Florida Power & Light Company, 9760 SW 344 St., Fla City, FL 33035

FPL

L-2001-75
10 CFR 54

APR 19 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Response to Request for Additional Information for the Review of the Turkey Point Units 3 and 4 License Renewal Application

By letter dated February 2, 2001, the NRC requested additional information regarding the Turkey Point Units 3 and 4 License Renewal Application (LRA). Attachment 1 to this letter contains the responses to the Requests for Additional Information (RAIs) associated with Appendix B, Aging Management Programs, Section 4.3, Metal Fatigue, Section 4.5, Containment Tendon Loss of Prestress, Section 4.6, Containment Liner Plate Fatigue, and Subsection 4.7.4 Crane Load Cycle Limit of the LRA.

Should you have any further questions, please contact E. A. Thompson at (305)246-6921.

Very truly yours,

R. J. Hovey
Vice President - Turkey Point

RJH/EAT/hlo

Attachment

A084

cc: U.S. Nuclear Regulatory Commission, Washington, D.C.

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Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251

Response to Request for Additional Information for the Review of the Turkey Point Units 3 and 4, License Renewal Application

STATE OF FLORIDA)
) ss
COUNTY OF MIAMI-DADE)

R. J. Hovey being first duly sworn, deposes and says:

That he is Vice President - Turkey Point of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

R. J. Hovey

Subscribed and sworn to before me this

19th day of April, 2001.

Olga Hanek

OLGA HANEK
Name of Notary Public (Type or Print)

OLGA HANEK
MY COMMISSION # CC 926970
EXPIRES: June 18, 2004
Bonded Thru Notary Public Underwriters

R. J. Hovey is personally known to me.

ATTACHMENT 1
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION DATED FEBRUARY 2, 2001 FOR THE REVIEW OF THE TURKEY POINT UNITS 3 AND 4, LICENSE RENEWAL APPLICATION

AGING MANAGEMENT PROGRAMS

RAI 3.8.1-1:
You state in the scope of this inspection program that it is intended to be a one time inspection of an oil cooler for one of the three shared auxiliary feedwater pumps. Provide justification for only inspecting the oil cooler of one of the three pumps. Also, provide justification for doing a one time inspection instead of multiple inspections with intervals of three or five years, as is generally prescribed in ASME Section XI programs for similar components.

FPL RESPONSE:
The three auxiliary feedwater (AFW) pump oil coolers are identical units and are subjected to the same internal environments and operating conditions. Therefore, the condition of one cooler is representative of all three coolers.

The one-time inspection will provide confirmatory information on the condition of the coolers. Although Turkey Point operating experience has not identified graphitic corrosion degradation of these coolers, the materials of construction and environment make them potentially susceptible to such degradation. This corrosion mechanism is not anticipated due to the quality of the water in the AFW System, and thus, a one-time inspection was selected. The results of the inspection will be evaluated to determine if further inspections are warranted. If significant loss of material is detected, the appropriate corrective action, including program revision, if needed, will be taken per the FPL 10 CFR 50 Appendix B Corrective Action Program.

RAI 3.8.1-2:
Provide the bases for the quantitative acceptance criteria which will be used to make the determination that inspection of other coolers and additional future monitoring are required.

FPL RESPONSE:
The Auxiliary Feedwater (AFW) Pump Oil Cooler Inspection Program, as described in LRA Appendix B, Subsection 3.1.1 (page B-10), consists of a confirmatory one-time inspection of one AFW oil cooler to verify that loss of material due to graphitic corrosion is not occurring. In the event significant loss of material is detected during this inspection, appropriate corrective actions will be established per FPL's 10 CFR 50 Appendix B Corrective Action Program. Evaluation of inspection results will consider the minimum required wall thickness for the component and a corrosion allowance. Follow-up inspections, if required, will be scheduled based upon actual corrosion rates or inspection findings.

RAI 3.8.2-1:
In Section 3.1.2 of Appendix B of the application you state that the auxiliary feedwater steam piping inspection program will provide for representative volumetric examinations to detect loss of material in the auxiliary feedwater steam piping between the steam supply check valves and each of the three auxiliary feedwater pump turbines. Provide a detailed description of how samples will be selected for the examination and the basis of the selection.

FPL RESPONSE:
The samples will be selected based on susceptibility to both internal and external loss of material which is based on the location, exposure to adverse conditions, and plant operating experience. For example, samples will be identified in areas where the piping is exposed to rain water and collection or ponding of the rainwater is expected. Other examples of selection criteria are also described in the response to RAI 3.8.2-2. Based upon inspection findings from the most susceptible locations, additional inspection points will be selected, as required.

RAI 3.8.2-2:
In Table 3.5-3 of the application you list piping/fittings, auxiliary feedwater pump turbine casings, valves, and steam traps as the in-scope components to be managed for aging effects by the Auxiliary Feedwater Steam Piping Inspection Program. In Section 3.1.2 of Appendix B to the application, under the Scope, you only list piping and fittings as components to be managed by the program. Clarify the above discrepancy, and discuss in detail how the program will be credited for aging management of the loss of material for the auxiliary feedwater pump turbine casings, valves, and steam traps, by addressing each of the pertinent review elements.

FPL RESPONSE:
As described in LRA Appendix B, Subsection 3.1.2 (page B-12), the Auxiliary Feedwater Steam Piping Inspection Program is credited for managing loss of material due to internal and external corrosion on auxiliary feedwater (AFW) steam piping between the steam supply check valves and each of the three AFW pump turbines. This includes any valves and steam traps located in these lines. As discussed in LRA Appendix B, Subsection 3.1.2, this program selects representative locations for inspection based upon the potential for exposure to a wetted environment. This includes sections of lines where water can accumulate, such as at the bottom of horizontal pipe runs and areas of contact with the lower section of wetted insulation. Having the least wall thickness, piping and fittings are considered the limiting components, and thus, the primary inspection points. However, where significant loss of material due to corrosion is detected, valves and steam traps would be inspected as required. Table 3.5-3 (pages 3.5-17 and 3.5-20) of the LRA inadvertently identifies internal and external loss of material as an aging effect requiring management for the AFW turbine casings, and credits the Auxiliary Feedwater Steam Piping Integrity Program for aging management. The aging management review of the AFW turbine casing demonstrated that loss of material is not an aging effect requiring management based on the following:

- The turbine casings are heavy-wall constructed components.
- There is low back pressure during operation.
- The casings are continuously drained precluding accumulation of water.
- The turbine casings have no external collection points or places where water would be expected to pond.

This has been confirmed by an inspection of an AFW turbine casing, after 17 years of operation, which showed no significant signs of corrosion. Therefore, LRA Table 3.5.3 (page 3-17) will be revised as follows to incorporate this correction.

Component /Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
INTERNAL ENVIRONMENT					
Auxiliary feedwater pump turbine casings	Pressure boundary	Carbon steel	Treated water - secondary Air/Gas	None	None required

Component /Commodity Group	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
EXTERNAL ENVIRONMENT					
Auxiliary feedwater pump turbine casings	Pressure boundary	Carbon steel	Outdoor	None	None required

RAI 3.8.3-1:
Provide a justification for your determination that a one time inspection of the emergency containment coolers is adequate. Operating experience with these coolers at other nuclear power plants indicates that loss of material caused by erosion and flow induced vibration can vary during plant operation due to unanticipated transients and flow conditions. It seems that a one time inspection would not be able to reliably detect damage over a long period of operation.

FPL RESPONSE:
As stated in Appendix B, Subsection 3.1.3 (page B-14) of the LRA, the aging effect requiring management for the Emergency Containment Coolers (ECCs) is loss of material due to erosion on the inside surface of the cooler tubes. Cracking due to flow induced vibration is not an aging effect requiring management as discussed in response to RAI 3.8.3-4. LRA Subsection 4.7.2 (page 4.7-2) provides the current licensing basis TLAA for the Emergency Containment Coolers (ECCs) tubes and demonstrates that minimum wall will be maintained for the existing operating period of 40 years. The analysis used conservative erosion rates and actual wall loss is expected to be less. Except for surveillance testing, the ECCs are normally not in operation and have minimal cooling water flow through the tubes (see UFSAR Section 6.3.2, page 6.3-6). Therefore, the tubes are not susceptible to unanticipated transients. The results of the inspection will be evaluated to determine an actual erosion rate to verify that the minimum required wall thickness for the ECC tubes will be maintained during the period of extended operation. As stated in Appendix B, Subsection 3.1.3 (page B-14) of the LRA, the evaluation of the inspection results may determine the need for additional testing, monitoring, and trending.

RAI 3.8.3-2:
You plan to only examine a percentage of the components within the scope of the program. You stated that you will choose those areas of greatest susceptibility based on geometry and flow conditions for the initial inspection sample. Provide the percentage of components that will be examined for the inspection, compared to the entire population under consideration. This information is needed for us to determine whether the results of a limited sample size may be considered representative of, and therefore applied to, the entire population.

FPL RESPONSE:
As stated in Appendix B, Subsection 3.1.3 (page B-14) of the LRA, a sample of tubes for examination will be selected based on geometry and flow conditions that represent those with the greatest susceptibility to erosion. All six Emergency Containment Coolers (ECCs) (3 in each unit) are identical and are subjected to the same cooling water conditions. Additionally, the ECCs are in service (during testing) approximately the same amount of time. On this basis, one ECC will be selected for inspection as representative of all six. The number of tubes to be inspected in this cooler will be in accordance with the reference below.

Reference: Squeglia, N. L "Zero Acceptance Number Single Sampling Plans", American Society for Quality Control, 1986 (third edition).

RAI 3.8.3-3:
Discuss the acceptance criteria which you will use for tube examination in the emergency containment coolers inspection program. You stated that you will verify that the minimum required wall thickness will be maintained. Clarify the source and basis for the acceptance criteria to be applied for this examination.

FPL RESPONSE:
The acceptance criteria for the Emergency Containment Cooler (ECC) tubes is minimum wall thickness plus margin based upon actual erosion rate. The minimum wall thickness for the ECC tubes is based on the coolers' design pressure as calculated per ASME Section III, Class C. As stated in Appendix B, Subsection 3.1.3 (page B-14) of the LRA, the results of the inspection will be evaluated to verify that the minimum required wall thickness for the ECC tubes will be maintained during the period of extended operation.

RAI 3.8.3-4:
Discuss how the acceptance criteria for the emergency containment cooler heat exchanger tubes consider fatigue failure due to flow induced vibration. If flow induced vibration is not considered, provide the technical justification for this position.

FPL RESPONSE:
As stated in Appendix B, Subsection 3.1.3 (page B-14) of the LRA, the aging effect requiring management is loss of material due to erosion on the inside surface of the ECC tubes. The acceptance criteria for this inspection is described in response to RAI 3.8.3-3 above. As discussed in LRA Appendix C, Section 5.2 (page C-20), vibration induced fatigue is fast acting and typically detected early in the component's life and corrective actions initiated to prevent recurrence. A review of Turkey Point operating experience for the ECCs did not indicate the presence of flow-induced vibration degradation conditions. Therefore, cracking due to mechanical fatigue is not an aging effect requiring management for the ECCs.

RAI 3.8.4-1:
The Field Erected Tanks Internal Inspection (FETII) Aging Management Program (AMP), described in Section 3.1.4 of Appendix B in the application is a one-time inspection of the condensate storage tanks, the refueling water storage tanks, and the shared demineralized water storage tank. Justify your use of a one-time inspection rather than periodic examinations for each of these tanks.

