cooling water system inspection program was developed in response to this generic letter and addresses the aging effects of loss of material due to various corrosion mechanisms, stress corrosion cracking, and fouling due to macro-organisms for those components subject to raw water (i.e., salt water) conditions. The program utilizes performance testing and evaluations, systematic inspections, leakage evaluations, and corrective actions to ensure that loss of material, cracking, or biological fouling do not lead to loss of component intended functions.

The staff finds the program scope in conformance with Generic Letter 89-13, and is therefore acceptable.

*[Preventive Actions]* The applicant stated that the intake cooling water system inspection program is preventive in nature, since it provides for the periodic inspection and maintenance of internal linings protecting the intake cooling water heat exchanger, performance monitoring, testing, and periodic tube inspections. Maintenance of the internal piping/component linings minimizes the potential loss of material due to corrosion that could impact the pressure boundary intended function. Performance monitoring and testing; channel head, tube sheet, and anode inspections; and tube examinations of component cooling water heat exchangers provide for early identification of internal fouling and tube degradation that could impact heat transfer and pressure boundary intended functions. External coatings are applied to portions of the intake cooling water system to minimize corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management. This is acceptable to the staff because coatings provide additional protection beyond the other preventive actions stated above. The staff therefore finds the applicant's preventive actions acceptable.

*[Parameters Monitored or Inspected]* The applicant stated that during inspections of the internal piping component, surface conditions of piping/components and their internal linings are visually inspected for degradation. Wall thickness measurements are taken when deemed necessary.

During performance monitoring, testing, and tube inspections of component cooling water heat exchangers, the applicant indicated that pressures, temperatures, and flows are measured as part of periodic performance testing of the component cooling water heat exchangers to verify heat transfer capability. This testing is supplemented by routine monitoring of differential temperatures across the heat exchanger during operation. Tube integrity of the component cooling water heat exchangers is monitored by periodic nondestructive examination (e.g., eddy current testing) to ensure detection of aging effects. This is acceptable to the staff because the parameters proposed to be monitored are considered adequate to manage the aging effects.

*[Detection of Aging Effects]* The applicant stated that during inspections of internal piping component, visual examination of the piping/components and their internal linings is performed. Additional nondestructive testing may be utilized to measure surface condition, and the extent of wall thinning based on the evaluation of the examination results is documented in accordance with the corrective action program. The staff finds this acceptable because these methods have been proven to be effective in detecting aging effects.

*[Monitoring and Trending]* The applicant states that inspections of the internal piping/components and frequencies are in accordance with commitments under Generic Letter 89-13. Internal piping/component inspections are performed periodically during refueling outages. Inspection frequencies are adjusted based upon experience and ensure the timely detection of aging effects.

During the performance monitoring, testing, and tube inspections of component cooling water heat exchangers, online monitoring of system parameters is used to provide an indication of flow blockage. Heat transfer testing results are documented and reviewed in plant procedures. The heat transfer capability is trended to ensure that the component cooling water heat exchangers satisfy safety analysis requirements. Component cooling water heat exchanger tube condition is determined by eddy current testing and documented accordingly. Heat exchanger tube cleaning, tube replacements, or other corrective actions are implemented as required.

The staff finds that the proposed methodologies will provide effective monitoring and trending of aging effects and are therefore acceptable.

*[Acceptance Criteria]* Biological fouling is considered undesirable and is removed or reduced during the inspection process of the internal piping/components. When required by procedure, wall thickness values are determined and evaluated.

During the performance monitoring, testing, and tube inspections of component cooling water heat exchangers, acceptance criteria are provided to ensure that the design-basis heat transfer capability is maintained and to determine when component cooling water heat exchanger cleaning and inspection are required. Differential pressure criteria guidelines are provided to ensure that the intake cooling water design-basis flow rate is maintained and to identify when back flushing or cleaning of the intake cooling water basket strainers is required.

In RAI 3.9.10-2 the staff requested the applicant to identify the specific plant procedures and applicable documents which contain detailed guidance related to the performance monitoring, testing and tube examinations of the component cooling water system piping and heating exchangers. Also, the applicant was requested to provide the acceptance criteria and bases for the evaluation of the inspection results. In its response, the applicant identified the applicable procedures related to the monitoring, testing and inspection of the heat exchangers. In addition, acceptance criteria are provided to ensure that design-basis and Technical Specification requirements for heat transfer capability are maintained. Guidelines are provided for cleaning, inspecting, and testing the heat exchangers. The applicant's response is considered reasonable and acceptable to close the issue of this RAI. With the resolution of the staff's concerns as discussed above, the staff finds the acceptance criteria acceptable.

*[Operating Experience and Demonstration]* The applicant states that the existing intake cooling water system inspection program has been an ongoing formalized inspection program at Turkey Point. The program was formally implemented as a result of Generic Letter 89-13, which recommended monitoring of service water systems to ensure that they would perform their safety-related function and based on experiences of biological fouling and corrosion throughout the industry. The conservative philosophy established within the program has been successful in managing the loss of material due to corrosion and fouling of the component cooling water heat exchanger. This program has been effective in maintaining acceptable component cooling water heat exchanger performance and addressing biological fouling of strainers and heat exchangers. Various sections of the intake cooling water piping, basket strainers, and heat exchangers are periodically examined using nondestructive examination to determine the effects of corrosion and biological fouling. Results are evaluated and components are either repaired or replaced as required.

The program has been reviewed by the NRC during several inspections with no significant deviations or violations identified. FPL Quality Assurance surveillance and reviews have been performed with no significant deficiencies identified. Procedures and practices were enhanced as a result of the recommendations provided from these inspections.

Metallurgical analysis of component cooling water heat exchanger tubes removed in 1991 and 1994 indicated that stress corrosion cracking was a potential root cause and, as a result, zinc anodes were installed and are inspected during tube cleaning. Analysis in 1996 of additional component cooling water heat exchanger tubes indicated that inside pitting was a potential failure mechanism and, as a result, a less abrasive cleaning tool was recommended. Both of these corrective actions have proven to be effective in minimizing repetitive failures.

A review of the Maintenance Rule database by the applicant and staff for the Intake Cooling Water and the Component Cooling Water Systems shows that the current aging management programs have supported system availability above the required performance criteria for the period from May 1996 through March 2000. No components have failed during that period. Therefore, based on operating experience the staff finds the program acceptable.

#### 3.9.10.3 FSAR Supplements

The staff has reviewed UFSAR Section 16.2.10 of Appendix A to the LRA and has confirmed that it contains the essential elements of the program.

#### 3.9.10.4 Conclusion

On the basis of its review as discussed above, the staff finds that the continued implementation of the intake cooling water system inspection program provides reasonable assurance that the aging effects of corrosion and biological fouling will be managed, such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

### 3.9.11 Periodic Surveillance and Preventive Maintenance Program

The applicant described its periodic surveillance and preventive maintenance program in Section 3.2.11 of Appendix B of the LRA. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the aging effects on those program-specific systems and structures will be adequately managed by this program during the period of extended operation as required by 10 CFR 54.21(a)(3).

#### 3.9.11.1 Summary of Technical Information in Application

The applicant specified that the periodic surveillance and preventive maintenance program applies to component/commodity groups in certain designated systems and structures. The program is intended for managing the aging effects of loss of material, cracking, fouling, loss of seal, and embrittlement of systems and structures. Activities of the program consist of periodic visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/refurbishment during the performance of periodic surveillance and preventive maintenance activities. The program also includes leak inspections of limited portions of the chemical and volume control systems.

The applicant indicated that the periodic surveillance and preventive maintenance program is an established program and its effectiveness has been demonstrated by early detection of component surface defects for timely actions to ensure structural integrity, and concludes that the program is consistent with the CLB, and will remain effective during the period of extended operation.

#### 3.9.11.2 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 intended function will be maintained consistent with the CLB throughout the period of extended operation for systems and structures included in the program.

The periodic surveillance and preventive maintenance program is for managing the aging effects of loss of material, cracking, fouling, loss of seal, and embrittlement of component/commodity groups in certain specified systems and structures. The program activities include periodic visual inspections for evidence of surface defects, followed by replacement or refurbishment as needed by the results of the inspection activities and by industry experience. It is an established program and has been effective in the past. The staff concurs that its continued implementation will serve its intended purpose.

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff's evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven programs are evaluated below.

*[Program Scope]* As indicated in the LRA, the program applies to component/commodity groups in the following systems and structures: chemical and volume control, control building ventilation, emergency containment filtration, emergency diesel generators and support systems, fire protection, instrument air, intake cooling water, residual heat removal, turbine building ventilation, and waste disposal. The structures consist of auxiliary building, emergency diesel generator buildings, turbine building, and yard structures. The staff finds that relevant systems and structures are included in the scope of the program, and therefore, the scope is adequate.

The staff finds that in Appendix B, subsection 3.2.11, page B-67 of the LRA, yard structures are listed as one category of structures for which aging effects are managed by the periodic surveillance and preventive maintenance program. However, this program was not included in the last column of Table 3.6-20 which identifies specific programs and activities for aging management of yard structures. Per RAI 3.9.22-2, the staff requested the licensee to clarify this discrepancy, or make appropriate modifications either to Table 3.6-20 or in the scope of the periodic surveillance and preventive maintenance program. The licensee indicated in their response (dated April 19, 2001) that yard structures were inadvertently listed in page B-7 of the LRA, and the list of structures will be revised to remove yard structures from the scope of the periodic surveillance and preventive maintenance program. The staff finds this acceptable because the aging effects associated with the yard structures are managed by the systems and structures monitoring program, boric acid wastage surveillance program, and the ASME Section XI, IWF ISI program.

*[Preventive or Mitigative Actions]* There are no preventive or mitigative actions applicable to the aging effects being managed by this program. However, this program in its entirety satisfies this program element.

[Detection of Aging Effects] The aging effects concerning loss of material, cracking, fouling, loss of seal, and embrittlement, will be detected by visual inspection of external surfaces for evidence of corrosion, cracking, leakage, or coating damage. For some equipment, aging effects are addressed by periodic replacement in lieu of visual inspection and refurbishment. The staff finds that the techniques used to detect aging effects are consistent with accepted engineering practice.

As indicated in the scope description, the periodic surveillance and preventive maintenance program is credited for managing several aging effects including embrittlement of structures, systems, and components. However, the embrittlement effect to be managed by this program is not shown in tables related to Section 3.3, 3.4, and 3.6. In addition, given that aging effects are detected by visual inspections, acceptance criteria on how embrittlement effects are managed and detected should be provided. The licensee indicated in their response that cracking is the aging effect resulting from embrittlement and requiring management for coated canvas and rubber in environments such as treated water, raw water, air/gas, etc., and for components/commodity groups such as intake cooling water pumps expansion joints, containment cooling ductwork flexible connectors, emergency diesel generator (EDG) air intake and exhaust system flexible couplings, and EDG air start system flexible hose. The periodic surveillance and preventive maintenance program will conduct periodic visual inspection for replacement of items found cracked. The response also identified sections in the LRA where more descriptions on the subject are provided. On the basis of the visual inspection to be performed periodically on the specified structures, to detect cracks, the applicant's response is acceptable.

*[Parameters Monitored or Inspected]* Surface conditions of systems, structures, and components are monitored, through visual inspections, for corrosion, fouling or in some cases, leakage, during the performance of periodic maintenance. On the basis of inspection results, refurbishment is performed as required. For some equipment, periodic replacement is performed on a specified frequency. The staff finds that the process is logical and reasonable.

The applicant indicated that this program will be enhanced with regard to the scope of specific inspections and their documentation. As indicated in Section 16.2.11 of the UFSAR supplement in Appendix A to the LRA, specific enhancements to the scope and documentation of some inspections performed under this program will be implemented prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4. The staff had requested the licensee to provide a description of the program enhancements. In response, a list of enhancements to the periodic surveillance and preventive maintenance program was provided, which includes descriptions on several enhanced maintenance procedures and activities for additional inspection on various components and attributes. On the basis of the enhancements to identify potential degradation under the specified program enhancements, the staff finds that the descriptions are adequate and acceptable.

