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

FPL

L-2001-236
10 CFR 54

NOV 1 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
License Renewal Safety Evaluation Report Open Item And Confirmatory Item Responses And Revised License Renewal Application Appendix A

By letter dated August 17, 2001, the NRC issued the Safety Evaluation Report with Open Items Related to the License Renewal of Turkey Point Nuclear Plant, Units 3 and 4. Attachment 1 to this letter provides responses to the open items and confirmatory items identified in the Safety Evaluation Report. In order to address commitments related to open items, confirmatory items, and other items from previous RAI responses, FPL has prepared a revised Appendix A to the Turkey Point Units 3 and 4 License Renewal Application (LRA) entitled, "Updated Final Safety Analysis Report Supplement." This revised LRA Appendix A also incorporates changes as a result of the LRA annual update (FPL Letter L-2001-234 dated October 22, 2001). Attachment 2 describes the changes to LRA Appendix A. Attachment 3 is the revised LRA Appendix A in its entirety.

Additionally, in a telephone conversation on September 27, 2001, the NRC requested a clarification related to visual inspection of the reactor vessel internals. This clarification is provided as Attachment 4.

Finally, Attachment 5 is provided to address the additional open item raised by the NRC regarding aging management of concrete.

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

Very truly yours,

John P. McElwain
Vice President - Turkey Point

JPM/EAT/hlo
Attachments (5)

A084

Rec'd at NRC/DCD
12/12/01

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)

John P. McElwain 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.

John P. McElwain

John P. McElwain

Subscribed and sworn to before me this

1st day of November, 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

John P. McElwain is personally known to me.

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

Chief, License Renewal and Standardization Branch
Project Manager - Turkey Point License Renewal
Project Manager - Turkey Point

U.S. Nuclear Regulatory Commission, Region II
Regional Administrator, Region II, USNRC
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ATTACHMENT 1
RESPONSE TO OPEN ITEMS IDENTIFIED IN
SAFETY EVALUATION REPORT RELATED TO LICENSE RENEWAL OF
TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4

OPEN ITEM 2.1.2-1:

The staff has reviewed and disagrees with the applicant's scoping criteria for seismic II over I piping systems. The staff's position is that the seismic II over I piping systems whose failure could prevent safety related systems and structures from accomplishing their intended functions should be within the scope of license renewal in accordance with the scoping requirements 10 CFR 54.4(a)(2). For these Seismic II/I Piping systems, the applicant should perform an AMR to determine if there are any plausible aging effects, and identify appropriate aging management programs. The applicant needs to clarify the scope of its seismic II over I piping systems (i.e., whether it includes non-safety-related piping systems that are connected to safety related piping systems as well as non-safety-related piping systems that are not connected to safety-related piping systems). The applicant also needs to address the criteria used to postulate breaks and cracks in non-safety-related piping systems that are within the seismic II over I scope, if it wishes to take credit for protection of safety-related systems. The applicant must demonstrate that plant mitigative features, which are provided to protect safety-related SSCs from a failure of non-safety-related piping systems, are within the scope of license renewal.

Note that further discussion is provided in Sections 2.1.2.1 and 3.4.16.4 of the SER.

FPL RESPONSE:

As noted in the Turkey Point License Renewal Application (LRA) and subsequent RAI responses (FPL letters L-2001-49 dated 3/22/01 and L-2001-113 dated 5/3/01), the following components and structural components have been included in the scope of license renewal to protect safety-related SSCs from a failure of non-safety related piping systems and other SSCs (scoping criteria 10 CFR 54.4(a)(2)):

1. Non-safety related piping segments and supports at safety-related/non-safety related functional boundaries which extend beyond the system pressure boundary valve to ensure the integrity of the safety-related/non-safety related functional system pressure boundary (LRA Tables 3.6-2 through 3.6-20).

2. Piping/component supports for non-safety related mechanical systems with the potential of "Seismic II over I" interaction with safety related components (LRA Tables 3.6-1 through 3.6-20).

3. Non-safety related conduit, cable trays, supports, and other structural components with the potential of "Seismic II over I" interaction with safety related components (LRA Tables 3.6-1 through 3.6-20).

4. Design features required to accommodate the effects of flooding such as curbing, platforms, sumps, and sump pumps (LRA Tables 3.6-1 through 3.6-20, and Table 3.4-7).

5. Design features required to accommodate the effects of spray, jet impingement, and pipe whip such as pipe whip restraints and internal barriers (LRA Tables 3.6-1 through 3.6-20).

As stated in LRA Section 2.1.1.3 (page 2.1-7), Turkey Point was not licensed for "seismic II over I", and is not committed to compliance with Paragraph C.2 of Regulatory Guide 1.29, except for the Reactor Coolant Pump (RCP) Oil Collection System (UFSAR Appendix 9.6A, Table 2.5, page 9.6A-83, and Subsection 3.10.3, page 9.6A-103). However, "seismic II over I" was conservatively considered for license renewal scoping based on FPL's understanding of NRC Staff guidance. As a result, FPL included the components and structural components noted above in the scope of license renewal for Turkey Point. Because the seismic interaction design feature is dependent upon the location of non-safety related systems or structures relative to the safety related systems and structures, an area based approach for scoping of "seismic II over I" was chosen. This approach identified the major structures of the plant containing both safety related and non-safety related components and structural components which are as follows:

- Containments
- Auxiliary Building
- Control Building
- Electrical Penetration Rooms
- Emergency Diesel Generator Buildings
- Intake Structure
- Main Steam and Feedwater Platforms
- Turbine Building
- Yard Structures

The component and structural component level scoping performed as part of the screening process then established the specific non-safety related seismic interaction component or structural component types located within the structure for inclusion in the license renewal scope. Note that the review for seismic, leakage, pipe rupture and other interactions of non-safety related components and structural components (SCs) that could potentially affect safety related SCs included non-safety related piping systems that are connected to safety related piping systems as well as non-safety related piping systems that are not connected to safety related piping systems. This review considered Turkey Point's current licensing basis (CLB) for seismic, leakage, pipe rupture and other interactions. Those items determined to have an interaction were included in the scope of license renewal, and an aging management review (AMR) was performed and reviewed by the NRC staff as part of the LRA.