FPL RESPONSE:
The Condensate Storage Tanks (CSTs), the Refueling Water Storage Tanks (RWSTs), and the Demineralized Water Storage Tank (DWST) are not currently inspected on a periodic basis. The Unit 4 CST was internally inspected and recoated in 1983. The Unit 3 CST was internally inspected, several 1/16" pits were weld repaired, and the tank was recoated in 1991. The need for recoating activities was attributed to operational practices and the original coatings being inadequate for the application, and both have been corrected. A review of plant specific operating experience revealed no other incidences of internal degradation for these tanks. In addition, periodic inspections of the external surfaces of these tanks have confirmed the tanks to be in good condition. Therefore, no significant aging is expected. The proposed one-time inspection of each tank serves to confirm that there are no aging effects requiring management. As stated in LRA Appendix B, Subsection 3.1.4 (page B-16), the inspection results will be evaluated to determine if any additional actions are needed.

RAI 3.8.4-2:
For each of the tanks to be examined as part of the FETII AMP, describe the locations within each of the tanks that are the most susceptible to corrosion and discuss why these locations are the most susceptible.

FPL RESPONSE:
The Field Erected Tanks Internal Inspection, as described in LRA Appendix B, Subsection 3.1.4 (page B-16), considers all locations within each of the tanks to be susceptible to corrosion. Previous inspections of the Unit 4 CST in 1983 and the Unit 3 CST in 1991 revealed corrosion at some of the welds at the roof to wall connection and coating degradation at several areas in the floor and wall of the tank. However, these conditions were attributed to operational practices and the original coatings being inadequate for the application, both of which have been corrected. Therefore, all accessible internal surfaces of the tanks will be visually inspected rather than focusing on limited select locations suspected of being "most susceptible to corrosion".

RAI 3.8.4-3:
Describe the visual examination procedures for the FETII AMP, including any lighting and resolution requirements. Also describe any provisions for additional volumetric or surface examinations in the event that the scheduled visual examination reveals extensive loss of material.

FPL RESPONSE:
As indicated in LRA Appendix B, Subsection 3.1.4 (page B-16), the visual inspection will consist of direct or remote means. Lighting and resolution requirements necessary to accomplish the internal inspection will be established and described in the implementing procedure to be utilized at the time of implementation. The internal surfaces will be examined for evidence of flaking, blistering, peeling, discoloration, pitting, or excessive corrosion.

If visual examination reveals significant loss of material, the condition would be resolved through the FPL 10 CFR 50 Appendix B Corrective Action Program, which may involve volumetric or surface examinations.

RAI 3.8.4-4:
Describe the relationship between the Chemistry Control AMP and the FETII AMP for the refueling water storage tanks, the demineralized water storage tank, and the condensate storage tanks. How do these programs interact?

FPL RESPONSE:
As described in LRA Appendix B, Subsection 3.2.4 (page B-47), the Chemistry Control Program manages aging for the internal surfaces of primary and secondary systems and structures (including the CSTs, RWSTs, and DWST) by controlling the internal environment water quality to preclude or minimize the underlying aging mechanisms that result in aging effects.

As described in LRA Appendix B, Subsection 3.1.4 (page B-16), the Field Erected Tank Internal Inspection will determine the extent of internal corrosion of the subject tanks. This one time visual inspection will determine if the Chemistry Control Program alone is adequate to manage aging of the tank internals, or if additional actions are necessary, such as periodic inspections.

RAI 3.9.1.2-1:
In describing preventive actions, you state that coatings, cathodic protection, and moisture barriers are not credited in the determination of the aging effects requiring management. However, it is the degradation of coating and moisture barriers and malfunction of the cathodic protection system that could give rise to the degradation of the protected safety related components. That is the reason Subsection IWE requires periodic examination of moisture barriers, and coatings. The effectiveness of these preventive measures should be periodically assessed as part of the aging management program for the protected components. Provide a summary of the procedures used for managing the effectiveness of these preventive measures.

FPL RESPONSE:
For clarification, moisture barriers located at the interface of the Containment liner and concrete floor are credited in the determination of aging effects for the containment liner plate (see LRA Table 3.6-2, page 3.6-52). Consequently, aging management is provided by the ASME Section XI, Subsection IWE Inservice Inspection Program, as described in LRA Appendix B, Subsection 3.2.1.2 (page B-30). Waterproofing membranes and waterstops are not credited in the determination of aging effects requiring management. See the response to RAI 2.4.1-1 (FPL Letter L-2001-49, dated March 22, 2001) for a discussion of Turkey Point waterproofing membranes and waterstops.

FPL recognizes the protective benefits of coatings and cathodic protection. Existing plant procedures ensure these protective measures are effective. However, these protective measures do not perform a license renewal intended function as defined in 10 CFR 54.4(a)(1), (2), and (3) and they are not credited in the determination of aging effects requiring management for protected structures and components. Therefore, coatings and cathodic protection do not require aging management review, and thus, do not require aging management.

RAI 3.9.1.2-2:
With respect to the detection of aging effects, the bottom liner plate of the containment structure at Turkey Point is covered with fill concrete, therefore, it is not feasible to perform direct examinations. Borated water leakage and thermal and shrinkage related cracking of the fill concrete could give rise to corrosion of the bottom liner plate. Describe any program, whether as part of the IWE ISI or as part of the maintenance rule programs, to detect the degradation and aging effects of the bottom liner plate. If no such programs exist, explain how you concluded the bottom liner plate is not susceptible to such degradation.

FPL RESPONSE:
The Turkey Point containment structures have bottom liner plates that are embedded in the concrete with no exposed surfaces. The 18-inch thick concrete over the bottom liner protects the steel from corrosion. Containment concrete components are constructed of dense, well-cured concrete consistent with the guidance provided in ACI 201.2R-77. The concrete was designed in accordance with ACI 318-63. The aggregates were tested in accordance with ASTM C295. The concrete over the bottom liner is not normally exposed to an aggressive environment. These features ensure concrete cracking is minimized. Consequently, the concrete over the containment liner plate provides adequate protection of the inaccessible portions of the liner plate. In addition, a moisture barrier is provided that prevents intrusion of moisture between the concrete and the inaccessible liner surfaces.

As noted in the response to RAI 3.6.1.2-4 (FPL Letter L-2001-61, dated March 30, 2001), inaccessible areas are managed by visually examining accessible areas of in-scope structures and other relevant structures for conditions that could indicate the presence of degradation to such inaccessible areas (see response to RAI 3.6.1.1-1). As indicated in Table 3.6-2 (page 3.6-52), the moisture barriers are managed by the ASME Section XI, Subsection IWE Inservice Inspection Program (see response to RAI 3.6.1.2-4), as described in LRA Appendix B, Subsection 3.2.1.2 (page B-30). Additionally, when events occur such as borated water leaks, potential degradation of inaccessible structures is evaluated as part of the Corrective Action Program. Finally, the containment liner plate is periodically pressure tested in accordance with the ASME Section XI, Subsection IWE, Inservice Inspection Program (Category E-P), described in LRA Appendix B, Subsection 3.2.1.2 (page B-30).

Based on the design features and programs discussed above, there is reasonable assurance that the containment liner plate will continue to perform its intended function throughout the period of extended operation.

RAI 3.9.1.2-3:
How does your process provide for the confirmation of the adequacy of repairs. Do you require reexamination of the repaired areas during subsequent inspection interval(s)?

FPL RESPONSE:
As detailed in the ASME Section XI, Subsection IWE, Inservice Inspection Program, LRA Appendix B, Subsection 3.2.1.2 (page B-30), Confirmation Process, when areas of degradation are identified, an evaluation is performed to determine if repair or replacement is required. The results are documented in accordance with the Corrective Action Program. Reexaminations are conducted for repaired flaws or areas of degradation to demonstrate that the repairs meet the acceptance standards.

RAI 3.9.1.2-4:
Based on the inspections performed prior to the implementation of Subsection IWE, provide a summary of significant events at Turkey Point related to:

1. Liner corrosion.

2. Major penetration leakage (equipment hatches, airlocks, main steam line, feedwater line) not meeting the Type B leakage rate requirements.

3. Leakage and corrosion of bellows (if applicable).

4. Isolation valve leakage (system or Type B test).

5. Type A tests not meeting the containment leak rate criteria.

Include the corrective actions taken to prevent such events in the future.

FPL RESPONSE:
Containment leak-tight verification and visual examination of the steel components that are part of the leak-tight barrier have been conducted at Turkey Point since initial startup. Prior to the development of the ASME Section XI, Subsection IWE, Inservice Inspection Program, described in LRA Appendix B, Subsection 3.2.1.2 (page B-30), examinations were performed in accordance with 10 CFR 50, Appendix J. Appendix J requires that licensees provide for pre-operational and periodic testing of the leak-tight integrity of the primary reactor containment, systems, and components that penetrate the containment. Appendix J requires that after the pre-operational leakage rate test is conducted, a set of three Type A tests (to provide a measure of reactor containment overall leakage rate) be conducted at equal intervals during each 10-year service period. The Appendix J tests performed at the Turkey Point units during the years of operation have not shown any loss of intended function of the containment steel components that were attributed to loss of material or other aging effects.

The following provides a summary of some significant events at Turkey Point related to:

1. Liner corrosion:

   Minor surface corrosion of the liner plate and penetration canisters has been observed on several occasions. The corroded areas were documented, evaluated, cleaned, and re-coated.

2. Major penetration leakage:

   LER 251-87-007 documents air leakage during leak rate testing of the personnel hatch not meeting the Technical Specification definition of containment integrity due to a misaligned operating linkage for an equalizing valve. The valve was replaced and the operating linkage realigned, and the airlock leak test was performed satisfactorily.

3. Leakage and corrosion of bellows:

   This item is not applicable because Turkey Point does not have containment penetration bellows.

4. Isolation valve leakage:

   A review of plant operating experience did not identify any significant events related to containment isolation valve leakage (system or Type B test). Identified leakage in excess of established limits for each penetration is addressed under the 10 CFR 50 Appendix B Corrective Action Program.

5. Type A tests not meeting the containment leak rate criteria:

   A review of plant operating experience did not identify any Type A tests not meeting the containment leak rate criteria.

The ASME Section XI, Subsection IWE, Inservice Inspection Program incorporates the inspection criteria and guidelines from the 10 CFR 50, Appendix J inspection program. In addition, the program provides for enhanced visual inspections of the containment steel components. Based on this information, continued examinations performed under the guidance of the ASME Section XI, Subsection IWE, Inservice Inspection Program provide reasonable assurance that the aging effects, loss of material and change in material properties, for the containment steel components will be managed for the extended period of operation.

RAI 3.9.1.3-1:
Section 3.2.1.3 of Appendix B of the application states that the extent and frequency of the IWF in-service inspection program of component supports is in accordance with ASME Section XI, Subsection IWF. Provide a description of the extent and frequency of the inspections, including sample selection. Also specify the edition and addenda of the ASME Code used for the program.

FPL RESPONSE:
The extent and frequency of inspections performed for the ASME Section XI, Subsection IWF Inservice Inspection Program, as described in LRA Appendix B, Subsection 3.2.1.3 (page B-34), are sufficient to ensure that aging effects are managed to ensure the supports' intended functions will not be compromised. The scope of examination performed during each 10-year interval includes 25% of Class 1 piping supports, 15% of Class 2 piping supports, and 10% of Class 3 piping supports, plus numerous unique supports other than piping supports.

The Third Ten Year Inservice Inspection Interval for both units is being conducted in accordance with the 1989 Edition of ASME Section XI (no addenda).