*[Monitoring and Trending]* System, structure, and component inspections are performed periodically during preventive maintenance or surveillance activities. Alternatively, some components are replaced on a specified frequency. Inspection and replacement frequencies are adjusted as necessary based on the results of these activities and industry experience. The staff finds that the process is reasonable. However, since this is an existing program, the applicant was requested to provide a brief description regarding how frequently the inspections were conducted, and components were replaced. In its response, the applicant indicated that the periodic surveillance and preventive maintenance program currently includes inspection frequencies ranging from two months to ten years depending upon the specific component and aging effect being managed and plant operating experience. Several examples of inspections that are part of this program and their current inspection frequencies were provided. Examples of some components that have 42 month replacement frequencies were also described. The applicant indicated that frequencies of replacement may be adjusted as necessary based on future plant-specific performance and/or industry experience. This is acceptable to the staff.

*[Acceptance Criteria]* Acceptance criteria and guidelines are provided in the applicant's implementing procedures for the inspections, refurbishments, and replacements, as applicable. The applicant was requested to provide a brief description of acceptance criteria and guidelines, and documentation on implementation procedures for the inspections, refurbishments, and replacements. In its response, the applicant indicated that acceptance criteria are tailored to each individual inspection considering the aging effect being managed. For example, inspections for loss of material provide guidance that requires evaluation under the corrective action program if there is evidence of loss of material beyond uniform light surface corrosion; visually detectable cracking requires evaluation under the corrective action program, and refurbishment and replacements are performed on a specified frequency based on plant experience and/or equipment supplier recommendations. In addition, inspection and surveillance procedures of the periodic surveillance and preventive maintenance program contain requirements for documenting the results of the inspections. On the basis of the

directions provided in plant procedures regarding the need for corrective action, the staff finds the applicant's response to be acceptable.

*[Operating Experience]* The applicant indicated that the periodic surveillance and preventive maintenance program is an established program at Turkey Point and has proven effective at maintaining the material condition of systems, structures, and components and detecting unsatisfactory conditions. The effectiveness of the program is supported by improved system, structure, and component material conditions and reliability, documented by internal and external industry assessments. The periodic surveillance and preventive maintenance program is subject to periodic assessments to ensure effectiveness and continuous improvement. The applicant was asked to demonstrate the effectiveness of the program in the operating experience and demonstration summary. In its response, the applicant indicated that the effectiveness of this program is demonstrated by the high level of system/equipment availability as documented via the plant's periodic assessments under the Maintenance Rule. For example, there have been no functional failures of intake cooling water system pumps, pump discharge check valves, or expansion joints since the inception of the replacement under the periodic surveillance and preventive maintenance program for these components. The staff finds that satisfactory operating experience provides evidence of the effectiveness of this program to manage the aging effects of the specified systems, structures and components.

#### 3.9.11.3 FSAR Supplement

The LRA indicated that the periodic surveillance and preventive maintenance program will be enhanced to address the scope of specific inspections and their documentation. As indicated in Section 16.2.11 of the UFSAR Supplement in Appendix A, specific enhancements to the scope and documentation of some inspections performed under this program will be implemented prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4.

#### 3.9.11.4 Conclusion

The staff has reviewed the information in Section 3.2.11 of Appendix B to the LRA. On the basis of this review, the staff concludes that the continued implementation of the periodic surveillance and preventive maintenance program provides reasonable assurance that the aging effects of loss of material, cracking, fouling, loss of seal, and embrittlement will be managed, such that the components and structural components within the scope of license renewal will continue to perform their intended functions consistent with the CLB during the period of extended operation.

### 3.9.12 Reactor Vessel Head Alloy 600 Penetration Inspection Program

#### 3.9.12.1 Summary of Technical Information in the Application

The applicant described its AMP of the reactor vessel head alloy 600 penetration in Section 3.2.12, "Reactor Vessel Head Alloy 600 Penetration Inspection Program," of Appendix B to the LRA. In Section 3.2.12 of the LRA, the applicant specified that the reactor vessel head Alloy 600 penetration inspection program (RVHPIP) is designed to manage the aging effect of cracking due to stress corrosion in the vessel head penetration (VHP) nozzles. The applicant qualified this statement by stating that the program would include a one-time volumetric examination of the VHPs to the Turkey Point, Unit 4 reactor vessel head, as well as visual examinations of the vessel head external surfaces at Turkey Point, Units 3 and 4, during scheduled outages consistent with the boric acid wastage surveillance program. The staff reviewed the subject AMP to determine whether the applicant has demonstrated that the effects of aging of the Alloy 600 VHPs will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.9.12.2 Staff Evaluation

The applicant credits the RVHPIP for managing aging effects in the Turkey Point Alloy 600 VHP nozzles. The current industry-wide program for monitoring cracking in Alloy 600 VHP nozzles is based on an integrated ranking and monitoring program for VHP nozzles developed by the Nuclear Energy Institute (NEI) in the late 1990s. This program is based on the industry's generic and plant-specific responses to Generic Letter 97-01 (GL 97-01), "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," which ranked the susceptibility of Alloy 600 VHPs to PWSCC based on probabilistic cracking models. Based on the susceptibility rankings for Turkey Point (Letter L-2001-65), the RVHPIP includes a volumetric examination of selected VHP nozzles of Unit 4 to be performed prior to 2007, whereas Unit 3 had a sufficiently low ranking to not require such examination throughout the license renewal period.

From November 2000 to April 2001, reactor coolant pressure boundary (RCPB) leakage from VHP nozzles was identified at four PWR plants. Supplemental examinations of the degraded nozzles indicated the presence of circumferential cracks in four of the CRDM nozzles. These findings are significant in that the cracking was reported to initiate from the OD side of the nozzle, either in the associated J-groove welds or heat-affected-zones, and not from the inside surface of the nozzles as was assumed in the industry responses to GL 97-01. In this recent experience, the degradation was severe enough to penetrate the RCPB, and the circumferential cracking is the first such finding in VHP nozzles in any PWR.

In response to the identified cracking, the NEI and the Materials Reliability Program (MRP) submitted Topical Report TP-1001491, Part 2, "PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plant (MRP-44)." This report included a revised susceptibility ranking model for PWR plants. This revised model places Turkey Point Units 3 and 4 within 10 EFPY of the same conditions evident at the plant which identified three circumferential cracks in its CRDM nozzles, within the current license term for each unit.

To address the potential safety implication of these findings, the NRC issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001. The bulletin (NRC ADAMS Accession Number ML012080284) emphasized the need to use effective examination techniques capable of detecting flaws in these nozzles, in an approach consistent with the relative susceptibility of the VHP nozzles.

Leakage of reactor coolant from the RCPB is not allowed by Turkey Point Technical Specifications, and therefore the overall approach of the RVHPIP described in the application (e.g., leakage detection) may not be consistent with current regulatory and industry efforts to resolve the potential issues of cracking in VHP nozzles. In accordance with the issues raised in NRC Bulletin 2001-001, aging management of PWSCC in the Turkey Point VHP nozzles is an emerging issue that needs to be resolved in coordination with ongoing industry efforts for the current license period. Since the RVHPIP is not consistent with the current status of the

NEI/MRP integrated program for monitoring and controlling PWSCC in VHP nozzles, and since the issues raised in NRC Bulletin 2001-001 are currently being resolved with licensees, the staff has not evaluated the RVHPIP against the AMP attributes listed in Section 2.0 of Appendix B to the application. The staff, therefore, could not complete its review of the acceptability of the RVHPIP for the license renewal term until this program is found acceptable for the current license period. The staff, therefore, considered the acceptability of the RVHPIP to be an open issue and issued Open Item 3.9.12-1 on the AMP.

As stated in Open Item 3.9.12-1, the applicant did not specify in Section 3.2.12 of Appendix B to the LAR whether it would continue to be a participant in the NEI program for managing primary water stress corrosion cracking (PWSCC) in Alloy 600 reactor vessel head penetrations (VHPs) of U.S. pressurized water reactors (PWRs), and whether the applicant would continue to use the reactor vessel head Alloy 600 penetration inspection program (RVHPIP) as a basis for evaluating the Alloy 600 VHPs in the Turkey Point nuclear units during the proposed extended operating terms for the units. The scope of the RVHPIP described in Section 3.2.12 of Appendix B to the LRA needs to be updated to reflect that the applicant will continue to implement program for monitoring and controlling cracking in U.S. VHP nozzles during the period of extended operating term. This includes updating the RVHPIP to reflect the information and relative rankings for the Turkey Point units in Topical Report MRP-44 to make it consistent with NEI's current integrated program for evaluating Alloy 600 VHPs in U.S. PWRs.

In FPL letter L-2001-236 responding to Open Item 3.9.12-1, the applicant stated that it will continue to be a participant in the industry programs for managing PWSCC in Alloy 600 reactor VHP nozzles of U.S. pressurized water reactors during the period of extended operation. The applicant informed the staff that, as documented in FPL's response to NRC Bulletin 2001-01 (refer to FPL Letter #L-2001-198 dated September 4, 2001), the work performed under the Electric Power Research Institute (EPRI) MRP and NEI is an integral part of the Turkey Point RVHPIP. The applicant stated that the bulletin response provides the Turkey Point Unit 3 and 4 rankings utilizing the latest industry PWSCC susceptibility model, in addition to updating reactor VHP inspection commitments, and that, as the industry gains experience, the ranking models will continue to be refined and thus, Turkey Point's RVHPIP will be updated to reflect the new information and relative rankings for Turkey Point Units 3 and 4 in the Topical Reports MRP-44 and 48, accordingly. The staff concludes that this approach will ensure that the RVHPIP will be modified as necessary based on the latest bases for monitoring for and controlling PWSCC in the Turkey Point VHP nozzles.

### 3.9.12.3 FSAR Supplement

The summary description provided in Appendix A, Chapter 16, Section 16.2.12 of the LRA is sufficient.

### 3.9.12.4 Conclusions

The staff has reviewed the information in Appendix B, Section 3.2.12 of the LRA and responses to the staff's RAIs and to Open Item 3.9.12. On the basis of this review, the staff has determined that the applicant has resolved the issue identified in Open Item 3.9.12-1 and has provided a sufficient basis for ensuring that the RVHPIP will be sufficient to monitor and control cracking in the Alloy 600 VHP nozzles and their associated J-groove welds and heat-affected-zones during the proposed operating terms for the Turkey Point units. This basis will ensure that the scope and attributes of RVHPIP will be consistent with the most recent scope and methods developed by the industry for monitoring and controlling PWSCC in U.S. VHP nozzles and their associated J-groove welds. The staff therefore concludes that Open Item 3.9.12-1 is resolved and the RVHPIP is acceptable to ensure that the applicant will monitor for and control PWSCC in the Turkey Point VHP nozzles during the extended periods of operation for the units.

### 3.9.13 Reactor Vessel Integrity Program

The applicant described its reactor vessel integrity program in Section 3.2.13 of Appendix B, to the LRA. For the reactor vessel, Section 3.2.4 and Table 3.2-1 of the LRA identify cracking, reduction in fracture toughness, loss of material, and loss of mechanical closure integrity as aging effects requiring management for the period of extended operation.

The Turkey Point Unit 3 and 4 reactor vessel integrity program is designed to manage reactor vessel irradiation embrittlement, and encompasses the following subprograms:

- reactor vessel surveillance capsule removal and evaluation
- fluence and uncertainty calculations
- monitoring effective full power years
- pressure/temperature limit curves

Through the reactor vessel integrity program, the applicant intends to comply with the requirements of 10 CFR 50.60, Appendices G and H, and 10 CFR 50.61.

The four subprograms are reviewed separately in the following paragraphs.

Criteria for the first 40 years are specified in 10 CFR Part 50, Appendix H, "Reactor Vessel Materials Surveillance Program," for monitoring changes in the fracture toughness of ferritic materials in the reactor beltline region to neutron irradiation, and thermal environments. Appendix H requires that the surveillance program design and withdrawal schedule meet the requirements of American Society for Testing and Materials (ASTM) E-185, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Vessels."

Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," describes general procedures acceptable to the NRC staff for calculating the effects of neutron irradiation embrittlement of the low alloy steels used for light water-cooled RVs. Surveillance data from the Appendix H program are used in RG 1.99, Revision 2 calculations, if applicable.

By letter dated February 8, 1985, the staff approved the combination of the Turkey Point Unit 3 and 4 material surveillance programs into a single integrated program. In a letter dated July 11, 1997, the staff approved BAW-1543, Revision 4, including Supplements 1 and 2, "Master Integrated Reactor Vessel Surveillance Program," to demonstrate continuous management of aging effects for all plants included in BAW-1543, Revision 4, Supplement 2. Turkey Point, Units 3 and 4, were included in this report. Turkey Point, Units 3 and 4, were a special case, since each of the other Babcock & Wilcox and Westinghouse plant-specific reactor vessel surveillance programs were prepared in accordance with ASTM E 185-82. The Turkey Point Unit 3 and 4 reactor vessels were purchased to the Summer 1966 Addenda to the 1965 Edition of the ASME Code. ASTM E 185-66 was the surveillance capsule standard in effect at the time the Turkey Point Unit 3 and 4 reactor vessels were purchased. Since the Turkey Point Unit 3 and 4 capsule withdrawal schedules meet the ASTM E 185 Edition that was current at the time the reactor vessels were purchased, the withdrawal schedules meet the requirements of Appendix H to 10 CFR Part 50. Staff approval in the 1985 and 1997 letters were for a 40-year license term.

### 3.9.13.1 Reactor Vessel Surveillance Capsule Removal and Evaluation

#### 3.9.13.1.1 Summary of Technical Information in the Application

The applicant described the reactor vessel surveillance capsule removal and evaluation in Appendix B, Section 3.2.13.1 of the LRA. The staff reviewed the program in Appendix B, Section 3.2.13.1 of the LRA to determine whether the applicant has demonstrated that the aging effects covered by the subject program will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

#### 3.9.13.1.2 Staff Evaluation

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff's evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are discussed below.

[*Program Scope*] The reactor vessel surveillance capsule removal and evaluation program manages the aging effect of reduction in fracture toughness due to neutron irradiation on the reactor vessel beltline forgings and circumferential welds. The aging effect is managed by performing Charpy V-notch and tensile tests on specimens that are irradiated in the reactor vessel. Based on the above, the program scope is appropriate.

[*Preventive or Mitigative Actions*] There are no preventive or mitigative actions associated with the reactor vessel surveillance capsule removal and evaluation program, nor did the staff identify a need for such actions.

[*Parameters Monitored or Inspected*] The parameter to be monitored is the increase in temperature at the 30 ft-lb energy from unirradiated and irradiated specimens. The tests are in accordance with the applicable ASTM standards identified in Section 5.0 of BAW-1543A, Revision 2. In addition, accumulated neutron fluence is monitored utilizing surveillance capsule dosimetry.

In RAI 3.9.13-1 (letter dated February 1, 2001) the staff requested additional information regarding modifications to the reactor vessel surveillance program to accommodate a 60-year license. In a letter dated April 19, 2001, the applicant provided the response to the requested information. Specifically, the applicant stated that the 48 EFPY peak neutron fluence (inside wall) for the Turkey Point circumferential welds is projected to be less than 4.5 X $10^{19}$ n/$cm^2$ which is equivalent to approximately 2.8 X $10^{19}$ n/$cm^2$ at 1/4T location. The Turkey Point Unit 4 "X" capsule is currently projected to be removed in 2007 at a fluence of 3.85 x $10^{19}$ n/$cm^2$

which is greater than the 1/4T fluence at 48 EFPY.

The amount of radiation embrittlement of the circumferential beltline welds is based on the methodology in RG 1.99, Revision 2 (See staff evaluation in section 4.2.2). The data in the Turkey Point surveillance capsules, including Capsule X, are used to monitor radiation embrittlement of the reactor vessel circumferential beltline welds. To date, capsules that contain weld metal and have neutron fluences exceeding $4 \times 10^{19}$ n/cm$^2$ have been withdrawn from Dresden 2 (Capsule 5), Maine Yankee (Capsule A-35), Prairie Island 1 (Capsules R and S), Prairie Island 2 (Capsule R) and Robinson (Capsule T). The measured increase in reference temperature for all these data, except for the Dresden 2 data, are within the RG predicted increase in reference temperature plus two standard deviation values, which indicates that the methodology in RG 1.99, Revision 2 is applicable for fluence exceeding $4 \times 10^{19}$ n/cm$^2$. Therefore, although the neutron fluence for Capsule X will not exceed the 48 EFPY peak neutron fluence for the Turkey Point circumferential welds, its data may be extrapolated to the higher fluence using the methodology in RG 1.99, Revision 2.

The applicant also stated that there are nine remaining standby capsules in the Turkey Point vessels from which to gather data on fluence, spectrum, temperature, and neutron flux. The last capsule will not be withdrawn prior to the 55$^{th}$ year as shown in LRA Appendix A, Table 4.4-2 (page A-10). The staff finds this response to be acceptable since the available surveillance capsule data are sufficient to monitor changes in the RV material due to neutron irradiation during the license extension period.

[*Detection of Aging Effects*] The aging of the affected components will be detected by quantifying the change in temperature at 30 ft-lb energy from unirradiated and irradiated specimens. The staff finds this approach to be acceptable since it will determine the increase in reference temperature due to irradiation.

[*Monitoring and Trending*] Empirical material fracture toughness and accumulated neutron fluence data are obtained from the vessel irradiated surveillance specimens. These data and the trend curves from RG 1.99, Revision 2, provide the basis for the value for reference temperature for nil-ductility transition ($RT_{NDT}$) and for determining reactor vessel heatup and cooldown limits. These data are monitored and trended to ensure continuing reactor vessel integrity. The surveillance capsule withdrawal schedule is specified in Chapter 4 of the UFSAR Supplement provided in Appendix A to the LRA. Turkey Point, Units 3 and 4, have sufficient surveillance capsules for the extended period of operation. Future decisions concerning the frequency of withdrawal of surveillance capsules will be based on changes in fuel type or fuel loading pattern. The staff finds this response to be acceptable since it will monitor operating changes and RV integrity.

[*Acceptance Criteria*] The acceptance criteria for fracture toughness are that the $RT_{PTS}$ value for each reactor vessel material shall remain below the screening criteria of 270 °F for plates and axial welds, and below 300 °F for circumferential welds. The requirement also includes a Charpy upper-shelf energy (USE) greater than 50 ft-lb. For materials whose Charpy USE fall below 50 ft-lb, there are provisions in Appendix G to 10 CFR Part 50 which must be followed. Specifically, the applicant must demonstrate that, during the period of extended operation, the Charpy USE has a margin of safety against fracture equivalent to that specified in Section XI of the ASME Boiler and Pressure Vessel Code. The staff finds this approach to be acceptable since it complies with 10 CFR 50.61, the PTS rule.

[*Operating Experience*] The reactor vessel surveillance capsule removal and evaluation program meets the requirements of 10 CFR Part 50, Appendix H, and has been in effect since the initial plant startup. This program has been updated over the years and has provided experience in addressing reduction in fracture toughness. Turkey Point Unit 3 and 4 pressure-temperature (P-T) limit curves have been updated using results from the vessel surveillance capsule specimen evaluations. Turkey Point, Units 3 and 4, have been evaluated to have values for $RT_{PTS}$ that are below the screening criteria in 10 CFR 50.61. The staff finds the applicant's description of operating experience acceptable.

3.9.13.1.3 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description of the reactor vessel surveillance capsule removal and evaluation program described in Appendix A to the LRA is acceptable.

3.9.13.1.4 Conclusion

The staff has reviewed the information in Section 3.2.13.1 of Appendix B of the LRA. On the basis of this review, the staff finds that the reactor vessel surveillance capsule removal and evaluation program for Turkey Point, Units 3 and 4, is acceptable, and the single integrated surveillance program between the two units will directly measure the increase in 30 ft-lb transition temperature as a function of neutron irradiation. The data will be applied to the Turkey Point Unit 3 and 4 reactor vessels, and the applicant will ensure that the fracture toughness values meet the requirements of 10 CFR Part 50 or the applicable sections of the ASME Boiler and Pressure Vessel Code, as described above under "Acceptance Criteria." Therefore, the staff finds that the aging effects associated with this program will be adequately managed in accordance with the CLB during the period of extended operation.

3.9.13.2 Fluence and Uncertainty Calculations

3.9.13.2.1 Summary of Technical Information in the Application

The applicant described its fluence and uncertainty calculations in Section 3.2.13.2 of Appendix B of the LRA. The staff reviewed the program in Appendix B, Section 3.2.13.2 of the LRA to determine whether the applicant has demonstrated that the aging effects covered by the subject program will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.9.13.2.2 Staff Evaluation

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff's evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are evaluated below.

[*Program Scope*] The purpose and scope of the reactor vessel fluence and uncertainty calculations are to provide accurate predictions of the actual reactor vessel neutron fast fluence value for use in the development of the P-T limit curves and pressurized thermal shock calculations. The staff finds the applicant's definition of program scope acceptable.

[*Preventive or Mitigative Actions*] There are no preventive or mitigative actions associated with the fluence and uncertainty calculations, nor did the staff identify a need for such actions. The staff finds this response acceptable since no preventive or mitigative actions are associated with this subprogram.

[*Parameters Monitored or Inspected*] The monitored parameters are the reactor vessel neutron fast fluence values, which are predicted based on analytical models meeting the requirements of draft NRC DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," and are benchmarked using dosimetry results that are available from the reactor vessel surveillance capsule removal and evaluation subprogram. Note that in the past, benchmarking has been supplemented by draft RG DG-1053 cavity (ex-vessel) dosimetry. In RAI 3.9.13.2-1 (letter dated February 1, 2001) the staff requested additional information regarding the reactor vessel fluence calculations. In a letter dated April 19, 2001, the applicant stated that the determination of the fluence is based on both calculations and measurements. The fluence prediction is made with calculations, and measurements are used to qualify the calculational methodology.

The applicant has implemented a pressure vessel radiation surveillance program at Turkey Point. The program is based on ASTM E185. Eight materials test capsules were placed in each unit (16 total). Additionally, external neutron dosimeters have been installed and analyzed. The program provides for the periodic removal of capsules and/or dosimeters for evaluation throughout the plant life. The present database at Turkey Point includes data evaluated from three Unit 3 capsules, two Unit 4 capsules, and cycle-specific cavity dosimetry measurements during Unit 3 Cycles 10 and 15. The results from these measurements, the Unit 3 and 4 operating histories, and calculated power distributions make up the database for the fluence calculations.

The most recent data calculations use discrete ordinates radiation transport (DORT) for the neutron transport calculation, a DORT post-processor code named DOTSOR for geometry conversion, and Bugle-96, an ENDF-B-VI-based cross-section library. The power distributions are based on the Westinghouse Advanced Nodal Code (ANC).

The fluence calculation methods include the following:

- The calculation uses detailed modeling of the capsules and cavity dosimeters that include significant structural and geometrical details necessary to define the neutron environment at points of interest.

- The transport calculation for the reactor model was carried out in the R,θ and R,Z coordinates using DORT and BUGLE-96. The R,θ model included 152 mesh points in the R direction covering the range from the center of the core to about 14 cm into the concrete shield to account for back scatter. In the azimuthal direction, 47 mesh points were used which models an octant of the reactor.

- The core power distribution used to determine the neutron source was calculated from ANC nodal calculations. The relative pin-by-pin distributions for each assembly location together with the cycle burnup for each assembly were used to determine the relative power output for each pin in the core, averaged over the cycle. The DOTSOR code was used to convert this power distribution from x,y to R,θ coordinates and to place the source in each mesh cell. The average assembly burnup was used to determine the source per group, the average neutrons per fission and the average energy per fission.

- Neutron dosimetry analysis of the passive sensors within the surveillance capsule, which included activation measurement and evaluation of their composition and location, are also considered in the development of fluence results.