As stated, the above approach was based on existing Turkey Point CLB pipe break assumptions regarding leakage, spray, jet impingement, etc. The NRC concern identified in the open item is that aging of non-safety related piping could change pipe break assumptions, and as a result, create additional interactions of non-safety related piping with safety related SCs that were not considered in FPL's original license renewal scoping. If these additional interactions could affect safety related functions, additional non-safety related piping may have to be included within the scope of license renewal. To address this concern, FPL has performed the following evaluation to establish what additional non-safety related piping should be included in the scope of license renewal.

1. For each of the major structures of the plant containing both safety related and non-safety related components and structural components, non-safety related piping systems containing fluid and/or steam were identified. This includes high energy and other piping.

2. If the identified non-safety related piping was in the scope of license renewal to address the other scoping criteria of 10 CFR 54.4(a), no additional evaluation of this piping was required since an AMR has already been performed and appropriate aging management programs (AMPs) identified to ensure intended functions are maintained. These AMRs and AMPs are included in the LRA and have already been reviewed by the NRC staff.

3. All remaining non-safety related piping from the completion of Steps 1 and 2 above was then assumed to fail anywhere along its length.

4. Based on the assumed failures from Step 3, and a review of design drawings and plant walk downs, the effects of pipe whip, jet impingement, physical contact (piping falling such that it physically contacts safety related equipment), spray, and/or leakage were evaluated to determine if these interactions could potentially impact safety related functions. Specifically, the effects of pipe whip, jet impingement, and physical contact were considered for all non-safety related high energy piping, and the effects of spray and leakage were considered for all other non-safety related piping. If the effects of these interactions were determined to impact safety related functions, the non-safety related piping and its associated components were identified as within the scope of license renewal. If there was no impact on safety related functions as a result of the effects of these assumed failures, the piping was determined not to meet the scoping criteria of 10 CFR 54.4(a)(2), and thus not within the scope of license renewal.

5. If the piping and associated components were determined to be within the scope of license renewal, an AMR evaluation was performed on these components based on AMRs performed on components of the same material exposed to the same internal and external environments.

The results of the above evaluation are presented below for each major structure of the plant containing both safety related and non-safety related components and structural components.

Containments

Pipe Whip/Jet Impingement/Physical Contact - There is no non-safety related high energy piping inside the Containments. All high energy piping is safety related and thus within the scope of license renewal.

Spray/Leakage - Safety related components inside the Containments are designed to accommodate the effects of leakage and spray, without loss of function, regardless of the source.

Results - No additional non-safety related piping is required to be included within the scope of license renewal.

Auxiliary Building

Pipe Whip/Jet Impingement/Physical Contact - There is no non-safety related high energy piping inside the Auxiliary Building. All high energy piping is safety related and thus within the scope of license renewal.

Spray/Leakage - The Auxiliary Building contains non-safety related piping and associated components that could potentially affect safety related electrical equipment if arbitrary failures are assumed. The specific piping is as follows:

- Small bore, carbon steel, service (potable) water piping and associated components in the main hallways and the electrical equipment room
- Small bore, stainless steel, chilled water piping and associated components in the electrical equipment room
- Small bore, stainless steel, primary water piping and associated components in various areas
- Small bore, stainless steel, chemical and volume control piping and associated components in various areas
- Small bore, stainless steel, primary sampling piping and associated components in various areas
- Small bore, stainless steel, waste disposal piping and associated components in various areas

Results - The piping and associated components noted above have been included in the scope of license renewal as meeting the scoping criteria of 10 CFR 54.4(a)(2). An AMR evaluation of these components based on AMRs of components of the same materials exposed to the same internal and external environments yields the results presented below.

TABLE 1
COMPONENTS MEETING 10 CFR 54.4(a)(2)
IN THE AUXILIARY BUILDING

Component/ Commodity Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effects Requiring Management**	**Program/ Activity**
Internal Environment					
Piping/ fittings Valves (Service Water-Main Hallways and Electrical Equipment Room)	Pressure Boundary	Carbon Steel	Raw water - city water	Loss of material	Systems and Structures Monitoring Program
Piping/ fittings Valves (Chilled Water-Electrical Equipment Room)	Pressure Boundary	Stainless Steel	Treated water	Loss of material	Chemistry Control Program
Piping/ fittings Valves (Primary Water-various areas)	Pressure Boundary	Stainless Steel	Treated water	Loss of material	Chemistry Control Program
Piping/ fittings Valves (Chemical Volume and Control-various areas)	Pressure Boundary	Stainless Steel	Treated water - borated	Loss of material	Chemistry Control Program
Piping/ fittings Valves (Sample System-various areas)	Pressure Boundary	Stainless Steel	Treated water - borated	Loss of material	Chemistry Control Program
Piping/ fittings Valves (Waste Disposal-various areas)	Pressure Boundary	Stainless Steel	Treated water - borated	Loss of material	Chemistry Control Program

TABLE 1 (continued)
COMPONENTS MEETING 10 CFR 54.4(a)(2)
IN THE AUXILIARY BUILDING

Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
External Environment					
Piping/ fittings Valves (Service Water-Main Hallways and Electrical Equipment Room)	Pressure Boundary	Carbon Steel	Indoor - not air conditioned	Loss of material	Systems and Structures Monitoring Program
Piping/ fittings Valves (Chilled Water-Electrical Equipment Room)	Pressure Boundary	Stainless Steel	Indoor - not air conditioned	None	None Required
Piping/ fittings Valves (Primary Water-various areas)	Pressure Boundary	Stainless Steel	Indoor - not air conditioned	None	None Required
Piping/ fittings Valves (Chemical Volume and Control-various areas)	Pressure Boundary	Stainless Steel	Indoor - not air conditioned	None	None Required
Piping/ fittings Valves (Sample System-various areas)	Pressure Boundary	Stainless Steel	Indoor - not air conditioned	None	None Required
Piping/ fittings Valves (Waste Disposal-various areas)	Pressure Boundary	Stainless Steel	Indoor - not air conditioned	None	None Required

Control Building

Pipe Whip/Jet Impingement/Physical Contact - There is no high energy piping in the Control Building.

Spray/Leakage - The Control Building contains small bore, galvanized carbon steel and copper, non-safety related service (potable) water piping and associated components that could potentially affect safety related electrical equipment if failures are assumed.