RAI 3.9.1.4-1:
In this section of the application, you discuss the aging management of the containment post-tensioning system components. However, Subsection IWL of Section XI of the ASME Code also requires the in-service inspection of the concrete and the post-tensioning system. If Subsection IWL is not used for aging management of the containment concrete, provide a description of the program for managing the aging of the containment concrete, including, the inspection interval, the personnel qualification requirements, the examination method(s), the acceptance criteria, and the quality assurance requirements related to this program.

If ASME Subsection IWL is used for managing the aging effects on concrete, but inadvertently left out in the description, supplement the description of the in-service inspection requirements for concrete. As ASME Subsection IWL does not contain specific acceptance criteria for examination of concrete, incorporate your criteria as part of this revised description.

FPL RESPONSE:
10 CFR 54.21(a)(3) requires that the effects of aging are adequately managed, so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation. Thus, aging effects that could cause a loss of intended function require aging management. The analysis of possible aging effects for containment reinforced concrete is summarized in LRA Subsection 3.6.1.1 (page 3.6-2). The analysis is based on concrete material properties, the applicable environments, and years of operating experience. Although the Turkey Point operating experience does include several Containment concrete aging effects, such as items documented in Appendix A of NUREG-1522 (e.g., scaling of the Unit 3 containment dome and discoloration at the junction of the Unit 3 dome and the ring girder), these aging effects were evaluated in accordance with the Corrective Action Program, as appropriate, and determined to be insignificant with no impact on intended functions. The analysis concludes that there are no aging effects that could cause a loss of intended function for containment concrete above groundwater. Therefore, no aging management programs are required for these components.

These concrete structures are inspected as part of the existing ASME Section XI, Subsection IWL Inservice Inspection Program, as described in LRA Appendix B, Subsection 3.2.1.4 (page B-37), and the existing structural inspections required by 10 CFR 50.65. These inspections have also confirmed that there are no aging effects requiring management for above groundwater concrete structures.

However, as described in the response to RAI 3.6.2.1-2 (FPL Letter L-2001-61, dated March 30, 2001), FPL proposes to modify the ASME Section XI, Subsection IWL Inservice Inspection Program to include aging management of Containment reinforced concrete above groundwater.

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." The program stipulates that concrete surfaces shall be subject to the acceptance criteria described in ASME Code Section IWL-3211. Concrete surfaces with examination results that do not meet the acceptance standards of IWL-3211 are subject to acceptance by evaluation and/or repair.

RAI 3.9.1.4-2:
You state that all metallic components are interconnected to an impressed current cathodic protection system (CPS) to prevent galvanic corrosion, and that this system 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. Considering the reliability of continuous sources, the CPS may or may not be effective at certain times (power outage, low battery voltage, etc.). Such incidents could lead to adverse effects to 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. Provide a summary of the procedures used to assess the effectiveness of the CPS.

FPL RESPONSE:
As described in the FPL response to RAI 3.9.1.2-1, FPL recognizes the protective benefits of cathodic protection. Existing plant procedures ensure cathodic protection systems are effective. However, these cathodic protection systems do not perform a license renewal intended function as defined in 10 CFR 54.4(a)(1), (2), or (3) and they are not credited in the determination of aging effects requiring management for protected structures and components. Therefore, cathodic protection does not require aging management review and thus, does not require aging management.

RAI 3.9.1.4-3:
Based on the inspections performed prior to the implementation of ASME Subsection IWL, as part of the plant's operating experience, provide a summary of significant events at Turkey Point related to:

1. Containment concrete (e.g. dome delamination, wide spread scaling).

2. Containment prestressing force (unusual systematic losses).

3. Corrosion of post-tensioning system hardware (breakage of wires or anchor-head components, water in the sheathing).

4. Grease leakage through concrete.

Include the corrective actions taken to alleviate such events in the future. Also, provide a description of the condition of the tendon galleries' environments and the measures implemented to control the environment and to alleviate the corrosion of vertical tendon anchorage hardware.

FPL RESPONSE:

The following provides a summary of operating experience at Turkey Point related to:

1. Containment Concrete:

   During original plant construction, the containment dome experienced delamination, which was reworked prior to initial plant start-up. However, the delamination was not age related.

   As noted in NUREG-1522, discoloration due to poor drainage and minor scaling have been observed on the concrete dome. These conditions were evaluated and determined to be insignificant with no impact on license renewal intended functions. The discoloration was corrected by removing debris blocking the roof drains and the minor scaling was evaluated to be insignificant and required no corrective action.

2. Containment Pre-stressing Forces:

   Containment pre-stressing forces have been periodically measured at Turkey Point. Results during the 20th year tendon surveillance and a subsequent containment re-analysis are described in the Turkey Point UFSAR, Section 5.1.3, Containment Design Analyses.

3. Corrosion of Post-tensioning System Hardware:

Minor corrosion of tendon hardware has been observed outside the Containments on several occasions. In these cases, the conditions were evaluated, accepted, cleaned, and re-coated.

Free water has been detected inside tendon sheathing; however, the resulting effects have not been significant. Lean concrete was added at the top of vertical tendons to eliminate ponding around the tendon caps.

Tendon wire corrosion and breakage was evaluated in response to IN 99-010 and determined not to be significant for Turkey Point.

4. Grease Leakage through Concrete:

There has been no grease leakage through concrete documented in the plant operating experience.

RAI 3.9.5-1:
In Section 3.2.5, Appendix B of the application you state that surveillance procedures require the closure of a second isolation valve in the containment spray headers when the pumps are started for testing. Identify the test procedures and the basis for determining the effectiveness of the preventive action.

FPL RESPONSE:
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. As a result, the aging management review for the containment spray headers identified loss of material as an aging effect requiring management and credited the Containment Spray System Piping Inspection Program, as listed on LRA Table 3.3-2 (page 3.3-12), for Component/Commodity Group "Valves, Piping/fittings (downstream of containment penetrations to Containment elevation 65')".

RAI 3.9.5-2:
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.

FPL RESPONSE:
Flow induced vibrations are 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.

RAI 3.9.5-3:
According to the information in Section 3.3.1 of the LRA, all piping/fitting joints in the Containment Spray System are accessible. However it is not clear whether or not they are accessible to ultrasonic test (UT) examinations. For those joints which may not be accessible to (UT) examinations provide additional information to describe the management of the aging effects.

FPL RESPONSE:
Per LRA Table 3.3-2 (page 3.3-12), the piping/fitting joints that require examination via the Containment Spray System Piping Inspection Program, as described in LRA Appendix B, Subsection 3.2.5 (page B-50), are located downstream of the containment penetrations to containment elevation 65'. All piping/fittings required to be examined are accessible to perform ultrasonic thickness measurements. This is an existing program and examinations have been previously performed on susceptible Containment Spray piping/fittings.

RAI 3.9.5-4:
Previous license renewal applicants identified loss of material and cracking due to stress corrosion as an issue for stainless steel components in this system. Discuss the differences in design, construction or operation of this system at Turkey Point that explain why the scope of your program is limited to loss of material for carbon steel components.

FPL RESPONSE:
As listed in LRA Table 3.3-2 (pages 3.3-11 through 3.3-15), the aging effects requiring management for stainless steel surfaces exposed to treated water and treated water-borated are loss of material and fouling (seal water heat exchanger tubes). The aging management program credited for managing these aging effects is the Chemistry Control Program as described in LRA Appendix B, Subsection 3.2.4 (page B-47).

As discussed in LRA Appendix C, Section 5.2 (page C-19), 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 through-wall 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 pre-service 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 upon the above, cracking due to SCC was determined not to be an aging effect requiring management for Containment Spray.

RAI 3.9.8-1:
In Section 3.2.8, Appendix B of the application you state that the scope of the fire protection program will be enhanced to include additional components. Provide the bases and guidelines which are to be used for selection of these additional components and indicate which additional components will be considered.

FPL RESPONSE:
As described in LRA Appendix C, Section 5.2 (page C-18), 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 LRA Appendix B, Subsection 3.2.8 (page B-56), 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 having aging effects requiring management under the Fire Protection Program are currently included in the scope of this program.

RAI 3.9.8-2:
Identify the specific programs which are credited for monitoring external and internal material degradation of the fire protection system components and piping.

FPL RESPONSE:
As identified in LRA Table 3.4-14 (page 3.4-71), the Galvanic Corrosion Susceptibility Inspection Program and the Fire Protection Program are credited for managing the aging effects of loss of material and cracking for internal and external surfaces of fire protection components and piping. These programs are described in LRA Appendix B, Subsections 3.1.5 (page B-18) and 3.2.8 (page B-56), respectively.

RAI 3.9.8-3:
Identify the specific fire protection procedures which specify the acceptance criteria for evaluating the inspection and test results of the components/piping. Also identify the applicable documents which list the parameters required to be monitored and controlled.

FPL RESPONSE:
Below is a list of the procedures included in the Fire Protection Program. These procedures specify the parameters to be monitored and/or controlled and the acceptance criteria.

- Fire Protection Program
- Electrical Raceway Protection Inspection
- Fire Barriers And Structural Steel Fireproofing Inspection
- Fire Protection Equipment Surveillance
- Fire Door Inspection
- Electrical Manhole Inspection
- Diesel Fire Pump Engine 18 Month Maintenance Inspection
- Fire Suppression System Annual Flush
- Spray/Sprinkler System Inspection
- Turbine Lube Oil Reservoir Fire Suppression System 18 Month Functional Test
- Main Transformer Fire Suppression System 18 Month Functional Test
- Startup Transformer Fire Suppression System 18 Month Functional Test
- C Bus Transformer Fire Suppression System 18 Month Functional Test
- Auxiliary Transformer And Hydrogen Seal Oil Unit Fire Suppression System 18 Month Functional Test
- Open Head Spray/Sprinkler 3 Year Air Flow Test
- Fire Main 3 Year Hydraulic Gradient Flow Test
- Fire Main Post Indicator Valve (PIV) Leak/Flow Path Valve Surveillance Test And System Flush
- Fire Protection Impairments

As requested at the FPL/NRC public meeting held on April 12, 2001, provided below is additional clarification regarding inspection and testing of sprinkler systems.

Per UFSAR Appendix 9.6A, Turkey Point's current licensing basis does not include 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 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 per 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 that 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 by movement of a tell-tale at the opening of each sprinkler head/spray nozzle in the system. Each spray nozzle is also visually inspected for obstruction.

Additionally, the "functionality test" procedures listed above actuate the system, including any spray nozzles to ensure functionality.

Based on feedback from the NRC Staff at the April 12, 2001 meeting, and open items identified on previous license renewal applications, Turkey Point proposes to perform testing of wet pipe sprinkler heads following the guidance of NFPA 25 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, Subsection 16.2.8 (page A-37) and Appendix B, Subsection 3.2.8 (page B-56).

RAI 3.9.10-1:
In Section 3.2.10, Appendix B of the application you state that the intake cooling water system program will be enhanced to improve documentation of the scope and the frequency of the inspections. Clarify whether or not the scope and frequency of the inspections will be enhanced or simply the documentation aspects of the existing inspection programs will be changed.

FPL RESPONSE:
The scope and frequency of inspections will not change nor be enhanced. The scope and frequency are based on years of operating and maintenance history. Program implementation has determined that the scope and frequency are effective in managing aging effects of the affected components. Therefore, only documentation will be enhanced.

RAI 3.9.10-2:
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 heat exchangers. Also provide the acceptance criteria and the bases for the evaluation of the inspection results.

FPL RESPONSE:
The below listed procedures monitor, test and inspect the heat exchangers.