- Calculation to measurement (C/M) comparisons indicated a C/M ratio greater than 1.0. The calculated values were used without modification, consistent with the recommendation of DG-1053.

- Fluence projections use power distributions which are representative of planned future fuel management using flux suppression inserts in the assemblies at the core flats. Core designs are controlled by limiting the power in the peripheral assemblies at these locations.

The staff finds this response to be acceptable since it is consistent with the recommendations in DG-1053.

[*Detection of Aging Effects*] Fluence values in excess of predicted values can result in lower fracture toughness values in reactor vessel materials due to irradiation embrittlement. The potential for these effects is determined using calculations of vessel fluence, empirical results from Charpy V-notch tests of irradiated specimens, and capsule dosimetry in accordance with the reactor vessel surveillance capsule removal and evaluation program. The staff finds this approach to be acceptable since the above-mentioned parameters are sufficient for determining predicted fluence valves.

[*Monitoring and Trending*] Neutron fluence and uncertainty calculations are performed to predict the neutron fast fluence. These calculations are verified using dosimetry results that are available from the reactor vessel surveillance capsule removal and evaluation program, as supplemented by the cavity (ex-vessel) dosimetry. The frequency of updating fluence and uncertainty calculations may change as additional data are obtained. Changes in fuel type or fuel loading pattern may also change the frequency of surveillance capsule withdrawal and the performance of neutron fluence and uncertainty calculations. The staff finds this approach acceptable since dosimetry results can be used to verify calculations to predict neutron fluence.

[*Acceptance Criteria*] Based on the calculations, the reactor vessel fluence uncertainty values are to be within the NRC-suggested ±20%. Calculated fluence values for fast neutrons (above 1.0 MeV) are compared with measured values. This methodology represents a continuous validation process to ensure that no biases have been introduced and that the uncertainties remain comparable to the reference benchmarks. The staff finds this approach to be acceptable since it is a continuous validation process.

[*Operating Experience*] The neutron fluence and uncertainty calculations for Turkey Point Units 3 and 4, have been performed in accordance with the guidelines of draft RG DG-1053 and validated using data obtained from the capsule dosimetry. The results of the fluence uncertainty values are to be within the NRC-suggested limit of ±20%. This has been validated by the comparison of the calculated fluence values with measured values. This methodology represents a continuous validation process to ensure that no biases have been introduced, and that the uncertainties remain comparable to the reference benchmarks. The staff finds the applicant's description of operating experience acceptable.

3.9.13.2.3 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description of the fluence and uncertainty calculations program described in Appendix A to the LRA is acceptable.

3.9.13.2.4 Conclusion

The staff has reviewed the information in Section 3.2.13.2 of Appendix B to the LRA. On the basis of this review, the staff finds the calculations are consistent with requirements of 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and finds the program acceptable for the period of extended operation.

3.9.13.3 Monitoring Effective Full Power Years

3.9.13.3.1 Summary of Technical Information in the Application

The applicant described the monitoring of effective full power years (EFPY) in Section 3.2.13.3 of Appendix B to the LRA. The staff reviewed the program in Section 3.2.13.3 of Appendix B to the LRA to determine whether the applicant has demonstrated that the aging effects covered by the subject program will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.9.13.3.2 Staff Evaluation

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff's evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are evaluated below.

[*Program Scope*] The purpose and scope of this program are to accurately monitor and tabulate the accumulated operating time experienced by the reactor vessel. The EFPY data are used to ensure that the power history is within ±0.3 effective full-power days (EFPD) of the plant computer generated value and to determine the period of time for which the P-T limit curves are applicable. The staff finds the applicant's definition of program scope acceptable.

[*Preventive or Mitigative Actions*] There are no preventive or mitigative actions associated with the monitoring of the EFPY program, nor did the staff identify a need for such actions. The staff finds this response acceptable since no preventive or mitigative actions are associated with this program.

[*Parameters Monitored or Inspected*] The program monitors and tabulates the accumulated operating time experienced by the Turkey Point Unit 3 and 4 reactor vessels. The EFPYs of plant operation are based on reactor incore power calculations obtained from the plant's nuclear applications software program. Site reactor engineers determine EFPY values by comparing burnup estimated from incore instrumentation to the thermal power calculated burnup. The staff finds this approach to be acceptable since it uses plant parameters to calculate EFPY of operation.

[*Detection of Aging Effects*] EFPY calculations are utilized for the prediction of neutron fast fluence and the determination of the reduction in fracture toughness of reactor vessel critical materials. The staff finds this approach to be acceptable since it facilitates the calculation of neutron fluence and the determination reduction of fracture toughness in beltline materials.

[*Monitoring and Trending*] This program monitors the reactor vessel EFPYs to be used in predicting the neutron fast fluence. Each Turkey Point unit is monitored to determine the EFPY of operation. These data are used to validate the applicability of the P-T limit curves for the next operating cycle. The staff finds this approach to be acceptable since it is used to monitor applicability of the P-T limit curves.

[*Acceptance Criteria*] Calculated effective full power years shall not exceed the Technical Specification limit for the validity of the pressure-temperature limit curves. The staff finds this approach acceptable because it is consistent with the requirements of 10 CFR 50.60.

[*Operating Experience*] The EFPY values are determined by comparing the fuel burnup to the thermal power calculated burnup. The fuel burnup comparisons have been found to be within the expected accuracy. The staff finds the applicant's description of operating experience acceptable.

#### 3.9.13.3.3 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description for the monitoring effective full power years program described in LRA Appendix A is acceptable.

#### 3.9.13.3.4 Conclusion

The staff has reviewed the information in Appendix B, Section 3.2.13.3 of the LRA. On the basis of this review, the staff finds this procedure meets the requirements of 10 CFR Part 50, Appendix G, and finds it acceptable for the period of extended operation.

### 3.9.13.4 Pressure-Temperature Limit Curves

#### 3.9.13.4.1 Summary of Technical Information in the Application

The applicant described the pressure-temperature limit curves in Appendix B, Section 3.2.13.4 of the LRA. The staff reviewed the program in Appendix B, Section 3.2.13.4 of the LRA to determine whether the applicant has demonstrated that the aging effects covered by the subject program will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

#### 3.9.13.4.2 Staff Evaluation

The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and cover all structures and components subject to AMR. The staff's evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this SER. This program satisfies the elements of corrective actions, confirmation process, and administrative controls. The remaining seven elements are evaluated below.

[*Program Scope*] The purpose and scope of this program are to establish P-T limit curves for the normal operating, inservice leak test, and hydrostatic test limits for the RCS, as applicable to the Turkey Point Unit 3 and 4 pressure vessels. The curves are used to limit operations based on the material properties of the vessel caused by neutron irradiation. The staff finds the applicant's definition of program scope acceptable.

[*Preventive or Mitigative Actions*] Pressure-temperature limit curves are provided to specify the maximum allowable pressure as a function of reactor coolant temperature in order to prevent or minimize the effects of reduced fracture toughness caused by neutron irradiation. The staff finds these actions acceptable since they will ensure that the plant is operating within the allowable pressure and temperature ranges.

[*Parameters Monitored or Inspected*] Pressure-temperature limit curves are generated assuming that a 1/4T surface flaw exists, and using the fracture mechanics methodology in ASME Section XI, Appendix G. The P-T curves are determined by using bounding input heatup and cooldown transients. The staff finds this approach to be acceptable since the P-T limits curves are generated to meet the requirements in Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50.

[*Detection of Aging Effects*] The P-T limit curves are not provided for the detection of aging effects but rather to prevent or minimize the effects of reduced fracture toughness caused by neutron irradiation. The staff finds this response acceptable since it clarifies the purpose of the P-T limit curves.

[*Monitoring and Trending*] The P-T limit curves are valid for a period expressed in EFPYs. These curves are updated prior to exceeding the EFPYs for which they are valid. The time period for updating P-T limit curves may change if conditions such as changes in fuel type or fuel loading pattern occur. The staff finds this approach acceptable since P-T limits curves will be updated prior to exceeding the applicable EFPYs.

[*Acceptance Criteria*] The P-T limit curves are valid for a specified number of EFPYs. The curves must be updated before this time period is exceeded. The staff finds this approach acceptable since the validity of the curves is monitored and the P-T limit curves are updated prior to exceeding the applicable EFPY.

[*Operating Experience*] Turkey Point, Units 3 and 4, operate in accordance with P-T limit curves that have been updated using the results of data obtained from surveillance capsule specimens. The P-T limit curves provide sufficient operating margin while preventing or minimizing the effects of reduced fracture toughness caused by neutron irradiation. The staff finds the applicant's description of operating experience acceptable.

3.9.13.4.3 FSAR Supplement

On the basis of the staff's evaluation described above, the summary description for the pressure temperature limit curves program described in LRA, Appendix A is acceptable. The staff's evaluation of the calculational methodology for the curves, and the extension to 48 EFPYs is described in the pressure-temperature limits TLAA section of this SER.

3.9.13.4.4 Conclusion

The staff has reviewed the information in Section 3.2.13.4 of Appendix B of the LRA. On the basis of this review, the staff finds this procedure meets the requirements of 10 CFR Part 50, Appendix G. Therefore, the staff finds it acceptable for the period of extended operation.

3.9.14 Steam Generator Integrity Program

The applicant described its AMP, steam generator integrity program, in Section 3.2.14 of Appendix B, of the LRA. The program is aimed at verifying the integrity of various steam generator components. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the effects of aging will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.9.14.1 Summary of Technical Information in the Application

As identified in Chapter 3, the steam generator integrity program is credited for aging management of the steam generators. The program manages the aging effects of cracking and loss of material and includes the following essential elements: inspection of steam generator tubing and tube plugs; steam generator secondary-side integrity inspections; tube integrity assessment; assessment of degradation mechanisms; primary-to-secondary leakage monitoring; primary and secondary chemistry control; sludge lancing; maintenance and repairs; and foreign material exclusion. Inspections and other aging management activities are performed in accordance with the Turkey Point Unit 3 and 4 Technical Specifications, and the program is structured to meet NEI 97-06, "Steam Generator Program Guidelines."

3.9.14.2 Staff Evaluation

The staff evaluation of the AMP focused on the program elements rather than details of specific plant procedures. To determine whether the AMPs are adequate to manage the effects of aging so that the intended functions will be maintained consistent with the CLB throughout the period of extended operation, the staff evaluated the following 10 elements: (1) program scope, (2) preventive or mitigative actions, (3) parameters monitored or inspected, (4) detection of aging effects, (5) monitoring and trending, (6) acceptance criteria, (7) corrective actions, (8) confirmation process, (9) administrative controls, and (10) operating experience.

The application indicates that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and apply to AMPs credited for license renewal and are performed, or in the case of new programs to be performed, in accordance with the applicant's quality assurance program. The staff's evaluation of the quality assurance program is provided separately in Section 3.1.2 of the SER. The remaining seven element are evaluated below.

[*Program Scope*] The steam generator integrity program ensures that steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The program is structured to meet the NEI 97-06, "Steam Generator Program Guidelines," which references several EPRI guidelines. These EPRI guidelines include steam generator examination, tube integrity assessments, both primary and secondary water chemistry, primary-to-secondary leakage, in situ pressure testing, and tube plug assessment. The program provides for comprehensive examinations of steam generator tubes and plugs to identify and repair degraded conditions before the degradation exceeds allowable limits. The staff finds that the scope of the steam generator integrity program is adequate.

[*Preventive or Mitigative Actions*] Preventive measures include primary and secondary chemistry control. As clarified in response to RAI 3.9.4-1, the applicant stated that the chemistry control program currently complies with the following industry guidelines:

- EPRI, TR-105714, Rev. 4, "PWR Primary Water Chemistry Guidelines," Vols. 1 and 2
- EPRI, TR-102134, Rev. 5, "PWR Secondary Water Chemistry Guidelines"

On the basis of the staff's review of the applicant's chemistry control program in Section 3.1.1 of this SER, the staff finds the preventive actions acceptable.