Results - The service water piping and associated components noted above have been included in the scope of license renewal as meeting the scoping criteria of 10 CFR 54.4(a)(2). An AMR evaluation of these components based on AMRs performed on copper and galvanized carbon steel components exposed to the same internal and external environments yields the results presented below.

TABLE 2
COMPONENTS MEETING 10 CFR 54.4(A)(2)
IN THE CONTROL BUILDING

Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
Internal Environment					
Service Water Piping/ fittings Valves	Pressure Boundary	Carbon steel - galvanized Copper	Raw water - city water	Loss of material	Systems and Structures Monitoring Program
External Environment					
Service Water Piping/ fittings Valves	Pressure Boundary	Carbon steel - galvanized Copper	Indoor - air conditioned	None	None required

Electrical Penetration Rooms

There is no piping in the Electrical Penetration Rooms.

Results - No additional non-safety related piping is required to be included within the scope of license renewal.

Emergency Diesel Generator Buildings

Pipe Whip/Jet Impingement/Physical Contact - There is no high energy piping in the Emergency Diesel Generator Buildings.

Spray/Leakage - The Unit 3 Emergency Diesel Generator Building contains small bore, carbon steel, non-safety related service (potable) water piping and associated components that could potentially affect safety related electrical equipment if arbitrary failures are assumed. The Unit 4 Emergency Diesel Generator Building contains small bore, stainless steel, non-safety related demineralized water piping and associated components that could potentially affect safety related electrical equipment if failures are assumed.

Results - The service and demineralized piping and associated components above have been included in the scope of license renewal as meeting the scoping criteria of 10 CFR 54.4(a)(2). An AMR evaluation of these components based on AMRs performed on carbon and stainless steel components exposed to the same internal and external environments yields the results presented below.

TABLE 3
COMPONENTS MEETING 10 CFR 54.4(A)(2)
IN THE EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
Internal Environment					
Piping/ fittings Valves (Service Water-Unit 3 Emergency Diesel Building)	Pressure Boundary	Carbon Steel	Raw water - city water	Loss of material	Systems and Structures Monitoring Program
Piping/ fittings Valves (Demin Water-Unit 4 Emergency Diesel Building)	Pressure Boundary	Stainless Steel	Treated water - other	Loss of material	Chemistry Control Program

TABLE 3 (continued)
COMPONENTS MEETING 10 CFR 54.4(A)(2)
IN THE EMERGENCY DIESEL GENERATOR BUILDINGS

Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
External Environment					
Piping/ fittings Valves (Service Water-Unit 3 Emergency Diesel Building)	Pressure Boundary	Carbon Steel	Indoor - not air conditioned	Loss of material	Systems and Structures Monitoring Program
Piping/ fittings Valves (Demin Water-Unit 4 Emergency Diesel Building)	Pressure Boundary	Stainless Steel	Indoor - not air conditioned	None	None Required

Intake Structure

Pipe Whip/Jet Impingement/Physical Contact - There is no high energy piping at the Intake Structure.

Spray/Leakage - This is an outdoor area. All safety related equipment is designed for outdoor service and as a result would not be impacted from leakage or spray.

Results - No additional non-safety related piping is required to be included within the scope of license renewal.

Main Steam and Feedwater Platforms

Pipe Whip/Jet Impingement/Physical Contact - All high energy piping within the Main Steam and Feedwater Platforms is located outdoors. Additionally, the non-safety related Main Steam and Feedwater piping in these areas is within the scope of license renewal because it meets other scoping criteria of 10 CFR 54.4(a) (See License Renewal Boundary Drawings 3-FW-03, 3-MS-01, 4-FW-03, and 4-MS-01).

Spray/Leakage - These are outdoor areas. All safety related equipment is designed for outdoor service and as a result would not be impacted from leakage or spray.

Results - No additional non-safety related piping is required to be included within the scope of license renewal.

Turbine Building

Pipe Whip/Jet Impingement/Physical Contact - All high energy piping within the Turbine Building is located outdoors. Additionally, significant portions of the non-safety related Main Steam and Feedwater piping and associated components are within the scope of license renewal because they meet other scoping criteria of 10 CFR 54.4(a) (See License Renewal Boundary Drawings 3-FW-01, -02, -03, 3-MS-01, -02, -03, 3-TG-01, 4-FW-01, -02, -03, 4-MS-01, -02, -03, and 4-TG-01). Other non-safety related high energy piping in the Turbine Building includes portions of the Auxiliary Steam, Condensate, Extraction Steam, and Feedwater Heater Drain and Vent Systems. Piping segments of the Auxiliary Steam, Condensate, Feedwater (beyond that noted above in the vicinity of the Unit 3 feedwater pump rooms), and Feedwater Heater Drains and Vents Systems could potentially affect safety related cable trays and conduit in certain areas of the Turbine Building if failures are assumed.

Spray/Leakage - This is essentially an outdoor area. All safety related equipment that is in proximity to non-safety related piping is designed for outdoor service and as a result would not be impacted from leakage or spray.

Results - The segments of the Auxiliary Steam, Condensate, Feedwater, and Feedwater Heater Drains and Vents system piping and associated components noted above have been included in the scope of license renewal as meeting the scoping criteria of 10 CFR 54.4(a)(2). An AMR evaluation of these components based on AMRs performed on carbon steel components exposed to the same internal and external environments yields the results presented below.

TABLE 4
COMPONENTS MEETING 10 CFR 54.4(a)(2)
IN THE TURBINE BUILDING

Component/ Commodity Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effects Requiring Management**	**Program/ Activity**
Internal Environment					
Piping/fittings Valves (Auxiliary Steam - various areas)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
Piping/fittings Valves (Condensate - outlet of #2 feedwater heaters to main feedwater pump suction)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
Piping/fittings Valves Unit 3 only (Feedwater - feedwater pump recirculation lines) (Feedwater Heater Drains and Vents - heater drain pump discharge to feedwater pump suction)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
Piping/fittings Valves (Feedwater Heater Drains and Vents - portions of the 3A, 3B, and 4B reheater drain tank drains)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
Piping/fittings Valves (Feedwater Heater Drains and Vents - #6 to #5 feedwater heater drains)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program