- Component Cooling Water Heat Exchanger Performance Monitoring
- Component Cooling Water Heat Exchanger Performance Test
- Component Cooling Water Heat Exchanger Cleaning

Acceptance criteria are provided to ensure design basis and technical specification requirements for heat transfer capability are maintained. Guidelines are provided for cleaning, inspecting, and testing the heat exchangers.

RAI 3.9.15-1:
With respect to the attribute related to the scope of systems and structures monitoring program, indicate how you will manage aging effects of structural components that are inaccessible for inspection. Discuss how you intend to manage or monitor aging effects of inaccessible structural components when conditions in accessible areas may not indicate the presence of degradation in inaccessible areas. Also, provide a summary discussion of specific program attributes that will be enhanced to address inspection requirements to manage certain aging effects pursuant to 10 CFR Part 54.

FPL RESPONSE:
Aging management of structural components that are inaccessible for inspection is accomplished by inspecting accessible structural components with similar materials and environments for aging effects that may be indicative of aging effects for inaccessible structural components. This is described in the Systems and Structures Monitoring Program, LRA Appendix B, Subsection 3.2.15 (page B-83). Since components in inaccessible areas have the same materials and environments as those in accessible areas, indications of degradation or the lack of indications in accessible areas is an effective way to manage components in inaccessible areas.

As described in the response to RAI 3.6.1.1-1 (FPL Letter L-2001-61, dated March 30, 2001), the Systems and Structures Monitoring Program is credited for managing aging of inaccessible containment concrete below groundwater. Aging effects are managed by performing visual inspections of the non-safety related tendon access gallery concrete below groundwater to provide early indication of potential aging effects for the containment concrete.

Currently, inspections that are within the scope of the Systems and Structures Monitoring Program are performed under a variety of plant programs and processes. For the renewal term, FPL plans to enhance these inspections by restructuring them to identify certain aging effects in accordance with 10 CFR 54, by adding specific structures and components not currently inspected under an existing program, and by improving documentation requirements. These enhancements will be incorporated prior to the end of the initial license term for Turkey Point, as described in LRA, Appendix A, Subsection 16.2.15 (page A-41).

RAI 3.9.15-2:

With respect to the attribute covering parameter monitored or inspected, the parameter description is incomplete. Augment the discussion to demonstrate that the specific parameters monitored or inspected are selected to ensure that aging degradation leading to loss of intended functions will be detected and to what extent the degradation can be determined. The parameters monitored or inspected must be commensurate with industry standard practice and, must also consider industry and plant specific operating experience. For concrete structural elements, typical parameters to be monitored or inspected are structural cracking, spalling, scaling, erosion, corrosion of reinforcement bars, settlements and deformation. For structural steel elements (including connections), typical parameters to be monitored or inspected are corrosion, cracking, erosion, discoloration, wear, pitting, gouges, dents, and other signs of surface irregularities. Augment and enhance this section of the plant-specific program to include sufficient details on the parameters monitored or inspected so that a staff technical audit can reach a conclusion that this program attribute is adequate.

FPL RESPONSE:

The Systems and Structures Monitoring Program, as described in LRA Appendix B, Subsection 3.2.15 (page B-83), manages the aging effects of loss of material, cracking, fouling, loss of seal, and change in material properties to ensure that aging degradation leading to loss of intended functions will be detected. The program provides for periodic visual inspection of concrete and masonry structures, steel structures, and system commodities and components (e.g., piping, ductwork, electrical raceway, valves, heat exchangers and electrical enclosures).

The parameters monitored are selected based on industry and plant experience to ensure that aging degradation that could lead to loss of intended function will be identified and addressed. Concrete and masonry parameters monitored include exposed rebar, cracking, rust bleeding, spalling, scaling, other surface irregularities, and settlement. Steel structure parameters monitored include corrosion, flaking, pitting, gouges, cracking, other surface irregularities, and missing parts. System commodity and component parameters include corrosion, flaking, pitting, gouges, cracking, fouling, other surface irregularities, protective coating degradation on select stainless steel pipe welds, leakage at limited locations, and missing parts.

RAI 3.9.15-3:
For the Systems and Structures Monitoring Program presented, provide additional description of the criteria for assessing or categorizing the overall condition of the structures and systems that are monitored. Also discuss Turkey Point specific criteria that are used to assess the severity of observed degradations and determine whether corrective action(s) are needed for the observed degradations. As applicable, briefly describe walkdown procedures, checklists, or inspection forms that are provided to personnel that implement "Systems and Structures Monitoring Program."

FPL RESPONSE:
Detailed structural and system/equipment material condition inspections are performed in accordance with approved plant procedures. Existing procedures include detailed guidance for inspecting and evaluating the material condition of systems, structures, and components within the scope of the program. The guidance includes specific parameters to be monitored and criteria to be used for evaluating identified degradation. In addition, the procedures provide sample forms to be used to document the analysis and the assessment, and a system checklist for documenting relevant information from a system walkdown. Material condition is assessed based on the parameters monitored as described in the response to RAI 3.9.15-2.

Conditions identified through the Systems and Structures Monitoring Program are evaluated to determine if the condition should be addressed under the FPL 10 CFR 50 Appendix B Corrective Action Program (i.e., deficient or unacceptable conditions). For example, the criterion for structural steel is loss of material exceeding 1/32" and the criterion for piping is any corrosion greater than uniform light surface corrosion.

RAI 3.9.15-4:
With respect to the monitoring and training (trending sic) aspects of the program, your discussion does not appear to specifically address the monitoring part. Pro-active monitoring and understanding of trending behavior is needed to monitor structural aging so that corrective actions can be taken prior to exceeding the acceptance criteria. Describe the monitoring and analysis activities that are to be included for each of the commodity groups to track the extent and rate of degradation and their relationship to the acceptance criteria in the program. If you do not plan a monitoring and trending attribute for your program, justify your conclusion.

FPL RESPONSE:
The Systems and Structures Monitoring Program is primarily credited for managing loss of material due to corrosion, as well as other aging effects identified in LRA Appendix B, Subsection 3.2.15 (page B-84). Monitoring is accomplished through detailed system and structure material condition inspections, performed periodically in accordance with approved plant procedures. When degraded conditions are identified, they are evaluated and corrected via the Corrective Action Program. Typically, this involves quantifying the extent of the condition, evaluating the capability of the structure or component to perform its intended function, and then designating appropriate corrective actions.

The Corrective Action Program includes periodic trending assessments and evaluations. When trends are identified, they are addressed under the Corrective Action Program. Further evaluation is performed including identification and implementation of programmatic improvements, as required. Programmatic improvements may include adjustment of program scope, frequency, acceptance criteria, and/or corrective actions. This process ensures that applicable aging effects are adequately managed.

RAI 3.9.15-5:
The discussion in the detecting of aging effects does not provide enough information for the staff to reach a reasonable assurance finding. Provide the inspection methods, inspection schedule (frequency), and inspector qualifications for each structure/aging effect combination to ensure that aging degradation will be detected and quantified before there is loss of intended functions. Describe the method(s) used to determine the frequency of inspections as well as the minimum walkdown frequency for the various applications of the structural and systems/component walkdowns. Also, the program description does not provide information about the training and qualifications of the personnel that (1) perform the inspections required by the program including structures and systems/components walkdowns and (2) evaluate the adequacy of the inspection/walkdown procedures and findings.

FPL RESPONSE:
As described in LRA Appendix B, Section 3.2.15 (page B-83), the Systems and Structures Monitoring Program employs the visual inspection method. Structures and structural commodities are visually inspected on an area basis, and system commodities and components are visually inspected on a system basis. Conditions documented and evaluated via the Corrective Action Program may employ other methods, such as volumetric examination, to determine the extent of degradation.

The inspection schedule varies depending on the system, structure, or component being inspected. Generally, inspections will be performed on a frequency of five years or less; however, as documented in the response to RAI 3.4.1-2 (FPL Letter L-2001-50, dated March 22, 2001), some inspections of the Intake Cooling Water (ICW) system will be performed on an 18-month frequency. These frequencies are based on Turkey Point plant experience regarding degradation rates and the ability of a structure or component to accommodate degradation without a loss of intended function. The frequency of inspections may be adjusted as necessary based on future inspection results and industry experience.

Personnel responsible for the performance of inspections and evaluation of inspection results are qualified in accordance with the Engineering Training Program (ETP), which is accredited by the Institute of Nuclear Power Operations and required by 10 CFR 50.120.

The inspection methods, inspection schedules, and personnel qualifications described above provide reasonable assurance that aging degradation will be detected and evaluated before there is loss of intended functions.

RAI 3.9.15-6:
The description of operating experience and the demonstration provided is too general and does not contain a description of the findings from the Maintenance Rule baseline inspection and subsequent Maintenance Rule inspection activities. Discuss your actions relating to the treatment of aging you identified prior to the loss of intended function or failures not detected prior to the loss of intended function. In addition, indicate whether these findings have been used to enhance or improve the proposed systems and structures monitoring program.

FPL RESPONSE:
Systems, structures and components have been periodically inspected for material condition at Turkey Point since the mid-1980s. As part of implementation of the Maintenance Rule, baseline inspections were performed in 1996. Periodic inspections continue to be performed as part of this program. Degraded conditions are documented under the Corrective Action Program. As part of the Corrective Action Program, actions to prevent recurrence are identified, such as plant modifications and program enhancements to address the affected item as well as related, generic implications. Additionally, periodic trend evaluations are performed as described in the response to RAI 3.9.15-4 to assess and initiate enhancements to plant programs, including the proposed Systems and Structures Monitoring Program.

TIME LIMITED AGING ANALYSES

SECTION 4.3 METAL FATIGUE

RAI 4.3.1-1:
In Section 4.3.1 of the application you discuss your evaluation of the fatigue time limited aging analyses for ASME Class 1 components. The discussion indicates that, based on your review of the plant operating history, you concluded that the number of cycles assumed in the design of the ASME Class 1 components are conservative and bounding for the period of extended operation. Table 4.1-8 of the Updated Final Safety Analysis Report contains a list of transient design conditions and associated design cycles. Provide the following information for each transient listed in Table 4.1-8:

1. The current number of operating cycles and a description of the method used to determine the number and severity of the design transients from the plant operating history.

2. 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.

FPL RESPONSE:
The following tables provide the results of a review of the plant operating history for Turkey Point Units 3 and 4. The review of actual operating cycles and the projection to 60 years of operation was based on the Turkey Point Fatigue Monitoring Program data compiled through June 8, 1998. Continued operating experience to date has reaffirmed the original review and operating cycle projections. The following tables list the design transients from Table 4.1-8 of the Updated Final Safety Analysis Report, design cycles, actual cycles experienced as of June 8, 1998 and the projected 60-year cycles.

TURKEY POINT UNIT 3 PROJECTED CYCLES AT 60 YEARS OF OPERATION					
TRANSIENT NUMBER	GENERAL TRANSIENT DESCRIPTION	UFSAR TABLE 4.1-8 DESIGN CYCLES	UNIT 3 ACTUAL CYCLES	UNIT 3 PROJECTED CYCLES	% USED AT 60 YEARS
1	Plant Heatup at 100°F/hour	200	90	156	78.00%
2	Plant Cooldown at 100°F/hour	200	89	155	77.50%
3	Unit Loading at 5%/minute	14500	272	2720	18.76%
4	Unit Unloading At 5%/Minute	14500	214	2140	14.76%
5	10% Step Increase of Full Power	2000	43	109	5.45%
6	10% Step Decrease of Full Power	2000	88	220	11.00%
7	50% Step Decrease of Full Power	200	68	167	83.50%
8	Reactor Trip	400	145	291	72.75%
9	Hydrostatic Test at 3107 psi and 100°F	1 (3)	1	1	100.00%
10	Hydrostatic Test at 2435 psi and 400°F	5 (4)	1	3	60.00%
11	Steady State Fluctuation	∞	(1)		
12	Feedwater Cycling at Hot Standby	2000	(2)		

Notes:

1. Not counted, not significant contributor to fatigue usage factor.
2. Not counted. Intermittent slug feeding at hot standby not performed.
3. Limited by Steam Generator Analysis. Represents pre-operational hydrostatic test.
4. Limited by Reactor Coolant Pump Analysis.