[*Parameters Monitored or Inspected*] The applicant applies volumetric inspection techniques, primarily eddy current testing, to detect degradation of the steam generator tubes and plugs. Inspection activities also monitor for leakage from tube plugs. In response to RAI 3.9.14-3, the applicant states that the scope of eddy current and visual inspections incorporate the guidance contained in NEI 97-06 and WCAP 15093, "Evaluation of EDF Steam Generator Internals Degradation – Impact of Causal Factors on the Westinghouse Models F, 44F, D and E2 Steam Generators" for detection of potential tube and plug degradation, and degradation of internal component's and the presence of loose parts. Examination personnel are qualified in accordance with the standards and criteria provided in NEI 97-06, examination techniques are qualified and validated for site-specific use in accordance with the standards and criteria contained in NEI 97-06, and steam generator tube integrity is assessed in accordance with performance criteria in NEI 97-06. Primary-to-secondary leakage is monitored during operation. The staff finds the parameters inspected under this program are acceptable because they will be effective in managing the specified aging effects.

[*Detection of Aging Effects*] The applicant stated that the extent and schedule of the inspections prescribed by the steam generator integrity program are designed to ensure that flaws do not exceed established performance criteria. The extent and schedule of the inspections are designed to ensure timely detection and replacement of leaking plugs. Lastly, detection of primary-to-secondary leakage during plant operation will assist in identifying potentially unacceptable tube degradation caused by the aging mechanisms. The staff agrees that these are acceptable methods for identifying steam generator degradation.

[*Monitoring and Trending*] The applicant's inspection intervals are based on technical specification requirements as well as the guidance contained in NEI 97-06. The inspections are expected to provide timely detection of cracking, pitting, and wear. In addition, the frequency and extent of plug inspections are expected to provide for timely detection of tube plug leakage. Lastly, daily monitoring of primary-to-secondary leakage will identify degradation of steam generator tubing. The staff finds the monitoring and trending activities acceptable.

[*Acceptance Criteria*] The program requires that steam generator tubes are removed from service in accordance with the requirements of the technical specifications and the steam generator integrity program. Any tube plug leakage detected requires tube plug replacement. Identified primary-to-secondary leakage is compared with the limits allowed by the technical specifications. In response to RAI 3.9.14-4, the applicant stated that Turkey Point power plant procedures for off-normal conditions associated with primary-to-secondary steam generator tube leakage incorporate the operational leakage performance criteria provided in NEI 97-06. These criteria are more restrictive and, thus, bound the technical specification primary-to-secondary leakage limits. The staff finds the acceptance criteria acceptable.

[*Operating Experience*] The current steam generator inspection activities have been evaluated against industry recommendations provided by EPRI and Westinghouse. The steam generator integrity program is not a new program, and has been effective at Turkey Point in ensuring the timely detection and correction of the aging effects of cracking and loss of material in steam generator tubes. The steam generator integrity program considers the guidance provided in NEI 97-06 which is all-inclusive in managing steam generator tube bundle and internals degradation. The staff agrees with the applicant's assessment of operating experience.

#### 3.9.14.3 FSAR Supplement

The staff has confirmed that the steam generator integrity program as described in the FSAR Supplement contains the appropriate essential elements.

#### 3.9.14.4 Conclusion

The staff has reviewed the information provided in Appendix B, Section 3.2.14 of the LRA. On the basis of this review, as set forth above, the staff concludes that the applicant has demonstrated that the steam generator integrity program will adequately manage aging effects for steam generators in accordance with the CLB throughout the period of extended operation.

### 3.9.15 Systems and Structures Monitoring Program

This program is covered in Section 3.1.3 of this safety evaluation report.

### 3.9.16 Thimble Tube Inspection Program

The applicant described its thimble tube inspection program in Section 3.2.16 of Appendix B of the LRA. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the aging effects of the incore instrumentation thimble tubes will be adequately managed by this program during the period of extended operation as required by 10 CFR 54.21(a)(3).

#### 3.9.16.1 Summary of Technical Information in the Application

The applicant specified that the thimble tube inspection program is for aging management of thimble tubes in Turkey Point, Units 3 and 4, by conducting inspection of a single thimble tube N-05 in Unit 3. The program utilizes eddy current test (ECT) to determine thimble tube wall thickness and predict wear rates for early identification of the need for corrective action before the potential thimble tube failure.

The applicant indicated that the thimble tube inspection program was created and implemented in both Units 3 and 4 in response to NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors." The selection of a single tube N-05 for aging management during the extended operation is based on assessment of previous inspections and the Time-Limited Aging Analysis (TLAA) results.

#### 3.9.16.2 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 intended function will be maintained consistent with the CLB throughout the period of extended operation for the incore instrumentation thimble tubes. The staff evaluation of the thimble tube inspection program focused on effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

It is noted that corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and covers all structures and components subject to an aging management review. The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this safety evaluation report. This program satisfies the elements of corrective actions, confirmation process and administrative controls. Because of the limited scope of the program, further elaboration on the remaining seven elements is not required.

The existing thimble tube inspection program for Turkey Point, Units 3 and 4, in response to NRC Bulletin 88-09, was started in early 1990s. The program required eddy current testing (ECT) of thimble tubes. The ECT inspections established the tube wall wear rates in both units. It was based on these wear rates and the TLAA results, the applicant determined that only a single tube (N-05 in Unit 3) requires inspection for the extended operation. Based on above discussion, the staff finds that to inspect only thimble N-05 in the licensee's thimble tube inspection program is acceptable.

Due to potential uncertainties in wear locations and wear rates, the staff feels that TLAA based on previous inspection results obtained in the early 1990s may not be realistic without verification, and the applicant may need to inspect all thimble tubes, or at least a sample of tubes, in both Units 3 and 4 during the one-time inspection for determining the status of wear in thimble tubes. In addition, if more inspections are needed as a result of the one-time inspection, further staff review and updating of the FSAR supplement is needed. Thus the applicant was requested to identify documentation and provide a description of the plant procedures related in a letter dated February 1, 2001, to thimble tube inspection, to provide justification regarding adequacy of inspecting a single tube in the program, and to provide criteria that will be used to determine the scope of additional tests, if necessary.

In its response dated April 19, 2001, the applicant indicated that the procedures for performance of thimble tube ECT inspection were created and used satisfactorily for the determination of thimble tube wall thinning in response to NRC Bulletin 88-09. These procedures consisted of a plant procedure and NDE department procedure. The plant procedure specifies all plant associated requirements, precautions and limitations for performing the thimble tube ECT, including acceptance criteria and corrective action program requirements. The NDE procedure, which is specific for the thimble tubes, provides all technical requirements for performing the thimble tube ECT, including the level of qualification of examination personnel and of others involved in the selection and calibrations of equipment to be used. On the basis of the conservative calculations performed, the Unit 3 thimble tube at location N-05 was determined to be the worst case concerning wall thinning rate. The calculated remaining life for thimble N-05 was determined to be approximately half the life of the thimble tube with the next highest wall thinning rate. On the basis of the considerable margin on the calculated remaining life of all the other thimble tubes tested when compared with the calculated remaining life of thimble N-05, it is reasonable to conclude that the results of ECT on thimble N-05 can be used to justify the acceptance of the other thimble tubes. The applicant indicated that the criteria for determining the scope of additional tests have not yet been established. However, for determining the need for additional ECT on other thimble tubes, consideration will be given to a major reduction on the predicted life of the thimble N-05 when using the test results to recalculate the remaining life of this thimble tube. On the basis of the results of the ECT on the thimble N-05, ECT may be performed on other thimble tubes that were previously tested and identified with high wall thinning rates. The selection of these tubes will depend on the recalculated remaining life of these tubes. The staff finds this acceptable.

Since a thimble tube failure will result in leakage of reactor coolant, it is prudent for the staff to know whether a leaking thimble tube can be isolated. Thus the applicant was requested to describe the corrective actions mentioned in page B-88 of the LRA if a tube leak does occur. In its response, the applicant indicated that manually operated isolation valves are provided for isolating thimble tubes. These valves may be closed after removal of the detector cable assembly. If a thimble tube leak does occur, the affected unit would be shut down in accordance with technical specification requirements. Repairs and subsequent testing would then be performed in accordance with the plant's corrective action program. Based on the above discussion, the staff finds that this is acceptable because the leaked thimble tube is isolable.

3.9.16.3 FSAR Supplement

In section 16.2.16 of appendix of LRA, the applicant states that this inspection will be performed prior to the end of initial operating license term. The staff finds this acceptable.

3.9.16.4 Conclusion

The staff has reviewed the information in Section 3.2.16 of Appendix B to the LRA. On the basis of this review, the staff concludes that the continued implementation of the thimble tube inspection program provides reasonable assurance that the aging effects of thimble tubes in Turkey Point, Units 3 and 4, will be managed, such that early detection of potential thimble tube wear will ensure timely corrective measures to mitigate thimble tube failure in accordance with the CLB during the period of extended operation.

# 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 TLAAs in Section 4.2 of the LRA. The TLAAs 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 TLAAs 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 TLAAs 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 TLAAs 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 ($\Delta RT_{NDT}$), 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), $\Delta RT_{NDT}$ 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 $\Delta RT_{NDT}$ 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 $\Delta RT_{NDT}$ 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:

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

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

and $\sigma_\Delta$ is the standard deviation for $\Delta RT_{NDT}$

For determining M, $\sigma_I = 0$ if a measured value is used. If a generic value is used, $\sigma_I$ is the standard deviation of the set of values used to obtain the mean value. For $\Delta RT_{NDT}$, $\sigma_\Delta = 28$ °F for welds and 17 °F for base metal (plate and forging), except that $\sigma_\Delta$ need not exceed one-half of the mean value of $\Delta RT_{NDT}$. 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 $\sigma$ values is used except that $\sigma_\Delta$ 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 $4.12 \times 10^{19}$ n/cm$^2$ and $4.07 \times 10^{19}$ n/cm$^2$ for Turkey Point Units 3 and 4, respectively. For conservatism, the applicant used a value of $4.5 \times 10^{19}$ 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 $RT_{NDT}$ °F | Margin °F | Chemistry Factor (CF) | Inside Surface fluence factor (ff) | ff x CF | $RT_{PTS}$ °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 EFPYs.

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 (~0.3 x $10^{19}$ 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 TLAAs 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 TLAAs 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 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 addition 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 shiftly 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 TLAAs 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 TLAAs 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/cooldown 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 addition 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/cooldown cycles, which is well above the design 200 heatup/cooldown 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 cooldown, 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 cooldown of the RCS. Therefore, the 500 heatup/cooldown 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 TLAAs 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 it is 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.

# 5. REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During its meeting on October 5, 2001, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the NRC staff's safety evaluation report (SER) on the license renewal application (LRA) for the Turkey Point Nuclear Plant, Units 3 and 4. The ACRS Subcommittee on Plant License Renewal reviewed the SER before its meeting with the NRC staff and the applicant on September 25, 2001. The subcommittee presented its findings during the October 5, 2001, ACRS full committee meeting. Due to the small number of open items, the subcommittee recommended not issuing an interim letter on its review of the license renewal SER with open items.

The staff issued its final SER with the resolutions of open items on February 27, 2002. The staff briefed the ACRS License Renewal Subcommittee on March 13, 2002, near the plant site in Florida City, Florida. The staff briefed the ACRS full committee on April 11, 2002, about the resolution of open items and the emerging issue of whether to include certain components within the scope of license renewal to ensure compliance with the requirements of the station blackout (SBO) rule (10 CFR 50.63).

During the 491st meeting of the ACRS full committee on April 11, 2002, the ACRS completed its review of the Turkey Point, Units 3 and 4, LRA and the staff's SER. The ACRS documented its findings in a letter to the Commission dated April 19, 2002. A copy of that letter is attached.