TABLE 4 (continued)
COMPONENTS MEETING 10 CFR 54.4(a)(2)
IN THE TURBINE BUILDING

Component/ Commodity Grouping	Intended Function	Material	Environment	Aging Effects Requiring Management	Program/ Activity
External Environment					
Piping/fittings Valves (Auxiliary Steam - various areas)	Pressure Boundary	Carbon Steel	Outdoor	None	None required
Piping/fittings Valves (Condensate - outlet of #2 feedwater heaters to main feedwater pump suction)	Pressure Boundary	Carbon Steel	Outdoor	None	None required
Piping/fittings Valves Unit 3 only (Feedwater - feedwater pump recirculation lines) (Feedwater Heater Drains and Vents - heater drain pump discharge to feedwater pump suction)	Pressure Boundary	Carbon Steel	Outdoor	None	None required
Piping/ fittings Valves (Feedwater Heater Drains and Vents - portions of the 3A, 3B, and 4B reheater drain tank drains)	Pressure Boundary	Carbon Steel	Outdoor	None	None required
Piping/fittings Valves (Feedwater Heater Drains and Vents - #6 to #5 feedwater heater drains)	Pressure Boundary	Carbon Steel	Outdoor	None	None required

Yard Structures

Pipe Whip/Jet Impingement/Physical Contact - All high energy piping within Yard Structures is located outdoors between the Containments, Main Steam and Feedwater Platforms, and the Turbine Building. Additionally, non-safety related Main Steam, Feedwater, and Blowdown piping and associated components in this area are within the scope of license renewal because they meet other scoping criteria of 10 CFR 54.4(a). Other non-safety related high energy piping in Yard Structures includes portions of the Auxiliary Steam, Condensate, and Feedwater Heater Drains and Vents Systems. Piping segments of the Auxiliary Steam, Condensate, and Feedwater Heater Drains and Vents Systems could potentially affect safety related cable trays and conduit in certain areas of Yard Structures if failures are assumed.

Spray/Leakage - This is an outdoor area. All safety related equipment is designed for outdoor service and as a result would not be impacted from leakage or spray.

Results - The segments of the Auxiliary Steam, Condensate, and Feedwater Heater Drains and Vents system piping and associated components noted above have been included in the scope of license renewal as meeting the scoping criteria of 10 CFR 54.4(a)(2). An AMR evaluation of these components based on AMRs performed on carbon steel components exposed to the same internal and external environments yields the results presented below.

TABLE 5
COMPONENTS MEETING 10 CFR 54.4(a)(2)
IN YARD STRUCTURES

Component/ Commodity Grouping	**Intended Function**	**Material**	**Environment**	**Aging Effects Requiring Management**	**Program/ Activity**
Internal Environment					
Piping/fittings Valves (Auxiliary Steam - various areas)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
Piping/fittings Valves (Condensate - downstream of #4 feedwater heaters to main feedwater pump suction line)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
Piping/fittings Valves (Feedwater Heater Drains and Vents - #6 to #5 feedwater heater drains)	Pressure Boundary	Carbon Steel	Treated water - Secondary	Loss of material	Chemistry Control Program Flow Accelerated Corrosion Program
External Environment					
Piping/fittings Valves (Auxiliary Steam - various areas)	Pressure Boundary	Carbon Steel	Outdoor	None	None required
Piping/fittings Valves (Condensate - downstream of #4 feedwater heaters to main feedwater pump suction line)	Pressure Boundary	Carbon Steel	Outdoor	None	None required
Piping/fittings Valves (Feedwater Heater Drains and Vents - #6 to #5 feedwater heater drains)	Pressure Boundary	Carbon Steel	Outdoor	None	None required

Conclusion

Based on the above, the scopes of the Chemistry Control Program, Flow Accelerated Corrosion Program, and the Systems and Structures Monitoring Program have been revised to include the components as noted above.

The evaluation presented above addresses the specific issues raised by the NRC staff regarding scoping for seismic II over I piping systems. As a result, FPL requests that Open Item 2.1.2-1 be closed, and that the appropriate sections of the SER be revised accordingly.

OPEN ITEM 3.9.12-1:

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

FPL RESPONSE:

FPL will continue to be a participant in the industry programs for managing primary water stress corrosion cracking (PWSCC) in Alloy 600 reactor vessel head penetrations (VHPs) of U.S. pressurized water reactors during the period of extended operation. As documented in FPL's response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" (FPL Letter #L-2001-198 dated 9/04/01), the work performed under the Electric Power Research Institute (EPRI) Material Reliability Program (MRP) and the Nuclear Energy Institute (NEI) is an integral part of the Turkey Point Reactor Vessel Head Alloy 600 Penetration Inspection Program. This bulletin response provides the Turkey Point Unit 3 and 4 rankings utilizing the latest industry PWSCC susceptibility model, in addition to updating reactor VHP inspection commitments. As the industry gains experience, ranking models will continue to be refined and thus, Turkey Point's RVHPIP will be updated to reflect the new information and relative rankings for Turkey Point Units 3 and 4 in the Topical Reports MRP-44 and 48, accordingly.

Conclusion

The scope of the Turkey Point Reactor Vessel Head Alloy 600 Penetration Inspection Program has been revised to document continued participation in industry programs for managing PWSCC of reactor VHPs during the period of extended operation as described above. Implementation of this program provides reasonable assurance that Reactor Coolant System components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. As a result, FPL requests that Open Item 3.9.12-1 be closed.

OPEN ITEM 4.3-1:

In Section 4.3 of the LRA, the applicant indicates that a generic evaluation of underclad cracks had been extended to 60 years using fracture mechanics evaluations based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service.

The applicant further stated that the number of design cycles and transients assumed in the WCAP-15338 analysis bounds the Turkey Point Units 3 and 4 design transients identified in UFSAR Table 4.1-8 and provided in Appendix A of the LRA. Therefore, the conclusions in the WCAP are applicable to Turkey Point reactor vessels. The Westinghouse Owners Group (WOG) has submitted for staff review topical report WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants (MUHP-6110." This report describes the fracture mechanics analysis that evaluates the impact of 60 years of operation on reactor vessel underclad crack growth and reactor vessel integrity. This report is under staff review. If as a result of this review, plant specific requirements are identified, the applicant will need to meet those plant specific requirements.