TURKEY POINT UNIT 4 PROJECTED CYCLES AT 60 YEARS OF OPERATION					
TRANSIENT NUMBER	**GENERAL TRANSIENT DESCRIPTION**	**UFSAR TABLE 4.1-8 DESIGN CYCLES**	**UNIT 4 ACTUAL CYCLES**	**UNIT 4 PROJECTED CYCLES**	**% USED AT 60 YEARS**
1	Plant Heatup at 100°F/hour	200	101	191	95.50%
2	Plant Cooldown at 100°F/hour	200	100	190	95.00%
3	Unit Loading at 5%/minute	14500	232	2320	16.00%
4	Unit Unloading At 5%/Minute	14500	219	2190	15.10%
5	10% Step Increase of Full Power	2000	45	112	5.60%
6	10% Step Decrease of Full Power	2000	55	123	6.15%
7	50% Step Decrease of Full Power	200	42	110	55.00%
8	Reactor Trip	400	169	337	84.25%
9	Hydrostatic Test at 3107 psi and 100°F	1 (3)	1	1	100.00%
10	Hydrostatic Test at 2435 psi and 400°F	5 (4)	1	3	60.00%
11	Steady State Fluctuation	∞	(1)		
12	Feedwater Cycling at Hot Standby	2000	(2)		

Notes:

1. Not counted, not significant contributor to fatigue usage factor.
2. Not counted. Intermittent slug feeding at hot standby not performed.
3. Limited by Steam Generator Analysis. Represents pre-operational hydrostatic test.
4. Limited by Reactor Coolant Pump Analysis.

The determination of actual operating cycles is part of the Fatigue Monitoring Program as discussed in Appendix B, Subsection 3.2.7 (page B-54) of the LRA. The Fatigue Monitoring Program has been an ongoing program at Turkey Point since initial unit startup and has evolved over many years of operation. The FPL Quality Assurance Department has previously audited this program with no identified findings. In addition, a comprehensive review conducted by Westinghouse concluded that the Fatigue Monitoring Program accurately identifies and classifies plant design cycles, and provides an effective and consistent method for categorizing, counting, and tracking design cycles. This assessment also included a review of the original design transient assumptions to determine if they bound all operating events. The assessment considered normal events such as plant heatup and cooldown, unit loading and unloading, 10% step load increase/decrease, and large step decrease. Upset conditions, such as reactor trip, loss of load, partial loss of flow and loss of AC power were also included in this severity review. The assessment concluded that the design cycle severity bounds actual plant operation.

With the exception of plant heatup, plant cooldown, and reactor trip transients, the Turkey Point projected number of cycles at 60 years is based on the mean frequency of occurrence on a per year basis through June 1998. This methodology provides a very conservative prediction of future cycles, in that the units typically experience a greater number of transients during the early years of operation. For the plant heatup and plant cooldown transients, it was determined that using the data for all years of operation through June 1998 results in overly conservative projections. Therefore, the mean frequency on a per year basis was derived from the more recent operating history (i.e., last ten years of operation) and was used as the basis for determining future cycles. Similarly, the frequency of occurrence of reactor trips at Turkey Point was higher at the beginning of life, compared to a much lower frequency after the units have operated for a number of years. It is estimated that the number of trips in the middle of unit life will be similar to the number of trips that have occurred during recent plant operating history. Conservatively, at the end of life an increase in the number of reactor trips is assumed. Therefore, reactor trip projections for the time interval between the current year and 50 years of operation are derived from the last ten years of operation. Between 50 and 60 years of operation, a similar number of trips to those that occurred during the first ten years of operation are assumed. In conclusion, for each of the transients listed in Table 4.1-8 of the UFSAR, the projected number of cycles at the end of the period of extended operation will be less than originally specified.

RAI 4.3.1-2:
Flaws in ASME Class 1 components that exceed the size of allowable flaws defined in IWB-3500 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 cracking mechanisms, or both, where applicable, during a specified evaluation period. 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. Provide the results of the analyses that indicate whether the flaws will satisfy the criteria in IWB-3600 for the period of extended operation.

FPL RESPONSE:
Currently there are no identified flaws in any Class 1 component that exceed the allowable flaw limits defined in IWB-3500. Hence, there are no analytical evaluations considering justification of any such flaws for the existing operating period nor the period of extended operation.

RAI 4.3.1-3:
Indicate whether calculations that meet the definition of a time limited aging analyses were performed in response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," and NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." Describe the actions to be taken to address these bulletins during the period of extended operation.

FPL RESPONSE:
Review of Turkey Point Units 3 and 4 documentation and correspondence regarding NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems", identified no calculations that meet the definition of a Time-Limited Aging Analysis (TLAA) as defined in 10 CFR 54.3. Turkey Point activities associated with NRC Bulletin 88-08, including nondestructive examination and hydrodynamic flow testing, demonstrated that thermal transients from in-leakage is unlikely to occur on either Turkey Point Unit 3 or 4. As documented by NRC letter dated September 23, 1991, FPL was advised that the requirements of NRC Bulletin 88-08 have been met and no further action is required. As such, there are no additional actions to be taken during the period of extended operation to address the considerations of NRC Bulletin 88-08.

As indicated in LRA Subsection 4.3.1 (page 4.3-2), the fatigue analysis of the Turkey Point Units 3 and 4 pressurizer surge lines was identified as a TLAA as defined in 10 CFR 54.3. The specific analysis reviewed for the pressurizer surge lines was the analysis performed in response to NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification". An evaluation of the pressurizer surge lines analysis determined the analysis to remain valid for the period of extended operation, in accordance with 10 CFR 54.21 (c)(1)(i).

Two programs have been credited in the Turkey Point LRA to address the fatigue analysis of the pressurizer surge lines. As discussed in LRA Subsection 4.3.1, continuation of the Fatigue Monitoring Program into the period of extended operation will assure that the pressurizer surge lines fatigue design cycle limits are not exceeded. The Fatigue Monitoring Program is considered a confirmatory program and is described in LRA Appendix B, Subsection 3.2.7 (page B-54). In addition, as stated in LRA Subsection 4.3.5 (page 4.3-7), Turkey Point has selected aging management to address pressurizer surge line reactor water environmental effects during the period of extended operation. The potential for crack initiation and growth, including reactor water environmental effects, is adequately managed during the extended period of operation by the Turkey Point ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program, described in LRA Appendix B, Subsection 3.2.1.1 (page B-27).

RAI 4.3.1-4:
In Section 3.2.3 "Pressurizers" of the application you state that Westinghouse report WCAP-14754 [sic] is not incorporated in the application. However, in Subsection 2.3.1.4 "Pressurizers" and Section 3.2.3 you also state that the Turkey Point 3 & 4 pressurizers are bounded by the description of the pressurizer in WCAP-14574 with regard to design criteria and features, modes of operation, intended functions, and environments/exposures. Table 2-10 of WCAP-14574 indicates that, based on current licensing basis fatigue calculations, the ASME Boiler & Pressure Vessel Section III Class 1 fatigue cumulative usage factor (CUF) criterion (CUF≤1.0) will be exceeded for several pressurizer subcomponents in less than the extended period of operation. We conclude that this is also applicable to the Turkey Point 3 & 4 pressurizers.

1. Show the ASME Section III Class 1 current licensing basis CUFs for the applicable subcomponents of Turkey Point 3 & 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 lists other off-normal and additional transients in Section 3.8.3, and recently discovered surge line inflow/outflow thermal transients described in Section 3.8.4. These thermal cyclic transients were not considered in the current licensing basis fatigue analyses of Westinghouse pressurizers, including Turkey Point 3 & 4. Provide the highest CUFs considering these transients for the following pressurizer subcomponents, for the extended period of operation:

   Surge nozzle
   Lower head region
   Heater wells
   Support skirt and flange

3. Describe the aging management programs that will be used to manage fatigue of the Turkey Point 3 & 4 pressurizer subcomponents for the extended period of operation, considering the transients listed above and environmental effects on fatigue.

FPL RESPONSE:
1. The FPL approach to addressing fatigue of the Turkey Point pressurizers for the period of extended operation is different than that in WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers". As stated in

Subsection 3.2.3.2.1 (page 3.2-19) of the LRA, cracking of the pressurizers due to fatigue is identified as a Time-Limited Aging Analysis (TLAA) and is addressed in Subsection 4.3.1 (page 4.3-2) of the LRA. Subsection 4.3.1 provides justification that the existing pressurizer component design cycles and cycle frequencies are conservative and bounding for the period of extended operation. As such, the Current Licensing Basis (CLB) cumulative usage factors (CUFs) for the Turkey Point pressurizers (provided below) are considered conservative and bounding for the extended period of operation.

COMPONENT	CLB CUF
Surge Nozzle	0.5202 (1)
Spray Nozzle	0.8906
Safety and Relief Nozzle	0.148
Lower Head, Heater Well	0.461 (1)
Lower Head Perforation	0.0165
Upper Head and Shell	0.7737 (2)
Support Skirt/Flange	0.0165
Manway Pad	0.0
Manway Cover	0.0
Manway Bolts	0.0
Welded Manway Diaphragm	0.0321
Support Lug	Not Installed
Instrument Nozzle	0.0627
Immersion Heater	0.004
Valve Support Bracket	Not Installed

Notes:

1. CUF value reflects more recent plant-specific analysis to incorporate the effects of pressurizer insurge/outsurge transients, as recommended by the Westinghouse Owners' Group.

2. Calculated fatigue usage factor is based on a conservative assumption that all spray transients will impinge directly on the pressurizer shell.

The CLB CUFs do not include consideration of environmental effects on the fatigue curves. The effects of environmentally-assisted fatigue on the pressurizer components are addressed through three approaches: (1) screening, (2) plant-specific evaluation, or (3) aging management. Each of these approaches is discussed in detail below.

Screening

The effects of environmentally assisted fatigue are a function of several parameters, including material type, temperature, and dissolved oxygen content. These effects on individual CUF load pairs can be as high as a factor of fifteen for stainless steel when all relevant conditions are present. However, it is typical for the overall effects for all load pairs for a given component location to be a factor of four or less, since environmental effects do not affect all individual load pairs (due to thresholds beyond which environmental effects are negligible). A factor of four is considered to be a conservative estimate of the maximum overall impact of environmental effects, and is based on the overall factor of 1.4 to 1.6 obtained from EPRI Report No. TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," plus an additional adjustment multiplier of 2.0 to accommodate more recent laboratory data that was not available at the time of the evaluation contained in EPRI Report No. TR-107515. Further discussion of the applicability of EPRI Report No. TR-107515 to Turkey Point, as well as the additional factor of 2.0 is provided in the response to RAI 4.3.5-1.

Based on the above, if a value of four is conservatively used to establish a screening value for CUF, a threshold CUF value of 0.25 is obtained. Thus, even under the most adverse environmentally assisted fatigue conditions, locations that possess a CUF of less than 0.25 are not expected to exceed allowable usage values during the period of extended operation.