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ACRS R-1992

April 19, 2002

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4

During the 491st meeting of the Advisory Committee on Reactor Safeguards, April 11-12, 2002, we completed our review of Florida Power and Light Company's (FPL's) license renewal application for the Turkey Point Nuclear Plant, Units 3 and 4, and the NRC staff's final safety evaluation report (SER) on the application. Our review included a plant visit and two meetings of our Plant License Renewal Subcommittee, one of which was conducted on March 13, 2002, in Florida City, Florida. During our review, we had the benefit of discussions with representatives of the NRC staff and FPL. In addition, we discussed written comments on Turkey Point from a member of the public. Our subcommittee also heard oral statements from a member of the public during the meeting in Florida City. We had the benefit of the documents referenced.

**Recommendation and Conclusion**

1. The FPL application for renewal of the operating licenses for Turkey Point, Units 3 and 4, should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that Turkey Point, Units 3 and 4, can be operated in accordance with their licensing bases for the period of extended operation without undue risk to the health and safety of the public.

**Background and Discussion**

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. FPL requested renewal of the operating licenses for Turkey Point, Units 3 and 4, for a period of 20 years beyond the current license terms, which expire on July 19, 2012 (Unit 3), and April 10, 2013 (Unit 4). The final SER documents the results of the staff's review of information submitted by FPL, including commitments that were necessary to resolve open items identified by the staff in the draft SER. The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) subject to aging management review; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the aging management programs. The staff also conducted four site inspections to verify the adequacy of the implementation of the methodology described in the application.

We met with the applicant and the staff on September 25 and October 5, 2001, to review the draft SER. We did not identify any new issues to be addressed by the staff and applicant other than the four open items already identified by the staff. The number of open items was small because the applicant implemented lessons learned from the previous license renewal applications and followed the guidance in Nuclear Energy Institute (NEI) Report 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule." This approach facilitated the review process.

The process implemented by the applicant to identify SSCs that are within the scope of license renewal has been effective. During our review we questioned why certain SSCs were not included in scope, and in all cases the applicant provided appropriate justification for the exclusion. Among these SSCs were the startup transformers that connect the plant to the offsite power source, which typically provides the alternate AC power source during a station blackout (SBO) event. The applicant argued that Turkey Point does not rely on restoration of offsite power to recover from an SBO event. Instead, it relies on the installed capability to cross-connect the emergency diesel generators (EDGs) from one unit to the other. During an SBO event, each of the four EDGs on site is capable of carrying all essential loads of both units. Sufficient diesel fuel is maintained on site to provide the required long-term alternate power source. During our visit to the site, the applicant used the plant simulator to demonstrate its ability to cross-connect the EDGs from the control room. This capability was used during Hurricane Andrew. On this basis, we concur with the applicant that the EDGs provide an effective alternate power source during an SBO event. Subsequently, the staff has determined, however, that components connecting the units to the offsite power source, including the startup transformers, are needed to fulfil the requirements of the SBO Rule. Therefore, they are part of the licensing basis and must be included in the scope of license renewal. The applicant has agreed to meet this requirement.

The applicant has performed a comprehensive aging management review of SSCs that are within the scope of license renewal. The applicant identified aging effects using many data sources, including previously submitted license renewal applications, Babcock & Wilcox license renewal generic information, industry operating experience, Turkey Point operating experience, the draft Generic Aging Lessons Learned report, and Westinghouse Owners Group (WOG) topical reports. As the first Westinghouse-designed reactor being considered for license renewal, Turkey Point participated in a WOG program that developed a series of generic topical reports to demonstrate that the aging effects of reactor coolant system components could be adequately managed throughout the period of extended operation. The WOG submitted four topical reports for NRC staff review and approval. The topical reports contain generic license renewal evaluations of pressurizers (WCAP-14574), Class 1 piping and associated pressure boundary components (WCAP-14575), reactor internals (WCAP-14577), and reactor coolant system supports (WCAP-14422).

The applicant did not incorporate these reports by reference in the Turkey Point license renewal application because the staff had not approved these reports at the time the application was submitted to the NRC. These reports were subsequently approved by the staff. In its application, the applicant addresses the applicability of these reports to Turkey Point SSCs to facilitate the staff review. We have reviewed these topical reports and found that, when supplemented by the Turkey Point plant-specific responses to the staff's open issues on the topical reports, they effectively support the Turkey Point license renewal application.

Appendix B of the application describes the 16 existing programs and the 7 new programs that FPL has implemented to manage aging effects during the period of extended operation. The resolution of staff questions and SER open items has resulted in additional commitments, including a program to deal with the adverse localized effects of heat on medium and low-voltage nonenvironmentally qualified (EQ) cables, connections, and electrical/instrumentation and control penetrations in containment, as well as an expanded number of piping segments to be managed to address the potential interaction of Class II piping with safety systems.

Unlike previous applicants, FPL has not proposed an aging management program for non-EQ medium-voltage cables that are exposed to significant moisture. The applicant stated that these cables are designed with lead sheath to prevent failure from moisture ingress. The applicant presented information, including significant industry operating experience, that indicates that this type of jacket provides an impermeable barrier. Based on this information, we agree with the applicant and the staff that no aging management program is needed for non-EQ medium-voltage cables that are subjected to significant moisture.

The Turkey Point application identifies cracking of the control rod drive mechanism (CRDM) penetration nozzles as an aging effect to be managed. Appendix B of the application describes the aging management program, "Reactor Vessel Head Alloy 600 Penetration Inspection Program (RVHPIP)," instituted to deal with this aging degradation mechanism. This program identifies primary water stress corrosion cracking (PWSCC) of Alloy 600 nozzles as the aging effect of concern and ties programmatic elements, such as the frequency of inspections, to the results of plant-specific and sister plant inspection findings. In response to an SER open item, the applicant has committed to continue its participation in the Electric Power Research Institute (EPRI) and NEI programs for managing PWSCC in Alloy 600 reactor vessel head penetration nozzles during the period of extended operation, and has made the NEI program and EPRI Materials Reliability Program (MRP) an integral part of the RVHPIP. This ensures that, as the industry gains more experience with this degradation mechanism, the applicant will update the RVHPIP to reflect the new information. Over the past 6 months, the applicant has performed inspections of upper heads of both units. No leakage of the CRDM penetration nozzles was identified.

A member of the public provided us with written comments expressing his concerns with the continued operation of Turkey Point. His concerns included potential voids in containment walls, the ability of Turkey Point to withstand Category 5 hurricanes, and the vulnerability of the site to external threats. Some of these concerns were echoed by another member of the public during the Subcommittee meeting on March 13, 2002 in Florida City. Based on information provided by the staff and the applicant during our meeting, we conclude that the issue of voids in containment walls has been appropriately resolved at Turkey Point. With regard to concerns about storm surges, the Individual Plant Examination of External Events for Turkey Point identifies such surges as small contributors to total risk. However, the staff should document its position on this issue. The staff is generically addressing concerns with external threats.

The staff has performed a comprehensive review of the FPL application. The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. Adequate programs have been established to manage the effects of aging so that Turkey Point, Units 3 and 4, can be operated in accordance with their current licensing bases for the period of extended operation, without undue risk to the health and safety of the public.

Sincerely,

**/RA/**

George E. Apostolakis
Chairman

References:


1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4," February 2002.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4," September 2001.
3. Nuclear Energy Institute Report 95-10, Revision 1, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," January 2000.
4. Westinghouse Owners Group Topical Report, WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers," July 1996.
5. Westinghouse Owners Group Topical Report, WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation for Class 1 Piping and Associated Boundary Components," August 1996.
6. Westinghouse Owners Group Topical Report, WCAP-14577, Revision 1, "License Renewal Evaluation: Aging Management for Reactor Internals," dated October 9, 2000.
7. Westinghouse Owners Group Topical Report, WCAP-14422, Revision 2, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports," February 1997.
8. Letter dated February 16, 2002, from Mark P. Oncavage, a public citizen, to Noel Dudley, Senior Staff Engineer, ACRS, transmitting safety concerns regarding the continued operation of Turkey Point through the license renewal period.
9. U. S. Nuclear Regulatory Commission, NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," draft report for public comment, April 2001.

# 6. CONCLUSIONS

The staff reviewed the license renewal application for Turkey Point Nuclear Plant, Units 3 and 4, in accordance with Commission's regulations and the NRC's draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," dated August 2000. The revised SRP was issued as NUREG-1800 in July 2001. 10 CFR 54.29 identifies the standards for issuance of a renewed license.

On the basis of its evaluation of the application as discussed above, the staff has determined that the requirements of 10 CFR 54.29(a) have been met.

The staff notes that any requirements of Subpart A of 10 CFR Part 51 are documented in the final plant-specific supplement to the Generic Environmental Impact Statement, dated January 2002.

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## APPENDIX A
## CHRONOLOGY

This appendix contains a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and Florida Power & Light Company (FPL) and other correspondence regarding the NRC staff's review of the Turkey Point Nuclear Plant, Units 3 and 4 (under Docket Nos. 50-250 and 50-251) for license renewal application (LRA).

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| September 8, 2000 | In a letter (signed by T. Plunkett), FPL submitted its LRA for Turkey Point Nuclear Plant, Units 3 and 4, as well as a copy of the boundary drawings to the NRC. |
| September 19, 2000 | In a letter (signed by C. Grimes), NRC informed FPL that the NRC received the Turkey Point Nuclear Plant, Units 3 and 4, LRA on September 11, 2000, and that Mr. Rajender Auluck was appointed as the project manager for the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| October 4, 2000 | In a letter (signed by D. Mathews), NRC informed FPL that the NRC staff has determined that FPL has submitted sufficient information that is complete and acceptable for docketing, proposed review schedule, and opportunity for hearing. |
| November 1, 2000 | In a meeting summary (signed by R. Auluck), NRC summarized the meeting held to familiarize the NRC staff with the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| December 22, 2000 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Sections 2.3.3.10 – 12 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| December 22, 2000 | In a meeting summary (signed by R. Auluck), NRC summarized the October 31, 2000, meeting with FPL regarding review of equipment qualification (EQ) calculations for the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| January 10, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Sections 2.3.4 and 3.5 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| January 17, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Sections 3.7, 4.4, and Appendix B, 3.2.6 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| January 17, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Sections 2.3.3.8, 2.4.2.8, and 2.4.2.10 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |

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| January 19, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 2.3.3.10 - 12 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on December 22, 2000. |
| January 24, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Section 2.3.3.14 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| January 31, 2001 | In a letter (signed by J. Wilson), NRC requested FPL provide additional information (RAI) regarding severe accident mitigation alternatives for Turkey Point Nuclear Plant, Units 3 and 4. |
| February 1, 2001 | In a letter (signed by S. Koenick), NRC requested that FPL provide additional information (RAI) on Sections 4.2, 4.7.1 and Appendix B Sections 3.1.5, 3.1.6, 3.1.7, 3.2.1.1, 3.2.2, 3.2.3, 3.2.4, 3.2.9, 3.2.11, 3.2.12, 3.2.13, 3.2.14, and 3.2.16 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 2, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Sections 2.1, 2.3.1, 2.3.2.2, 2.3.3.3, 2.3.3.4, 2.4.1, and 2.4.2.4 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 2, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Section 3.2 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 2, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Section 3.3 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 2, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Section 3.4 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 2, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Section 3.6 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 2, 2001 | In a letter (signed by R. Auluck), NRC requested that FPL provide additional information (RAI) on Sections 4.3, 4.5, 4.6, 4.7.4 and Appendix B, Sections 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.2.1.2, 3.2.1.3, 3.2.1.4, 3.2.5, 3.2.8, 3.2.10, and 3.2.15 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 8, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 2.3.4 and 3.5 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on January 10, 2001. |

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| February 14, 2001 | In a meeting summary (signed by S. Koenick), NRC summarized the January 4, 2001, meeting with FPL to discuss staff questions and potential requests for additional information (RAIs) for the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| February 16, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 2.3.3.8, 2.4.2.8, and 2.4.2.10 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on January 17, 2001. |
| February 26, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Section 2.3.3.14 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on January 24, 2001. |
| March 1, 2001 | In a letter (signed by R. Newton), the Westinghouse Owners Group (WOG) submitted for staff review topical report WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants (MUHP-6110)." |
| March 22, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Section 3.4 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| March 22, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 2.1, 2.3.1, 2.3.2.2, 2.3.3.3, 2.3.3.4, 2.4.1, and 2.4.2.4 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| March 30, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 3.7, 4.4, and Appendix B, 3.2.6 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on January 17, 2001. |
| March 30, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Section 3.3 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| March 30, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Section 3.6 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| March 30, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs regarding severe accident mitigation alternatives for Turkey Point Nuclear Plant, Units 3 and 4, requested on January 31, 2001. |