FPL RESPONSE:

By letter dated October 15, 2001, the NRC issued the Safety Evaluation Report (SER) accepting WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants". The SER identified two applicant action items. Applicant action item (1) requires the applicant with a 3-loop reactor pressure vessel (RPV) to indicate whether the number of design cycles and transients assumed in the WCAP-15338 analysis bounds the number of cycles for 60 years of operation of its RPV. By FPL letter L-2001-65, RAI 4.3.2-1, FPL identified that WCAP-15338 is applicable and bounding for Turkey Point Units 3 and 4, and as such has addressed this applicant action item. Applicant action item (2) requires that those applicants for license renewal referencing the WCAP-15338 report for the RPV components ensure that the evaluation of the TLAA is summarily described in the FSAR supplement. The TLAA summary is provided in Subsection 16.3.2.2 (page A-47) of Appendix A of the Turkey Point LRA, and as such has addressed this applicant action item.

Conclusion

Based on the above, FPL requests that Open Item 4.3-1 be closed.

OPEN ITEM 3.8.4-1:

a. The staff requests that the applicant provide the specific acceptance criteria for the one-time field erected tanks internal inspection. The acceptance criteria should clearly state the threshold at which additional inspections, beyond the one-time inspection, will be implemented. The staff requests this information so that we can determine whether the acceptance criteria support the detection and evaluation of the aging effect loss of material such that the intended functions will be maintained throughout the period of extended operation.

b. As part of the RAI 3.8.4-4 (actually 3.8.4-3), the applicant was asked to describe any provisions for additional volumetric or surface examinations in the event that the scheduled one-time visual examination reveals extensive loss of material. In response, the applicant stated that the lighting and resolution requirements necessary to accomplish the internal tank inspections have not yet been established but the inspection requirements will be documented in the implementing procedure. The program requirements will need to be resolved as part of this review. This is part of open item 3.8.4-1.

c. As part of RAI 3.8.4-1, the staff requested that the applicant justify a one-time inspection program rather than periodic inspections for each of the tanks. In response, the applicant stated that the condensate storage tanks (CSTs), the refueling water storage tanks (RWSTs), and 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 $^{1}/_{16}$ inch 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. The applicant further stated that a review of plant specific operating experience revealed no other incidences of internal degradation for these tanks. Resolution of the uncertainty as to whether RWSTs and DWST are included in this statement is part of open item 3.8.4-1.

FPL RESPONSE:

ITEM a

Acceptance criteria is defined in Table A.1-1 of NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, as follows:

> "Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the structure or component intended function(s) are maintained under all CLB design conditions during the period of extended operation."

The acceptance criteria for the internal inspection of field erected tanks internal inspection will be the design corrosion allowance. Thus, any loss of material greater than the tank's corrosion allowance will require corrective action to ensure the tank's intended functions are maintained under all CLB design conditions. In addition, the subject tanks are protected by an internal coating system. The internal tank inspection will include detailed examination of the internal coatings. Any coatings deficiencies will be documented, evaluated, and repaired as necessary.

The threshold at which additional inspections, beyond the one-time inspection, will be implemented is corrosion of the tank steel. Thus, if corrosion is observed, appropriate corrective actions will be implemented and additional inspection will be scheduled based on the corrective actions implemented. If no corrosion is observed, then no additional inspections will be scheduled prior to the end of the extended period of operation.

ITEM b

Although the internal tank inspection will not be an ASME Section XI inspection, the lighting and resolution requirements will be the same as those specified for a VT-3 inspection described in IWA-2210 of ASME Section XI, for code year in effect at the time of the inspection. Using this inspection technique, if corrosion of the tank steel is identified that exceeds the acceptance criteria, then the condition will be documented, evaluated and corrective actions taken as appropriate under FPL's Corrective Action Program. This will consider the use of additional inspections using industry proven volumetric or surface examination techniques, as well as followup inspections, if needed. These additional and followup inspections would be established based on consideration of the extent of the corrosion

of the tank steel, the cause of the degradation, and the corrective actions to be implemented.

ITEM c

The review of plant operating experience revealed no incidences of internal degradation for CSTs, RWSTs, or DWST, other than the inspections, repairs, and recoating activities described above for the CSTs.

Although the RWSTs and DWST are not currently inspected internally on a periodic basis, the DWST was recently inspected as part of a pre-inspection performed by divers and the cognizant engineer prior to the installation of a floating cover inside the tank. The DWST inspection did not identify any degraded coatings or tank corrosion. Discussions with the cognizant engineer revealed that there were no signs of degradation inside the DWST. Thus, it is reasonable to anticipate that internal degradation of the DWST is not occurring, and use of the one time Field Erected Tank Internal Inspection provides reasonable assurance that the intended function of the DWST will be maintained.

Additionally, the RWSTs, CSTs, and DWST are externally inspected periodically as part of the Systems and Structures Monitoring Program. If these inspections were to identify corrosion of the tank steel, then the condition will be documented and evaluation and corrective actions taken as appropriate under FPL's Corrective Action Program. This would require the evaluation of the cause of the corrosion, including whether it was initiated internally or externally.

The Field Erected Tanks Internal Inspection is listed as a one-time inspection because no significant aging is expected. The purpose of the one-time inspection is to confirm that there are no aging effects requiring management. However, as stated in LRA Appendix B, Subsection 3.1.4 (page B-16), the results of each field erected tank internal inspection will be evaluated to determine if any additional actions are needed. If the inspections yield no degradation, then additional inspections will not be necessary prior to the end of the period of extended operation. However, if the inspection reveals internal surface degradation, then the degradation will be evaluated and repaired, as necessary, and additional inspections will be scheduled, as needed.

Conclusion

Based on the above, the scope of the Field Erected Tanks Internal Inspection has been modified to include the changes noted above. As a result, FPL requests that Open Item 3.8.4-1 be closed.

Confirmatory Item 3.0-1:

The staff reviewed the applicant's summary descriptions of the aging management programs (AMPs), and the evaluations of the time-limited aging analyses (TLAAs) provided by the applicant in Appendix A, "Safety Analysis Report Supplement," of the LRA, to ensure they are consistent with the requirements of 10 CFR 54.21(d). The staff identified several areas where the resolution of the open item or a commitment by the applicant needs to be included to meet the intent of 10CFR 54.21(d). The additional information involved the following:

FSAR Item 3.1.2-1:

The applicant has established a Quality Assurance Program to provide assurance that corrective actions, administrative controls, and confirmation process apply to all aging management programs credited for license renewal. The FPL Quality Assurance Program, described in the FPL Topical Quality Assurance Report, is in compliance with the requirements of 10 CFR 50, Appendix B.