Using a screening value of 0.25, all Turkey Point pressurizer components shown in the table above are eliminated from further consideration, with the exception of the following:

Surge Nozzle
Spray Nozzle
Lower Head, Heater Well
Upper Head and Shell

Note from the above table of pressurizer CUF values, that the use of a screening value of 0.25 actually provides a margin of between six and seven with respect to the CUF allowable of 1.0, since the largest CUF value remaining in the pressurizer after applying the screening criteria is 0.148 (for the safety and relief nozzle).

The spray nozzle, lower head, heater well, and upper head and shell are addressed via a plant specific evaluation, as discussed below. The surge nozzle is addressed by aging management, also as discussed below.

Plant-Specific Evaluation

From a review of the Turkey Point pressurizer analysis, the total spray nozzle CUF of 0.8906 is comprised primarily from the fatigue damage of four transient combinations. These four transient combinations contribute a usage factor of 0.740 to the overall CUF of 0.8906. The four transients are (1) inadvertent auxiliary spray, (2) normal spray above the differential temperature limits allowed by plant procedures during heatup/cooldown, (3) normal spray during heatup/cooldown within differential temperature limits allowed by plant procedures, and (4) normal spray during plant loading and unloading at 5%/minute. Transients (1), (2), and (4) contain excess conservatism and can be adjusted to provide a more representative fatigue damage value for the pressurizer spray nozzle.

Transient (1) consists of ten cycles of inadvertent auxiliary spray, coupled with recovery, assuming a ΔT of 320°F. This contributes a fatigue usage of 0.167. There have been no recorded incidents of inadvertent auxiliary spray in the 28 years of operation of either Turkey Point unit. In addition, there are no projected occurrences of this event through the period of extended operation considering the current Turkey Point operating methodology. Nevertheless, conservatively assuming two cycles of inadvertent auxiliary spray for the remainder of plant life, the CUF contribution for this transient is reduced from 0.167 to 0.033.

Transient (2) consists of 200 cycles of heatup spray with a ΔT of 320°F, coupled with cooldown spray with a ΔT of 405°F. This transient contributes 0.263 to the total usage factor. Administratively, plant procedures prohibit the use of spray with a ΔT greater than 320°F. Adjusting the alternating stress intensity based on a cooldown spray ΔT of 320°F, the allowable number of cycles increases from 760 to 1,750. This reduces the CUF contribution for this transient from 0.263 to 0.114.

Transient (4) consists of 33,360 cycles of a combined spray transient with a ΔT of 160°F, followed by recovery. This transient is representative of the combined cycles from transients such as reactor trip and loss of load, and includes 29,000 cycles due to plant loading and unloading at

5%/minute. From the tables in the response to RAI 4.3.1-1, the maximum projected number of plant loading/unloading cycles for the period of extended operation is 2,720 cycles (plant loading) for Unit 3, and 2,190 cycles (plant unloading) for Unit 4. This reduces the total cycles for consideration from 33,360 to 9,270, which reduces the CUF contribution for this transient from 0.196 to 0.054.

Therefore, utilizing a more realistic cycle count for transients (1) and (4), and reducing the differential temperature assumption for transient (2) to the maximum value allowed by plant procedures, the overall CUF for the pressurizer spray nozzle is reduced from 0.8906 to 0.4656.

Further considerations for a reduction in the calculated CUF can be directed to the original pressurizer spray nozzle fatigue analysis methodology. The original analysis was performed utilizing the computer program Seal Shell 2. This is an axisymmetric plate shell program that did not account for the stress mitigation effects of local flexibility, which leads to conservative stress estimates. Shell programs of this type have since been replaced with finite element programs that are capable of addressing local flexibility, thus leading to reduced and more realistic stress estimates. Moderate stress reductions resulting from improved stress analysis techniques would lead to a significant decrease in fatigue usage. Section 4.2 of Sandia Report No. SAND 94-0187, "Evaluation of Conservatisms and Environmental Effects in ASME Code, Section III, Class 1 Fatigue Analysis," provides evidence of the reductions in stress and CUF values that can result when "interaction models" using Seal Shell 2 methodology are reanalyzed using finite element techniques. Due to geometric similarities, the charging inlet nozzle and the safety injection nozzle evaluated in the Sandia report are most applicable for comparison to the pressurizer spray nozzle at Turkey Point. For the charging nozzle, finite element analysis resulted in a 34% reduction in alternating stress intensity, which caused a 233% increase in the allowable number of cycles for the critical load set pair. The total CUF was reduced by a factor of 2.6 (from 1.9 to 0.73). For the safety injection nozzle, the CUF was reduced by a factor of 8.2 (from 2.273 to 0.278). Thus, reanalysis of the pressurizer spray nozzle with a modern finite element program can be expected to reduce the maximum CUF by more than a factor of two, and thereby reduce the total CUF below the screening criteria of 0.25.

The pressurizer lower head, heater well location is evaluated next. In order to evaluate the influence of reactor water

environmental effects, the very conservative surge line hot leg nozzle environmental multiplier of 4.2 from NUREG/CR-6260 can be applied to the pressurizer lower head, heater well location. This gives an environmental fatigue CUF of 1.94 for the pressurizer lower head. Using the F_{en} approach, with intermittent influence from reactor water environmental effects, this CUF would be reduced slightly. The worst-case F_{en} is 1.9 for surge line calculations from EPRI Report No. TR-107515, and an additional adjustment of 2.0 as discussed in the response to RAI 4.3.5-1 for austenitic stainless steel material, yields an environmental multiplier of 3.8. Applying this factor, the environmental CUF becomes 1.76. Therefore, the environmental fatigue CUF for the pressurizer lower head, heater well location is somewhere between 0.461 (design basis CUF) and 1.94 (CUF using a conservative NUREG/CR-6260 multiplier of 4.2), with an expected value of 1.20 (average of above values). Based on inherent margins in the calculational process, the low risk significance associated with these penetrations, current visual inspections performed on these penetrations as part of the ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program (as described in LRA Appendix B, Subsection 3.2.1.1, page B-27), and the fact that the surge line is significantly more limiting from a fatigue perspective when considering reactor water environmental effects, no additional actions beyond present aging management activities are planned for the heater wells for the period of extended operation.

With respect to the upper head and shell, the original Turkey Point pressurizer analysis conservatively assumed the spray transient impinged directly on the upper shell of the pressurizer. The transient associated with this impingement contributed to almost all of the CUF for the upper head and shell. A study issued by Westinghouse in 1989 established that water droplets from the pressurizer spray nozzle do not impinge on the pressurizer shell at a pressurizer pressure above 320 psig. In accordance with Turkey Point operating procedures, the pressurizer bubble is collapsed and the pressurizer is taken water solid at pressures between 325 and 350 psig. As such, the Westinghouse study is applicable to Turkey Point, and direct spray impingement does not occur in the Turkey Point pressurizers. Without direct impingement, the associated transient is eliminated, and the reported CUF of 0.7737 for the upper head and shell reduces to a negligible value.

Since actual projected cycle counts were utilized in the pressurizer spray nozzle evaluation above, FPL will either: (1) modify the Turkey Point Fatigue Monitoring Program to limit transient accumulations to the values used in the above

evaluation, (2) perform a more refined evaluation for the spray nozzle to show acceptable CUF values for 60 years, or (3) track CUF values in addition to cycle counts to ensure CUF values remain acceptable. Use of one of these three approaches will ensure that the CUFs for the spray nozzle remain valid for the period of extended operation, including the consideration of reactor water environmental effects.

Aging Management

As indicated in Subsection 4.3.5 (page 4.3-7) of the LRA, Turkey Point has selected aging management to address pressurizer surge line fatigue during the period of extended operation. For the pressurizer surge lines, the potential for crack initiation and growth, including reactor water environmental effects, is adequately managed during the period of extended operation by the Turkey Point ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program. The pressurizer surge nozzle is considered to be part of the pressurizer surge line and, as such, is included in the program scope described in Subsection 4.3.5 of the LRA.

2. The CLB CUFs for the Turkey Point pressurizer surge nozzles, lower heads, heater wells, and support skirt/flanges are provided in the table included in the response to question 1 above. These CUFs include consideration of the off-normal and additional transients discussed in Section 3.8.3 of WCAP-14575, as applicable, including specific consideration of insurge/outsurge transients.

3. As stated in the LRA, two programs are credited to manage cracking due to fatigue of pressurizer components during the period of extended operation. As stated in Subsection 4.3.1 (page 4.3-2) of the LRA, continuation of the Turkey Point Fatigue Monitoring Program into the period of extended operation will assure that the pressurizer design cycle limits are not exceeded. The Fatigue Monitoring Program is considered a confirmatory program and is described in LRA Appendix B, Subsection 3.2.7 (page B-54). As stated in LRA Subsection 4.3.5 (page 4.3-7) and in the response to question 1 above, the potential for crack initiation and growth, including reactor water environmental effects, is adequately managed during the period of extended operation by the Turkey Point ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program described in LRA Appendix B, Subsection 3.2.1.1 (page B-27). Also, see the response to RAI 4.3.5-2.

RAI 4.3.5-1:
In Section 4.3.5 of the application you discuss your evaluation of the impact of the reactor water environment on the fatigue life of components. The discussion references the fatigue sensitive component locations for an early vintage Westinghouse plant identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The application indicates that the results of the NUREG/CR-6260 studies were used to scale up the Turkey Point plant-specific usage factors for the same locations to account for environmental effects. The application also indicates that the later environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," were considered in the evaluation. Provide the results of the usage factor evaluation for each of the six component locations listed in NUREG/CR-6260. Discuss how the factors used to scale up the Turkey Point plant-specific usage factors were derived. Also discuss how the later environmental data provided in NUREG/CR-6583 and NUREG/CR-5704 were factored into the evaluations.

FPL RESPONSE:
The older-vintage Westinghouse CUF results in NUREG/CR-6260 are applicable to Turkey Point Units 3 and 4. The description of the "Older Westinghouse Plant" evaluated in NUREG/CR-6260 matches Turkey Point in the design codes used, as well as the analytical approach and techniques used. In addition, the evaluated transient cycles match or bound Turkey Point. The identified actions in NUREG/CR-6260 to reduce the calculated usage factors are also applicable to the Turkey Point component analyses. These locations were evaluated in NUREG/CR-6260 on the basis of ASME Code curves and interim fatigue curves. Table 1 shows the evaluated results for these locations. The column headed by U_{code} is the fatigue usage value computed by the standard ASME Code evaluation as tabulated in the NUREG/CR-6260 evaluation. The column headed by U_{PTN} shows Turkey Point plant-specific fatigue usage values computed for these same locations. The column headed by U_{6260} shows fatigue usage values evaluated with the revised interim environmental fatigue curves for the 40-year period of operation, as tabulated in the NUREG/CR-6260 evaluation.

The calculations reported in NUREG/CR-6260 were based on the interim reduced fatigue design curves given in NUREG/CR-5999, "Interim Fatigue Design Curves for Carbon, Low Alloy, and Austenitic Stainless Steels in LWR Environments". Such an approach penalizes the component location fatigue analysis unnecessarily, because research has shown that a combination of

environmental conditions is required before reactor water environmental effects become pronounced. This research finding would suggest that the use of the NUREG/CR-6260 results is conservative, in that environmental effects were included for all load combinations regardless of whether all relevant conditions that trigger environmental effects were present. However, more recent laboratory fatigue data in simulated reactor water environments have been generated by Argonne National Laboratory (ANL) for carbon and low-alloy steels (published in NUREG/CR-6583), and for austenitic stainless steels (published in NUREG/CR-5704), and these data make such a conclusion premature, as explained in more detail below. Therefore, in order to adequately capture potential environmental effects at Turkey Point, FPL made use of the NUREG/CR-6260 results with a very conservative additional penalty factor to account for the more recent laboratory data. The derivation of these penalty factors for carbon/low alloy steel and austenitic stainless steel is described in the sections that follow.