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| April 12, 2001 | In a letter (signed by R. Anand), NRC requested that WOG provide additional information (RAI) on WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants." |
| April 19, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Section 3.2 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| April 19, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 4.2, 4.7.1, and Appendix B, Sections 3.1.5, 3.1.6, 3.1.7, 3.2.1.1, 3.2.2, 3.2.3, 3.2.4, 3.2.9, 3.2.11, 3.2.12, 3.2.13, 3.2.14, and 3.2.16 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 1, 2001. |
| April 19, 2001 | In a letter (signed by R. Hovey), FPL provided its response to the NRC RAIs on Sections 4.3, 4.5, 4.7.4, and Appendix B Sections 3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.2.1.2, 3.2.1.3, 3.2.1.4, 3.2.5, 3.2.8, 3.2.10, and 3.2.15 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| April 24, 2001 | In a meeting summary (signed by S. Koenick), NRC summarized the March 20, 2001, meeting with FPL to discuss draft responses to requests for additional information (RAIs) for the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| April 25, 2001 | In an audit Report (signed by T. Quay), NRC issued "Turkey Point Units 3 and 4 License Renewal Application — Scoping/Screening Methodology and Quality Assurance Attribute Audit Report (TAC NOS. MA9939 and MA9943)." |
| May 2, 2001 | In a meeting summary (signed by R. Auluck), NRC summarized the January 24, 2001, meeting with FPL to discuss staff questions and potential requests for additional information (RAIs) for the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| May 3, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to NRC RAI 2.1-2 on Section 2.1 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |
| May 11, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to the NRC RAIs on Section 3.7 and Appendix B, Section 3.2.6 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on January 17, 2001. |
| May 11, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to an NRC RAI on Section 3.6 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 2, 2001. |

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| May 29, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to an NRC RAI on Section 4.2 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 1, 2001. |
| May 29, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to the NRC RAIs on Sections 3.7 and 4.4 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on January 17, 2001. |
| June 15, 2001 | In a letter (signed by R. Bryan), WOG provided its response to the NRC RAI on WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants." |
| June 25, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to an NRC RAI on Appendix B, Section 3.1.7 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on February 1, 2001. |
| July 18, 2001 | In a letter (signed by R. Hovey), FPL provided its supplemental response to the NRC RAIs on Sections 2.3.3.10 – 12 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA requested on December 22, 2000. |
| July 23, 2001 | In a letter (signed by H. Christensen), NRC issued its Inspection Report Nos. 50-250/01-09 and 50-251/01-09 documenting the results of its scoping and screening inspection. |
| July 31, 2001 | In a letter (signed by R. Bryan), WOG provided its revised response to the NRC RAI on WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants." |
| August 13, 2001 | In a letter (signed by T. Jones), FPL provided a clarification to its RAI 3.2.3-3 response provided in its April 19, 2001, letter on Section 3.2 of the Turkey Point Nuclear Plant, Units 3 and 4, LRA. |
| August 17, 2001 | In a letter (signed by D.B. Matthews), NRC issued its "Safety Evaluation Report With Open Items Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4." |
| August 29, 2001 | In a letter (signed by R. Auluck), NRC issued a correction to the NRC's transmittal letter of August 17, 2001. |
| September 25, 2001 | In a transcript (issued by N. Gross, court reporter), NRC issued its official transcript of the Advisory Committee for Reactor Safety Plant License Renewal Subcommittee Meeting for the Turkey Point Units 3 and 4 LRA. |
| October 29, 2001 | In a letter (issued by H. Christensen), NRC issued Inspection Report Nos. 50-250/01-11, and 50-251/01-11regarding the |

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| | inspection of the Turkey Point facility as it relates to the FPL's application for license renewal for the Turkey Point Units 3 and 4. |
| October 30, 2001 | In a letter (issued by C.I. Grimes), NRC provided additional clarification to FPL regarding its regulatory position for aging management of concrete. |
| November 1, 2001 | In a letter (signed by J.P. McElwain), FPL provided "Turkey Point Units 3 and 4. . . . License Renewal Safety Evaluation Report Open Item and Confirmatory Item Responses and Revised License Renewal Application Appendix A." |
| November 7, 2001 | In a letter (signed by J.P. McElwain), FPL provided "Turkey Point Units 3 and 4. . . . License Renewal Safety Evaluation Report Open Item Regarding Aging Management of Concrete." |
| November 8, 2001 | In a letter (signed by R. Auluck), NRC provided FPL with a revised schedule for the NRC's review for the Turkey Point, Units 3 and 4, LRA. |
| December 17, 2001 | In a letter (signed by J P. McElwain), FPL provided "Turkey Point Units 3 and 4. . . . License Renewal Safety Evaluation Report Open Item and Confirmatory Item Responses and Revised License Renewal Application Appendix A." |
| February 1, 2002 | In a memorandum (signed by B.S. Mallett), Acting Regional Administrator, Region II, provided his recommendations regarding the license renewal for the Turkey Point Units 3 and 4. |
| February 15, 2002 | In a meeting summary (signed by R. Auluck), NRC summarized the October 4, 2001, meeting with FPL to discuss the open items identified in the SER related to Turkey Point Units 3 and 4, LRA. |
| February 27, 2002 | By letter (signed by C.I. Grimes), NRC issued "Safety Evaluation Report Related to the License Renewal of Turkey Point, Units 3 and 4. |

**APPENDIX B**
**REFERENCES**

This appendix contains a listing of references used in preparing the safety evaluation report during the review of the license renewal application (LRA) for Turkey Point, Units 3 and 4, under Docket Nos. 50-250 and 50-251.


## American Concrete Institute (ACI)

ACI 201.2R-77, "Guide for Making a Condition Survey of Concrete in Service."

ACI 201.1R, "Guide for Making a Condition Survey of Concrete in Service."

ACI 318-63, "Building Code Requirements for Reinforced Concrete."

## American Society of Mechanical Engineers (ASME)

ASME Boiler and Pressure Vessel Code Section III, 1965.

ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components.

ASME Section XI as modified by Code Case N-481.

## American Institute of Steel Construction (AISC)

AISC, "Manual of Steel Construction."

## American National Standards Institute (ANSI)

ANSI B31.1, "USA Standard Code for Pressure Piping," 1968.

ANSI Z88.2, "Practices for Respiratory Protection."

ANSI B30.2-1976, "Overhead and Gantry Cranes."

## American Nuclear Society (ANS)

ANS/ANSI Standard N46.2, "Quality Assurance Program Requirements for Post Reactor Nuclear Fuel Cycle Facilities."

## American Society for Testing Materials

ASTM C-295, "Practice for Petrographic Examination of Aggregates for Concrete."

ASTM D-4176, "Standard Test Method for Free Water and Particulate Contamination in Distillate Fuels (Clear and Bright Pass/Fail Procedure)."

ASTM D-1796, "Standard Test Method for Water and Sediment in Fuel Oils by the Centrifuge Method (Laboratory Procedure)."

ASTM A-193, "Specification for Alloy-Steel and Stainless Steel Bolting Materials for High-Temperature Service."

ASTM E-185, "Standard Practice for Conducting Surveillance Tests for Light- Water Cooled Nuclear Power Vessels."

**Babcock and Wilcox**

BAW-1543A, Revision 2, "Master Integrated Reactor Vessel Surveillance Program."

BAW-1543, Revision 4, including Supplements 1 and 2, "Master Integrated Reactor Vessel Surveillance Program."

BAW-2178, "Low Upper Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level C and D Service Loads."

BAW-2312, Revision 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."

BAW-2325, "Reactor Vessel Working Group, Response to RAI Regarding Reactor Pressure Vessel Integrity."

**Electric Power Research Institute (EPRI)**

EPRI TR-102134-R5, "PWR Secondary Water Chemistry Guidelines."

EPRI TR-105714-R4, "PWR Primary Water Chemistry Guidelines."

EPRI TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," December 1997.

EPRI NP-5769, "Degradation of Bolting in Nuclear Power Plants," April 1988.

EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments."

EPRI NP-1558, "A Review of Equipment Aging Theory and Technology."

EPRI NSAC-202L-R2, "Recommendations for Effective Flow-Accelerated Corrosion Program."

## Florida Power and Light (FPL)

### *Correspondence*


Letter from R.E. Uhrig (FPL) to J.P. O'Reilly (NRC), "Turkey Point Unit 4, Docket No. 50-250 [sic], IE Bulletin 82-02," July 15, 1983.

Letter from J.W. Williams, Jr. (FPL) to J.P. O'Reilly (NRC), "Turkey Point Unit 3, Docket No. 50-250, IE Bulletin 82-02," March 9, 1984.

Letter from J.W. Williams (FPL) to D.G. Eisenhut (NRC), "Application for Amendment to Licenses DPR-31 and DPR-41, Combining Reactor Materials Surveillance Programs at Both Units into a Single Integrated Program," February 8, 1985.

Letter from R.J. Hovey (FPL) to NRC, "Revised Pressure-Temperature (P/T) Curves, and Cold Overpressure Mitigation System (CMOS) Setpoints," July 7, 2000.

Letter from R.J. Hovey (FPL) to NRC, "Supplemental Response to Request for Additional Information for the Review of the Turkey Point Units 3 and 4 License Renewal Application," May 29, 2001.

Letter from R.S. Kundalkar (FPL) to NRC, "St. Lucie Units 1 and 2 and Turkey Point Units 3 and 4, Docket Nos. 50-335, 50-389, 50-250, and 50-251, Response to NRC Bulletin 2001-01," September 4, 2001.


### *Turkey Point Nuclear Plant, Unit 3 & 4, Plant Procedures and Technical Products*


ENG-QI 5.3, Revision 2, "License Renewal System/Structure Scoping," March 29, 1999.

ENG-QI 5.4, Revision 2, "License Renewal Screening," March 29, 1999.

ENG-QI 5.5, Revision 4, "License Renewal Aging Management Review," April 21, 2000.

ENG-QI 5.6, Revision 4, "License Renewal Time Limited Aging Analysis," February 24, 2000.

PTN-ENG-LRSP-99-0063, Revision 2, "License Renewal System/Structure Scoping Report," October 30, 2000.

PTN-ENG-LRSC-99-0037, Revision 3, "License Renewal Screening Results Summary Report – Structures and Structural Components," November 27, 2000.

PTN-ENG-LRSC-99-0049, Revision 3,"License Renewal Screening Results Summary Report – Containment Structure and Internal Structural Components," August 15, 2000.

FPL Document Package No. 25, Rev. 4, "Samuel Moore Cables."

### *Reports*

Florida Power & Light (FPL) Topical Quality Assurance Report.

Turkey Point Unit 3 and 4 Updated Final Safety Analysis Report.

### *Submittals*

Florida Power and Light Company Application for Renewed Operating Licenses — Turkey Point Units 3 and 4, September 8, 2000.

## Institute of Electrical and Electronics Engineers, Inc. (IEEE)

IEEE Std. 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

IEEE Std. 334-1974, "Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations."

## National Fire Protection Agency (NFPA)

NFPA 10, "Portable Fire Extinguishers."

NFPA 14, "Standpipe and Hose Systems."

NFPA 25, "Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems."

## Nuclear Electric Insurance Limited (NEIL)

NEIL Property Loss Prevention Standard

## Nuclear Energy Institute

NEI submittal of December 11, 1998, "Responses to the NRC Requests for Additional Information on Generic Letter 97-01," [3-248].

NEI/MRP submittal of Topical Report TP-1001491, Part 2, "PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for U.S. PWR Plant (MRP-44)," [3-247].

NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule," Revision 1, January 2000.

NEI 97-06, "Steam Generator Program Guidelines."

## Sandia National Laboratories

SAND 93-7070, "Aging Management Guideline for Commercial Nuclear Power Plants — Heat Exchangers"

SAND 96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants — Electrical Cable and Terminations," Sandia National Laboratories for the U.S. Department of Energy, September 1996.

## U.S. Nuclear Regulatory Commission (NRC)

### *Statements of Consideration (SOC)*

60 *Federal Register,* No. 88, "Nuclear Power Plant License Renewal: Revisions," pp 22461 – 22495.

### *Bulletins (BL)*

NRC BL 79-01B, "Guidelines for Evaluation Environmental Qualification of Class IE Electrical Equipment in Operating Reactors," January 14, 1980.

NRC BL 79-17, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants," July 26, 1979.

NRC BL 80-11, "Masonry Wall Design," May 8, 1980.

NRC BL 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," June 2, 1982.

NRC BL 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," June 22, 1988.

NRC BL 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988.

NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification," December 20, 1988.

NRC BL 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," August 3, 2001.

### *Circular*

NRC Circular 76-06, "Stress Corrosion Cracks in Stagnant, Low-Pressure Stainless Piping Containing Boric Acid Solution at PWRs," November 22, 1976.

### *Code of Federal Regulations*

10 CFR 50.34, "Contents of Application; Technical Information," Section (a)(1).

10 CFR 50.48, "Fire Protection."

10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

10 CFR 50.55a, "Codes and Standards."

10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light-Water Nuclear Power Reactors for Normal Operation."

10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

10 CFR 50.63, "Loss of All Alternating Current Power."

10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

10 CFR 50.71, "Maintenance of Reports, Making of Reports."

10 CFR 50.120, "Training and Qualification of Nuclear Power Plant Personnel."

Appendix A to 10 CFR Part 50, "General Design Criteria."

Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements."

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements."

Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."

Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

10 CFR Part 100, "Reactor Site Criteria."

***Correspondence with FPL***

Letter From A. Schwencer (NRC) to R. Uhrig (FPL) concerning "Flooding Safety Evaluation Report," September 4, 1979.

Letter from L. Raghavan (NRC) to J.H. Goldberg (FPL), "Turkey Point Units 3 and 4 – Review of Babcock and Wilcox Owners Group Materials Committee Reports – Upper-Shelf Energy," October 19, 1993.

Letter from R. Croteau (NRC) to J.H. Goldberg (FPL), Turkey Point Units 3 and 4 – Generic Letter (GL) 92-01, Revision 1, Reactor Vessel Structural Integrity, May 9, 1994.

Letter from R. Croteau (NRC) to J.H. Goldberg (FPL), Turkey Point Units 3 and 4 – Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping, June 23, 1995.

Turkey Point Units 3 and 4 Operating License Amendment 191/185, September 25, 1996.

Letter from K. Jabbour (NRC) to T. Plunkett (FPL), "Generic Letter 97-01, 'Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations': Review of the Responses for the Turkey Point Plant, Units 3 and 4," January 27, 2000.

Letter from K. Jabbour (NRC) to T. Plunkett (FPL), "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Boron Credit in the Spent Fuel Pool (TAC Nos. MA7262 and MA7263)," July 19, 2000.

Letter from K. Jabbour (NRC) to T. Plunkett (FPL), "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Pressure-Temperature Limits and Cold Overpressure Mitigation System Requirements (TAC Nos. MA9500 and MA9502)," October 30, 2000.

Letter from K. Jabbour (NRC) to T. Plunkett (FPL), "Turkey Point Plant, Unit 3 – Relief Request Regarding Safety Evaluation of Risk-Informed Inservice Inspection Program (TAC No. MA8111)," November 30, 2000.

Letter from K. Jabbour (NRC) to J.A. Stall (FPL), "Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles' Responses for the Turkey Point Plant, Units 3 and 4," November 14, 2001.

***Correspondence Other***

Letter from D. Matthews (NRC) to J. Taylor (Framatome Technologies), "Babcock and Wilcox Owners Group (B&WOG) Reactor Vessel Working Group Report BAW-1543, Revision 4, Supplement 2, Supplement to the Master Integrated Reactor Vessel Surveillance Program (TAC No. M98089)," July 11, 1997.

Letter from C.I. Grimes (NRC) to D. Walters (NEI), "Guidance on Addressing GSI 168 for License Renewal," Project 690, June 2, 1998.

Letter from C.I. Grimes (NRC) to D. Walters (NEI), License Renewal Issue No. 98-0013, Degradation-Induced Human Activities, June 5, 1998.

Memorandum from A. Thadani to W. Travers, "Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," December 26, 1999.

Letter from C.I. Grimes (NRC) to D. Walters (NEI), License Renewal Issue No. 98-0012, Consumables, March 10, 2000.

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NRC GL 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," May 2, 1989.

NRC GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," July 18, 1989,

NRC GL 91-17, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants'," October 17, 1991.

NRC GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," May 18, 1995.

NRC GL 96-04, "Boraflex Degradation in Spent Fuel Pool Storage Racks," June 26, 1996.

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NRC IN 87-44, "Thimble Tube Thinning in Westinghouse Reactors," September 16, 1987.

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NRC IN 93-61, "Excessive Reactor Coolant Leakage Following a Seal Failure in a Reactor Coolant Pump or Reactor Recirculation Pump," August 9, 1993.

NRC IN 93-84, "Determination of Westinghouse Reactor Coolant Pump Seal Failure," October 20, 1993.

NRC IN 93-90, "Unisolatable Reactor Coolant System Leak Following Repeated Application of Leak Sealant," December 1, 1993.

NRC IN 96-32, "Implementation of 10 CFR 50.55a(g)(6)(ii)(A), 'Augmented Examination of Reactor Vessel'," June 5, 1996.

NRC IN 97-31, "Failures of Reactor Coolant Pump Thermal Barriers and Check Valves in Foreign Plants," June 3, 1997.

NRC IN 97-88, "Experiences During Recent Steam Generator Inspections," December 16, 1997.

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NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

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NUREG-1437, Volume 1, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," May 1996.

NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," August 1995.

NUREG-1705, "Safety Evaluation Report Related to the License Renewal of Calvert Cliffs Nuclear Power Plant, Units 1 and 2," December 1999.

NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," March 2000.

NUREG-1739, "Analysis of Public Comments on the Improved License Renewal Guidance Documents," July 2001.

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NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

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# APPENDIX C
# ABBREVIATIONS

| | |
|---|---|
| A/C | air conditioning |
| ABVS | auxiliary building ventilation system |
| ACI | American Concrete Institute |
| ACRS | Advisory Committee on Reactor Safeguards |
| AMP | aging management program |
| AMR | aging management review |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| ASME | American Society of Mechanical Engineers |
| ASTM | American Society for Testing and Materials |
| ATWS | anticipated transient without scram |
| | |
| B&W | Babcock and Wilcox |
| BL | bulletin |
| BTP | branch technical position |
| | |
| CASS | cast austenitic stainless steel |
| CBVS | control building ventilation system |
| CCW | component cooling water |
| CCSRVS | computer/cable spreading room ventilation system |
| CFR | *Code of Federal Regulations* |
| CLB | current licensing basis |
| CMAA | Crane Manufacturers Association of America |
| CRDM | control rod drive mechanism |
| CRVS | control room ventilation system |
| CS | condensate system |
| CST | condensate storage tank |
| CUF | cumulative usage factor |
| CVCS | chemical and volume control system |
| | |
| DBD | design-basis document |
| DCEIRVS | dc equipment/inverter room ventilation system |
| DG | draft regulatory guide |
| DOR | Division of Operating Reactors |
| DWST | demineralized water storage tank |
| | |
| ECCS | emergency core cooling system |
| ECT | eddy current testing |
| EDG | emergency diesel generator |
| EDGB | emergency diesel generator building |
| EDGBVS | emergency diesel generator building ventilation system |
| EER | electrical equipment room |
| EERV | electrical equipment room ventilation |
| EFPD | effective full-power day |
| EFPY | effective full-power year |
| EOL | end of life |

| | |
|---|---|
| EPRI | Electric Power Research Institute |
| EQ | environmental qualification |
| ESF | engineered safety features |
| FAC | flow-accelerated corrosion |
| FP | fire protection |
| FPL | Florida Power and Light Company |
| FSAR | final safety analysis report |
| FSER | final safety evaluation report |
| GALL | generic aging lessons learned |
| GEIS | generic environmental impact statement |
| GL | generic letter |
| GSI | generic safety issue |
| HEPA | high-efficiency particulate air (filter) |
| HVAC | heating, ventilation, and air conditioning |
| IASCC | irradiation-assisted stress-corrosion cracking |
| IEB | Inspection and Enforcement Bulletin |
| IEEE | Institute of Electrical and Electronics Engineers |
| IGSCC | intergranular stress-corrosion cracking |
| IN | information notice |
| INPO | Institute of Nuclear Power Operations |
| IPA | integrated plant assessment |
| ISI | inservice inspection |
| ITS | improved technical specification |
| LBB | leak-before-break |
| LOOP | loss of offsite power |
| LRA | license renewal application |
| MCRE | main control room environment |
| MFS | main feedwater system |
| MIC | microbiologically influenced corrosion |
| MRV | minimum required value |
| NDE | nondestructive examination |
| NEI | Nuclear Energy Institute |
| NEPA | National Environmental Policy Act |
| NRC | Nuclear Regulatory Commission |
| NUREG | NRC technical report designation |
| PLL | prescribed lower limits |
| PTS | pressurized thermal shock |
| PWR | pressurized-water reactors |
| PWSCC | primary water stress-corrosion cracking |
| QA | quality assurance |

| | |
|---|---|
| RAI | request for additional information |
| RCP | reactor coolant pump |
| RCS | reactor coolant system |
| RG | regulatory guide |
| RHR | residual heat removal |
| RI-ISI | risk-informed ISI |
| RPV | reactor pressure vessel |
| RT | reference temperature |
| RVHPIP | reactor vessel head Alloy 600 penetration inspection program |
| SC | structure and component |
| SCC | stress-corrosion cracking |
| SER | safety evaluation report |
| SFP | spent fuel pool |
| SI | safety injection |
| SOC | statement of considerations |
| SPCS | steam and power conversion systems |
| SRP | standard review plan |
| SSC | structure, system, and component |
| TBVS | turbine building ventilation system |
| TEMA | Tubular Exchanger Manufacturers Association |
| TLAA | time-limited aging analyses |
| TS | technical specification |
| UFSAR | updated final safety analysis report |
| USE | upper-shelf energy |
| UT | ultrasonic testing |
| VHP | vessel head penetration |
| WCAP | Westinghouse Owners Group generic technical report |
| WOG | Westinghouse Owners Group |

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This document is a safety evaluation report regarding the application to renew the operating licenses for Turkey Point Units 3 and 4, which was filed by the Florida Power and Light Company by letter dated September 8, 2000 and received by the NRC on September 11, 2000. The Office of Nuclear Reactor Regulation has reviewed the Turkey Point Units 3 and 4, license renewal application for compliance with the requirements of Title 10 of the Code of Federal Regulations, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings.

In its submittal of September 8, 2000, the Florida Power and Light Company requested renewal of the Turkey Point, Units 3 and 4 operating licenses (License Nos. DPR-31 and DRP-41, respectively), which were issued under Section 104b of the Atomic Energy Act of 1954, as amended, for a period of 20 years beyond the current license expiration dates of July 19, 2012 and April 10, 2013, respectively. The Turkey Point, Units 3 and 4 are located in Miami-Dade County east of Florida City, Florida. Each unit consists of a Westinghouse pressurized-water reactor nuclear steam supply system designed to produce a core thermal power of 2300 megawatts or approximately 693 net megawatts electric.

The NRC Turkey Point Units 3 and 4 license renewal project manager is Rajender Auluck. Dr. Auluck may be contacted by calling 301-415-1025 or by writing to the License Renewal and Environmental Impacts, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

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