Further Discussion in SER Section 3.1.2.3 FSAR Supplement:

The applicant has provided a summary description of the programs and activities for managing the effects of aging and the evaluation of time-limited aging analyses for the period of extended operation in UFSAR Chapter 16, which is also included in Appendix A to the LRA. The UFSAR Supplement provides a brief explanation of the new and existing programs that the applicant will use to manage the effects of aging. The explanation contains a summary of several important technical attributes, such as inspections and techniques used to identify aging effects. The quality assurance programs, which include three attributes (corrective actions, confirmation process, and administrative controls), are not described in the UFSAR Supplement. However, the applicant has provided a detailed description of the technical and quality assurance attributes in Appendix B to the LRA.

For non-safety-related structures and components that are subject to an AMR for license renewal, an applicant has an option to expand the scope of its 10 CFR Part 50, Appendix B, program to include these structures and components to address corrective actions, confirmation process, and administrative controls for aging management during the period of extended operation. In accordance with Appendix A.2, "Quality Assurance for Aging Management Programs (Branch Technical Position IQMB-1)," Section

A.2.2, Item 2 to the draft SRP, the applicant should document a commitment to expand the scope of its 10 CFR Part 50, Appendix B, quality assurance program to include non-safety-related structures and components in the UFSAR Supplement consistent with Section 2 of Appendix B to the LRA. Several aging management programs pertain to both safety-related and non-safety-related SSCs. Therefore, committing to the FPL Quality Assurance Program for all aging management programs is acceptable. The applicant may develop another approach to meet Branch Technical Position IQMB-1. This is Confirmatory Item 3.1.2-1.

FPL Response to FSAR Item 3.1.2-1:

LRA Appendix B Section 2.0 addresses Aging Management Program Attributes. Two attributes, Corrective Actions and Administrative Controls, were identified as common to all programs and are described in this section as being under the guidance of the FPL Quality Assurance Program. Confirmatory Actions are described in each individual program by stating that the followup actions will be entered into the corrective action program. The FSAR Supplement Section 16.0 is being revised to include the following:

> "FPL has established and implemented a Quality Assurance Program to provide assurance that the design, procurement, modification and operation of nuclear power plants conform to applicable regulatory requirements. The FPL Quality Assurance Program, described in the FPL Topical Quality Assurance Report, is in compliance with the requirements of 10 CFR 50, Appendix B. The FPL Quality Assurance Program meets the requirements provided by regulatory guidance and industry standards as listed in Appendix C of the FPL Topical Quality Assurance Report. Corrective Actions, Confirmatory Actions, and Administrative Controls apply to all aging management programs credited for license renewal and performed, or in the case of new programs, to be performed, in accordance with the FPL Quality Assurance Program."

FSAR Item 3.7-1:

In response to the staff's RAI 3.7.1-1, the applicant has proposed an aging management program for non-EQ cables, connections, and electrical/I&C penetrations.

Further Discussion in SER Section 3.7.3 FSAR Supplement:

In response to the staff's RAI 3.7.1-1, the applicant proposed an AMP for non-EQ cables, connections, and electrical/I&C penetrations. The acceptability of the AMP is evaluated in Section 3.7.2.1 of this SER. The applicant committed to include the AMP in the LRA. The applicant will submit the FSAR supplement update and it should include a summary description of this program to be consistent with 10 CFR 54.21(d). This is confirmatory Item 3.7-1.

FPL Response to FSAR Item 3.7-1:

The new program is described in the revised FSAR Supplement, LRA Appendix A, Chapter 16, Subsection 16.1.8, Containment Cable Inspection Program as follows:

> "The Containment Cable Inspection Program manages potential aging of non-EQ cable, connections, and penetrations. This aging management program consists of periodic visual inspection of accessible non-EQ cables, connections and penetrations within the scope of license renewal located in the containment structures that may be installed in adverse localized environments. The inspections will be implemented before the end of the initial operating license terms for Turkey Point Units 3 and 4."

FSAR ITEM 4.2-1:

Staff evaluation in Section 4.2.2 of the SER concludes that the summary description for the RCS TLAAs described in the LRA, Appendix A, are acceptable and meets the requirements of 10 CFR 54.21(d). However, as discussed, the applicant must apply the chemistry factor ratio adjustment described in RG 1.99, Rev. 2, Position 2.1, to the surveillance data when submitting the 48 EFPY P-T limits curves for review and approval. This adjustment is necessary to ensure accurate assessment of the data.

Further Discussion in SER Section 4.2.3 FSAR Supplement:

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

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

These changes should be incorporated into the FSAR supplement for the P-T limits TLAA. This is confirmatory item 4.2-1.

FPL Response to FSAR Item 4.2-1:

LRA Appendix A, Subsection 16.3.1.3 is revised to address items identified in the NRC Safety Evaluation for Turkey Point Technical Specification Amendments 208/202, issued October 30, 2000. Specifically, this change will ensure that Chemistry Factor for the reactor pressure vessel weld, as discussed in Regulatory Guide 1.99, Revision 2, Position 2.1, is considered in submittal of the 48 EFPY Pressure-Temperature curves. Also, this subsection is being revised to ensure that reactor vessel circumferential weld (heat number 72442) is tracked and considered in future submittals.

FSAR Item 4.3-1

a. In response to RAI 4.3.5-5, the applicant committed to perform additional evaluation of the surge line. The applicant committed to either (1) further refinement of the fatigue analysis to lower the CUFs to below 1.0, or (2) repair of the affected locations, or (3) replacement of the affected locations, or (4) management of the effects of fatigue by an inspection program that has been reviewed and approved by the NRC.

b. In response to RAI 4.3.5-1, the applicant performed an evaluation of the RPV outlet nozzle and the RPV shell at the core support pads using the projected number of transient cycles. The applicant committed to either (1) modify the Turkey Point FMP to limit transient accumulations to those used in the above evaluations, (2) perform a more refined evaluation of 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.

c. In response to RAI 4.3.1-4, the applicant used the actual projected number of transient cycles for the spray nozzle evaluation. The applicant committed to either (1) modify the Turkey Point FMP to limit transient accumulations to the values used in the spray nozzle evaluation, (2) perform a more refined evaluation for the spray nozzle to show acceptable CUFs for 60 years, or (3) track CUF values in addition to cycle counts to ensure CUF values remain acceptable.