Revised Environmental Fatigue Methodology for Carbon and Low-Alloy Steels

Laboratory fatigue data in simulated reactor water environments have been generated by ANL for carbon and low-alloy steels, and published in NUREG/CR-6583 in March 1998. These data do not differ substantially from the data used in several EPRI generic studies that evaluated the effects of environmental fatigue (EPRI Report Nos. TR-107515 and TR-110043 for PWRs). However, the change in strain threshold may have a significant effect, so that effect was evaluated.

The following recalculation is based on one of the examples contained in EPRI TR-105759, a BWR carbon steel feedwater piping location with a design-basis fatigue usage factor of 0.1409 for 40 years. An alternating stress threshold of 30 ksi (approximating the alternating strain threshold of 0.10%) was used initially to adjust the incremental fatigue usage for eight out of thirty-one load pairs, giving an additional (environmental) fatigue usage of 0.0477, for a 40-year adjusted total of 0.1886. The overall F_{en} multiplier in this case was 1.38 (1.68 for the eight affected load pairs).

Reducing the alternating stress threshold to 21 ksi (approximating the alternating strain threshold of 0.07%) would require an environmental adjustment for at least six additional load pairs (for a total design fatigue usage of 0.0803 for the fourteen load pairs). Assuming that the F_{en} multiplier of 1.68 would continue to apply for the fourteen affected load pairs, the

estimate for the adjusted fatigue usage factor would be 0.1409 - 0.0803 + 1.68 (0.0803) = 0.1955. The overall F_{en} multiplier increases only to 1.39.

Because the additional load pairs that would have to be included contribute relatively small increments to the total CUF, the change in the strain range threshold does not cause a significant impact on the calculated fatigue usage. Therefore, using the results of NUREG/CR-6260, without further modification (i.e., penalty factor = 1.0), provides a reasonable estimate of the impact of potential environmental fatigue effects for carbon and low alloy steels.

Revised Environmental Fatigue Methodology for Austenitic Stainless Steels

Laboratory fatigue data in simulated reactor water environments have also been generated by ANL for austenitic stainless steels, and published in NUREG/CR-5704 in April 1999. These data are significantly more penalizing than the data used in the EPRI generic studies (EPRI Report Nos. TR-107515 and TR-110043 for PWRs).

As discussed in Section 7 of NUREG/CR-5704, the environmental shift is 2.5 for the case of relatively low temperature (<200°C), all strain rates, and either high or low dissolved oxygen. The environmental shift may be as high as 15 for relatively high temperature (>200°C), low dissolved oxygen (<0.05 ppm), and a low (bounding) strain rate (<0.0004%/sec). The overall environmental shift for a component location would be less than these values, since not all load pairs in the CUF calculation will be affected by environmental conditions.

For most of the component locations evaluated in the EPRI generic studies, these most recent data do not pose a problem for the demonstration that the 60-year CUF is less than 1.0, including reactor water environmental effects. Again, a significant benefit accrues to the F_{en} approach in this regard because most of the penalizing thermal transients lie below the threshold temperature of 200°C. Therefore, the environmental shift is relatively low, provided that separate multipliers are used for the portions of the transient that are above and below 200°C. However, for the most fatigue-sensitive locations, (e.g., surge line elbows), the environmentally-adjusted CUF increases over that calculated in the EPRI generic studies by a factor of about two. Therefore, a conservative approach to accounting for the more recent laboratory data for stainless steel material is to apply an additional penalty factor of 2.0 to the NUREG/CR-6260 results.

Applicability of EPRI Report No. TR-107515

Some of the factors utilized by FPL in the evaluation of environmental effects on CUF are based on the results documented in EPRI Report No. TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant." In that report, the resulting overall environmental factors were based on evaluation of feedwater system, chemical volume and control system, and surge line system components. In particular, results for the surge line components were used to help establish environmental multipliers for stainless steel material.

The use of the pressurizer surge line environmental multipliers from EPRI Report No. TR-107515 are appropriate for application to Turkey Point since the CUF values for the pressurizer surge line are primarily affected by actual events that are very close in magnitude to design events. For example, Section 3.3.2.3 of EPRI Report No. TR-107515 documents that pressurizer heatup and cooldown events drive the pressurizer surge nozzle CUF, along with associated stratification effects. The heatup and cooldown events are carefully controlled operations that do not deviate significantly from the severity assumed in the design evaluation. Similar results are observed for stratification. Therefore, CUF values obtained in the EPRI study do not differ substantially from original design values. Since the Turkey Point CUF evaluations are based on design transient severity, the use of the results for a component primarily affected by design transient severity was deemed appropriate. Selection of stainless steel material ensured that bounding multipliers were obtained, since environmental effects in stainless steel material are more pronounced than in carbon/low alloy steel material.

Turkey Point Specific Evaluation

Section 5.5 of NUREG/CR-6260 describes the results for the older vintage Westinghouse plant, which corresponds to a plant very similar to Turkey Point. Those results provide an assessment of selected components (both ASME Section III, Class 1 and ANSI B31.1 components) with respect to environmental fatigue. The components evaluated included the reactor pressure vessel (RPV) shell at the core support pads, the inside surface of the RPV inlet and outlet nozzles, the pressurizer surge line hot leg nozzle safe end, the charging nozzle, the safety injection nozzle, and the residual heat removal (RHR) piping tee. The NUREG/CR-5999 CUF results shown in Table 5-98 of NUREG/CR-6260 (i.e., CUFs based on design stresses with environmental effects included) provide a basis from which to evaluate the Turkey Point components with respect to environmental fatigue. The CUFs from NUREG/CR-6260 are summarized in Table 1.

On the basis of the fatigue usage values in Table 1, all NUREG/CR-6260 component fatigue usage values, based on design-basis cycle definitions, are less than 1.0 except for the pressurizer surge line hot leg nozzle safe end (U = 4.248). However, because these fatigue values were based on original design CUF values that are different than the current values for Turkey Point (including the recent power uprate evaluation performed in 1995), the CUF values need to be adjusted to estimate appropriate values for Turkey Point. As described in the paragraphs below, an adjustment is made to the CUF values to accommodate this difference.

In the case of RPV shell at core support pads, the CUF results considering environmental effects are documented in Section 5.5.1 of NUREG/CR-6260. From Section 5.5.1.2 and associated Table 5-84 of NUREG/CR-6260, it is seen that 99% of the CUF is attributed to 200,000 cycles of frictional forces/vibrational loadings on this component. Discussions with Westinghouse personnel conclude that the 200,000 cycles of this load result from a combination of steady state fluctuations and transient fluctuations attributed to the design thermal events. Further, Westinghouse characterizes the steady state fluctuations as relatively rapid in nature, similar to earthquake loading. Therefore, the portion of these events caused by steady state fluctuations would not experience environmental effects because of high strain rates that exceed the threshold value above which environmental effects do not apply. Based on the actual transient counts projected to 60 years for Turkey Point for all design basis transients (as compiled by Westinghouse), as well as the assumption that the 200,000 cycles are directly related to the total number of design basis transient events, a revised number of cycles of 34,189 was determined for the limiting Turkey Point unit. Based on this reduced number of cycles, a revision to the NUREG/CR-6260 CUF calculation is provided in Table 2. The CUF calculation shown in Table 2 separates the 34,189 cycles into a transient portion where environmental effects apply (12,479 cycles), and a steady state fluctuation portion where environmental effects are neglected (21,710 cycles). The resulting CUF is calculated to be 0.134, as shown in Table 2. This value replaces the 0.891 value calculated in Table 5-84 of NUREG/CR-6260.

The Turkey Point design CUF for the RPV shell at core support pads is 0.478 compared to the value of 0.290 reported for the inside surface in NUREG/CR-6260. Therefore, an estimate of the NUREG/CR-5999 CUF for Turkey Point is (0.478/0.290) * 0.134 = 0.221, which is well below the allowable value of 1.0 and is not considered to represent a fatigue concern when environmental effects are considered. Because these results were determined for a low alloy/carbon steel material, no additional adjustment is required to incorporate more recent laboratory data, as

discussed above. Finally, it is noteworthy that this location is currently inspected per the Turkey Point ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program described in LRA Appendix B, Subsection 3.2.1.1 (page B-27).

Similarly, for the RPV outlet nozzle, the Turkey Point design CUF is 0.530 compared to the value of 0.193 reported for the inside surface in NUREG/CR-6260. Therefore, an estimate of the NUREG/CR-5999 CUF for Turkey Point is (0.530/0.193) * 0.499 = 1.37. This value can be shown to be acceptable considering environmental effects if the projected number of cycles is considered, as shown in Table 3. The calculation shown in Table 3 is a repeat of the calculation performed in Table 5-87 of NUREG/CR-6260, except that the projected transient counts for Turkey Point for 60 years are used. The revised CUF is 0.372, which becomes 1.02 for 60 years after scaling by (0.530/0.193). This CUF value, which includes environmental effects, is considered to be acceptable based on other conservatism remaining in the process. One such factor is that the most fatigue-sensitive location in the RPV outlet nozzle at Turkey Point is at the nozzle-shell junction on the outside surface, where reactor water environmental effects do not apply. Because these results were determined for a low alloy/carbon steel material, no additional adjustment is required to incorporate more recent laboratory data, as discussed above.

Similarly, for the RPV inlet nozzle, the Turkey Point design CUF is 0.447 compared to the value of 0.135 reported in NUREG/CR-6260. Therefore, an estimate of the NUREG/CR-5999 CUF for Turkey Point is (0.447/0.135) * 0.302 = 1.0, which is equal to the allowable value. Considering other conservatism remaining in the process, such as actual cycle counts, this CUF value is acceptable. Again, the most fatigue-sensitive location in the RPV inlet nozzle at Turkey Point is at the nozzle-shell junction on the outside surface, where reactor water environmental effects do not apply. Because these results were determined for a low alloy/carbon steel material, no additional adjustment is required to incorporate more recent laboratory data, as discussed above.

For three of the remaining NUREG/CR-6260 locations (safety injecting nozzle, charging nozzle, RHR piping tee), which are stainless steel locations, the CUF results are acceptable, even with a factor of two applied to incorporate more recent laboratory data, as discussed above. Note that the U_{6260} fatigue usage values reflect the 40-year values from NUREG/CR-6260, since the Turkey Point cycle projections are predicted to remain within the 40-year design basis values for sixty years of operation.

Since actual projected cycle counts were utilized in the RPV outlet nozzle and the RPV shell at the core support pads evaluations above, FPL will either: (1) modify the Turkey Point Fatigue Monitoring Program to limit transient accumulations to the values used in the above evaluations, (2) perform a more refined evaluation for the RPV outlet nozzle and the 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. Use of one of these three approaches will ensure that the CUFs for the RPV outlet nozzle and the RPV shell at the core support pads remain valid for the period of extended operation, including the consideration of reactor water environmental effects.

It is noteworthy that the CUF results presented in this response include maximum environmental effects in that conservative strain rates were utilized in the NUREG/CR-6260 calculations. In addition, environmental effects were uniformly applied without consideration of threshold criteria that might indicate an absence of environmental conditions. Finally, an additional penalty factor was applied to account for the most recent laboratory data. Therefore, the environmental adjustments to the CUF results are considered to be very conservative.

Based on the results of NUREG/CR-6260, as well as the additional results presented in this response, all candidate locations for environmental fatigue effects, except for the surge line hot leg nozzle, have been addressed. The surge line hot leg nozzle is addressed in the response to RAI 4.3.5-2 below.

TABLE 1
FATIGUE USAGE VALUES FOR EVALUATED LOCATIONS USING DESIGN BASIS CYCLES FROM NUREG/CR-6260

Location	U_{code}	U_{PTN} (4)	U_{6260}
RPV Shell at Core Support Pads	0.290	0.478	0.891
RPV Inlet Nozzle (inside surface)	0.135 (1)	0.447 (5)	0.302
RPV Outlet Nozzle (inside surface)	0.193 (1)	0.530 (5)	0.499
Surge Line Hot Leg Nozzle	0.900 (2)	0.944	4.248
Safety Injection Nozzle	0.046	(3)	0.327
Charging Nozzle	0.030	(3)	0.319
RHR Piping Tee	0.022	(3)	0.205

Notes:

1. Inside surface fatigue usage values were utilized because the higher CUFs reported for the outside surface locations are not exposed to the reactor water environment.

2. NUREG/CR-6260 found that the analyses were very refined, with little opportunity for usage factor reduction.

3. This location was evaluated to ANSI B31.1 rules and does not produce a fatigue usage value.

4. U_{PTN} represents fatigue usage computed specifically for the transients developed for Turkey Point including the recent power uprate evaluation performed in 1995.

5. These are CUF values for outside surface locations.

TABLE 2
REVISED FATIGUE USAGE CALCULATION FOR REACTOR PRESSURE VESSEL LOWER HEAD AND SHELL FOR 60 YEARS INCLUDING ENVIRONMENTAL EFFECTS

Transient Pair	S_{alt}	S_{alt} (adjusted)	N	n	u
OBE A/OBE B	19.86	22.07	44,636	400	0.009
Frictional forces/vibration (with environmental - represents transient cycles)	13.48	14.98	226,736	12,479	0.055
Frictional forces/vibration (without environmental - represents steady state fluctuation cycles)	13.48	14.98	310,942 Note 1	21,710	0.070
				Total	0.134

Notes:
1. This value was obtained from the ASME Code fatigue curve for carbon/low alloy steel.

TABLE 3
REVISED FATIGUE USAGE CALCULATION FOR RPV OUTLET NOZZLE FOR 60 YEARS INCLUDING ENVIRONMENTAL EFFECTS

Transient Pair	S_{alt}	S_{alt} (adjusted)	N	n	u
Heatup/cooldown	15.5	17.22	139,150	381 (1)	0.0003
Plant loading/unloading	17.0	18.89	92,575	2,320 (2)	0.025
OBE A/OBE B	18.85	20.94	57,109	400	0.007
Combination	29.5	32.78	8,179	2,760	0.337
				Total	0.372

Notes:
1. The actual total number of heatup and cooldown cycles projected to 60 years for Turkey Point was used.

2. Consistent with Table 5-87 of NUREG/CR-6260, the maximum number of plant loading and unloading cycles projected to 60 years for Turkey Point (2,720), less the number of OBE A/OBE B cycles (400) was used.

RAI 4.3.5-2:
In Section 4.3.5 of the application you indicate that the pressurizer surge line required further evaluation for environmental fatigue during the period of extended operation. You further indicated that you would use an aging management program to address fatigue of the surge line during the period of extended operation. Your 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. You have not provided a technical basis demonstrating the technical adequacy of your proposal. Provide a detailed technical evaluation which demonstrates that 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, 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 one.

FPL RESPONSE:
The proposed aging management program to address fatigue of the Turkey Point Units 3 and 4 pressurizer surge lines during the period of extended operation is similar to the approach documented in the ASME Boiler and Pressure Vessel Code, Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix L. However, FPL recognizes that to date, the NRC has not endorsed the Appendix L approach. The primary NRC concerns with Appendix L include crack aspect ratio and acceptable fatigue crack growth rates (including environmental effects).

As noted in LRA Subsection 4.3.5 (page 4.3-10), three pressurizer surge line welds on Unit 3 have each been ultasonically examined twice, and on Unit 4, one weld has been ultrasonically examined three times, two welds have each been ultasonically examined twice, and an additional weld has been ultasonically examined once. No reportable indications have been found. This subsection of the LRA goes on to state that FPL plans to inspect

all surge line welds on both units during the fourth inservice inspection interval, and prior to entering the extended period of operation. The results of these inspections will be utilized to assess the appropriate approach for addressing environmentally-assisted fatigue of the surge lines. The approach developed could include one or more of the following:

1. Further refinement of the fatigue analysis to lower the CUF(s) to below 1.0, or
2. Repair of the affected locations, or
3. Replacement of the affected locations, or
4. Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Should FPL select Option 4 (i.e., inspection) to manage environmentally-assisted fatigue during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC prior to entering the period of extended operation.

SECTION 4.5 CONTAINMENT TENDON LOSS OF PRESTRESS

RAI 4.5-1:
You have performed the time-limited aging analysis of the prestressing forces in accordance with 10 CFR 54.21(c)(1)(ii) to demonstrate that after considering the projected loss of tendon prestress forces, the residual prestressing forces in each direction (i.e., hoop, meridional, and dome) will remain above the minimum required prestressing forces for the extended period of operation. Provide the following information:

1. Curves showing the comparison of the projected measured prestressing forces (i.e. trend lines) versus the minimum required prestressing forces in each major direction, with a short description of the method used to project the measured forces (for both Turkey Point units, if they are different).

2. How do the trend lines represent the large number of exempt tendons (i.e. not subjected to lift-off testing because of personnel safety considerations)?

FPL RESPONSE:
1. FPL has compared the projected measured prestressing forces (i.e., trend lines) versus the predicted lower limits and minimum required prestressing forces for each tendon type (hoop, dome, and vertical) for each unit. Measured forces have been projected to year 60 in the form of trend lines using linear regression analysis of the tendon lift-off test data for all tendon surveillances performed through the year 2000. The results are summarized in the following table:

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

2. Per the safety evaluation transmitted via NRC letter from R.W. Hernan to T.F. Plunkett, dated 10/20/99, the NRC authorized FPL's Relief Request 20 to exempt hoop and dome tendons located above the main steam platforms for personnel safety reasons. The exempted tendons represent a small percentage of the total population of tendons available for testing. 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 is representative of the small number of exempted tendons.

SECTION 4.6 CONTAINMENT LINER PLATE FATIGUE

RAI 4.6-1:
With respect to Item 2 in Section 4.6, provide the basis for determining that the original projected number of maximum reactor coolant system design cycles is conservative enough to envelop the projected cycles for the extended period of operation. Also, provide the basis for the projected number of cycles for the extended period of operation for the containment liner plate and the containment liner penetrations.

FPL RESPONSE:
Item 2 in LRA Section 4.6 (page 4.6-1) deals with thermal cycling due to containment interior temperature varying during heatup and cooldown of the Reactor Coolant System (RCS). The containment liner plate was designed for 500 cycles of RCS heatup and cooldown. This is well above the maximum allowable heatup/ cooldown cycles of 200 for the RCS. As demonstrated in the response to RAI 4.3.1-1 above, the projected cycles of heatup/cooldown for the extended period of operation, are well within the original design limits. Therefore, the 500 RCS heatup and cooldown thermal cycles assumed for the containment liner plate bound the expected number of cycles for the period of extended operation.

RAI 4.6-2:
In Item 4 in Section 4.6 you state that the design of the containment penetrations meet the general requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel, Section III.

1. Did the fatigue analyses of the main steam piping, feedwater piping, blowdown piping and letdown piping, containment penetration assemblies and welds include stresses due to restrained piping system thermal expansion loads, in addition to the stresses due to local thermal expansion? If they did not, explain why you consider the analyses to be adequate.

2. Were the stresses due to restrained piping system thermal expansion loads, and the stresses due to local thermal expansion of the penetration assemblies also included in the fatigue analysis of the containment liner plate? If they were not included, explain why you consider the analyses to be adequate.

FPL RESPONSE:
As indicated in Item 4, Section 4.6 (page 4.6-2) of the LRA, the design of the containment penetrations meets the general requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III. 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 of the ASME Boiler and Pressure Vessel Code, Section III, Article 4 and the allowable strain limits prescribed in Paragraph N-414. As stated in the UFSAR, the evaluation of the penetrations considers stresses from the effects of pipe loads, pressure loads, thermal loads, deadloads 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 (see UFSAR Figure 5.1-16) are designed in accordance with the applicable fatigue requirements of the Code and that all penetrations have been reviewed for a conservative number of cycles expected during unit life. As discussed in Section 4.6 (page 4.6-2) of the LRA and Appendix 5B of the UFSAR, the Containment was evaluated for 500 cycles of heatup and cooldown including the above-mentioned conservative number of penetration cycles. This is well above the maximum allowable heatup/cooldown cycles of 200 for the Reactor Coolant System. As demonstrated in the response to RAI 4.3.1-1 above, the projected

cycles of heatup/cooldown for the extended period of operation, are well within the original design limits. Hence, the analyses associated with the containment liner and penetrations have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

RAI 4.6-3:
How did you include the effects of leak rate pressure testing in the fatigue analysis of the containment liner plate and the containment liner penetrations? Provide justifications, if these effects were not included.

1. Provide the minimum number of allowable cycles determined under the thermal cycling design loading conditions for the containment liner plate and the containment liner penetrations.

2. Are the containment liner plate and the containment liner penetrations included within the scope of the Turkey Point Fatigue Monitoring Program, referred to in Section 4.3.1 of the application? If not, provide justifications for not including these components in the program.

FPL RESPONSE:
As required by Section III of the ASME Boiler and Pressure Vessel Code, the effects of leak rate pressure testing (significant pressure fluctuations) are included in the containment liner analyses.

1. As stated in LRA Section 4.6 (page 4.6-1), the containment liner is designed for 500 thermal cycles. As stated in LRA Section 4.3.4 (page 4.3-6), the piping is limited to 7000 equivalent full temperature cycles. Both loading conditions are included in the design of the containment liner plate and the containment liner penetrations.

2. As described in Section 3.2.7 of Appendix B to the LRA, (page B-54), the Fatigue Monitoring Program is a confirmatory program for fatigue of class 1 components in the Reactor Coolant System (RCS). Consequently, the containment liner plate and liner penetrations are not included in the Fatigue Monitoring Program referred to in LRA Appendix B Subsection 4.3.1 (page 4.3-2).

SUBSECTION 4.7.4 CRANE LOAD CYCLE LIMIT

RAI 4.7.4-1:
In Section 4.7.4 of the LRA, the applicant identified the crane 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 grantry 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. The applicant further 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. In order to determine the adequacy of the applicant analyses, the applicant is requested to provide the load cycles experienced thus far, and cycles estimated to occur up to the end of the extended period of operation including the conditions and assumptions used in it's analyses for the applicable cranes. The applicant is also requested to provide the basis of the 200,000 load cycle limit for the spent fuel pool bridge cranes.

FPL RESPONSE:
As described in LRA Subsection 4.7.4 (page 4.7-5), actual crane usage is far less than qualified usage over the extended life of the plant. Consequently, FPL 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 8,000 cycles in 60 years.

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.

As described in LRA Subsection 4.7.4 (page 4.7-5), 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 CMAA Specification No. 70, Specifications for Electric Overhead Traveling Cranes.

Based on the above, the Turkey Point cranes will continue to perform their intended function throughout the period of extended operation.