Further Discussion in SER Section 4.3.3 FSAR Supplement:

The applicant's FSAR supplement for metal fatigue is provided in Appendix A, Section 16.3.2, of the LRA. The applicant described the TLAA evaluations and the transient cycle logging program. As described above, the applicant should update the FSAR supplement to provide a more detailed discussion of its proposed program to address environmental fatigue effects.

FPL Response to FSAR Item 4.3-1:

a. LRA Appendix A Subsection 16.3.2.5 is revised to include the options identified in the evaluations of the pressurizer surge lines. The last paragraph of Subsection 16.3.2.5 will be replaced with the following:

"For the pressurizer surge lines, FPL will inspect all surge line welds on both units during ASME Section XI inservice inspection plan fourth interval, and prior to entering the extended period of operation. The results of these inspections will be utilized to assess 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 NRC approved inspection program."

b. LRA Appendix A Subsection 16.3.2.5 is revised to include the options identified in the evaluations for the reactor pressure vessel outlet nozzles and the reactor pressure vessel shell at the core support pads. Subsection 16.3.2.5 will be revised to include the following:

"Since actual projected cycle counts were utilized in the reactor pressure vessel outlet nozzles and the reactor pressure vessel shell at the core support pads evaluations, FPL will either:

1. Modify the Fatigue Monitoring Program to limit transient accumulations to the values used in the evaluations, or
2. Perform a more refined evaluation for the reactor pressure vessel outlet nozzles and the reactor pressure vessel shell at the core support pads to show acceptable CUFs for 60 years, or
3. Track CUF values in addition to cycle counts to ensure CUF values remain acceptable."

c. LRA Appendix A Subsection 16.3.2.5 is revised to include the options identified in the evaluations for the pressurizer spray nozzles. Subsection 16.3.2.5 will be revised to include the following:

"Since actual projected cycle counts were utilized in the pressurizer spray nozzle evaluations, FPL will either:

1. Modify the Fatigue Monitoring Program to limit transient accumulations to the values used in the evaluations, or
2. Perform a more refined evaluation for the pressurizer spray nozzles to show acceptable CUFs for 60 years, or
3. Track CUF values in addition to cycle counts to ensure CUF values remain acceptable."

FSAR Item 3.8.4-1:

The applicant's summary description for the field erected tanks internal inspection program is provided in Section 16.1.4 of Appendix A to the LRA, and provides an overview of the one-time inspection as described in Section 3.1.4 of Appendix B of the LRA. The FSAR Supplement should be modified to reflect the applicant's response to the Open Item 3.8.4-1.

Further Discussion in SER Section 3.8.4.3 FSAR Supplement:

The staff reviewed UFSAR Section 16.1.4 of Appendix A to the LRA and concluded the applicant needs to update this section following resolution of open item 3.8.4-1.

FPL Response to FSAR Item 3.8.4-1:

Based on discussion with and acceptance by the NRC of the response to Open Item 3.8.4-1 at a public meeting on October 4, 2001, no change to the FSAR supplement is required. The description submitted with the LRA in Appendix meets the intent of 10 CFR 54.21(d).

FSAR Items 3.9.2-1 - A staff evaluation of applicant is Boroflex surveillance program is provided in Section 3.9.2 of this SER. The staff requests this applicant update its UFSAR Supplement to include a description of Boroflex and the enhancements to the related maintenance programs.

Further Discussion in SER Section 3.9.2.3 FSAR Supplement:

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

Response to FSAR Item 3.9.2-1:

The Turkey Point LRA Appendix A will be revised as follows:

Chapter 14

No changes required. The changes were incorporated in Revision 17 of the UFSAR submitted to the NRC by FPL letter L-2001-086, dated April 16, 2001.

Chapter 16 Section 16.2.2

Revise the first paragraph describing the program as follows:

"The Boraflex Surveillance Program manages the aging effect of change in material properties (including shrinkage, gap formation, and dissolution) for the Boraflex material in the spent fuel storage racks."

Revise the second paragraph as follows:

"The program includes periodic areal density testing (or other approved testing methods if available) of the encapsulated Boraflex material in the spent fuel storage racks to provide detailed information on the condition of the panels. The frequency of the Boraflex condition monitoring surveillance ensures timely detection of degradation impacting intended function and is consistent with regulatory commitments."

Confirmatory Item 4.4.2-1:

In response to the staff's concern regarding the wear cycle aging effects on motors, the applicant stated that the wear cycling is normally not the limiting factor in the qualified life of the equipment and is not discussed in the qualification package. The applicant further stated that a motor should be able to withstand 35000 to 50000 starts according to Volume 6 of the EPRI Power Plant Electrical Reference Series (page 6-46). Thus, the wear cycle aging effect is considered insignificant for these motors. The applicant committed to revise the EQ documentation packages for Westinghouse and Joy motors to include a reference to Volume 6 of the EPRI Power Plant Electrical Reference Series (page 6-46). This will be tracked as confirmatory item 4.4.2-1.

FPL Response:

FPL revised the EQ documentation packages for the Westinghouse and Joy motors to include a reference to the EPRI Power Plant Electrical Reference Series (page 6-46). The revised documentation packages were provided for NRC review during the aging management review inspection at Turkey Point in August and September 2001. No further action is required.

ATTACHMENT 2
LIST OF CHANGES TO THE
TURKEY POINT UNITS 3 AND 4
UFSAR SUPPLEMENT (LRA APPENDIX A)

Table 4.1-8

Revised table to update Design Cycles for Transients 9, 10, 11, and 12 (Reference RAI 4.3.1-1 response, FPL letter L-2001-75 dated April 19, 2001).

Page 14D-56

The page number and revision number of this page were revised in UFSAR Revision 17. This page is being changed to reflect those changes. There were no technical changes associated with this page.

Section 16.0

Added paragraph to address aging management program attributes corrective actions, confirmatory actions, and administrative controls and the FPL Quality Assurance Program (Confirmatory Item 3.0-1, FSAR Item 3.1.2-1).

Section 16.1.6

This subsection is being revised to include the following:

a. A clarification on visual inspections consistent with Attachment 4 to this letter.

b. The schedule for performing the reactor vessel internals inspections (Reference RAI 3.8.6-3 response, FPL letter L-2001-65 dated April 19, 2001).

c. A commitment to submit a report to the NRC prior to the end of the initial operating term for Unit 3. The report will summarize the understanding of aging effects applicable to the reactor vessel internals and will contain a description of the Turkey Point inspection plan (Reference RAI 3.8.6-4 response, FPL letter L-2001-65 dated April 19, 2001).

Section 16.1.7

Added a commitment to submit a report to the NRC describing the details of the Small Bore Class 1 Piping Inspection (Reference FPL Letter L-2001-136 dated June 25, 2001).

Section 16.1.8

This is a new subsection that describes the Containment Cable Inspection Program (Confirmatory Item 3.0-1, FSAR Item 3.7-1).

Section 16.2.1.1

Added a commitment to perform VT-1 examinations of the core support lugs for the period of extended operation (Reference RAI 3.2.4-1 response, FPL letter L-2001-76 dated April 19, 2001).

Section 16.2.1.4

Added a commitment for the ASME Section XI Subsection IWL Inservice Inspection Program to inspect containment reinforced concrete above groundwater for concrete degradation (Reference RAI 3.6.2.1-2 response, FPL letter L-2001-61 dated March 30, 2001, as modified by this letter. See Attachment 5)

Section 16.2.2

Included a description of applicable aging effects of boraflex and the program enhancement to perform periodic areal density testing (Confirmatory Item 3.0-1, FSAR Item 3.9.2-1).

Section 16.2.8

Added a commitment to perform testing of wet pipe sprinkler heads following the guidance of NFPA 25 commencing in the year 2022 (Reference RAI 3.9.8-3 response, FPL letter L-2001-75 dated April 19, 2001).

Section 16.2.12

The scope of the Turkey Point Reactor Vessel Head Alloy 600 Penetration Inspection Program described in LRA Subsection 3.2.12 (page B-70) has been revised to document continued participation in industry programs for managing primary water stress corrosion cracking of reactor vessel head penetrations during the period of extended operation (Open Item 3.9.12-1).

Section 16.3.1.3

Revised to address items identified in the NRC Safety Evaluation for Turkey Point Technical Specification Amendments 208/202, issued October 30, 2000. Specifically, this change will ensure that Chemistry Factor for the reactor pressure vessel weld, as discussed in Regulatory Guide 1.99, Revision 2, Position 2.1, is considered in submittal of the 48 EFPY Pressure-Temperature curves. Also, this subsection is being revised to ensure that reactor vessel circumferential weld (heat number 72442) is tracked and considered in future submittals (Confirmatory Item 3.0-1, FSAR Item 4.2-1).

Section 16.3.2.5

This section is being revised to include commitment options identified in the evaluations for the pressurizer surge lines, reactor pressure vessel outlet nozzles and reactor pressure vessel shell at the core support pads, and pressurizer spray nozzles (Confirmatory Item 3.0-1, FSAR Item 4.3-1).

Appendix A

UPDATED
FINAL SAFETY ANALYSIS REPORT
SUPPLEMENT

INTRODUCTION

This appendix contains the Updated FSAR (UFSAR) Supplement required by 10 CFR 54.21(d) for the Turkey Point Units 3 and 4 License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Chapter 3 and Appendix B of the Turkey Point LRA provide descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Chapter 4 of the LRA contains the evaluations of the time-limited aging analyses for the period of extended operation. These LRA sections have been used to prepare the program and activity descriptions that are contained in the UFSAR Supplement. The UFSAR Supplement will be incorporated into the Turkey Point Units 3 and 4 UFSAR following issuance of the renewed operating licenses for Turkey Point. Upon inclusion of the UFSAR Supplement in the Turkey Point UFSAR, changes to the descriptions of the programs and activities for their implementation will be made in accordance with 10 CFR 50.59.

UFSAR CHAPTER 4.0 CHANGES

For the combination of normal plus design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal plus no-loss-of-function earthquake loadings, the stresses in the support structures are limited to values necessary to ensure their integrity, and to keep the stresses in the Reactor Coolant System components within the allowable limits as given in Appendix 5A.

4.1.5 CYCLIC LOADS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor coolant system temperature and pressure changes. These cyclic loads are introduced by normal power changes, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes and their bases are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes and are not intended to be an exact representation of actual transients or actual operating experience. For example the number of cycles for unit heatup and cooldown at 100°F per hour was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss of flow and loss of load transients are not included in Table 4.1-8 since the tabulation is only intended to represent normal design transients, the effect of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

over the range from 15% full power up to but not exceeding 100% of full power, the Reactor Coolant System and its components are designed to accommodate 10% of full power step changes in unit load and 5% of full power per minute ramp changes without reactor trip. The turbine bypass and steam dump system make it possible to accept a step load decrease of 50% of full power without reactor trip.

4.1.6 SERVICE LIFE

The service life of Reactor Coolant System pressure components depends upon the material irradiation, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to any appreciable material irradiation effects. The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM-E 185 standards.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as result of operations such as leak testing and heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (part III), Boiler and Pressure Vessel Code for Class "A" Vessels, the unit operating conditions have been established for the initial 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients. The evaluation for extended plant design life concludes that the <u>40-year</u> design cycles envelope the <u>60-year</u> extended design life.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

TABLE 4.1-8

DESIGN THERMAL AND LOADING CYCLES - 60 YEARS

	Transient Design Condition	Design Cycles
1.	Station heatup at 100°F per hour	200
2.	Station cooldown at 100°F per hour	200
3.	Station loading at 5% of full power/min	14,500
4.	Station unloading at 5% of full power/min	14,500
5.	Step load increase of 10% of full power (but not to exceed full power)	2000
6.	Step load decrease of 10% of full power	2,000
7.	Step load decrease of 50% of full power	200
8.	Reactor trip	400
9.	Hydrostatic test at 3107 psig pressure, 100°F temperature	1(3)
10.	Hydrostatic test at 2435 psig pressure and 400°F temperature	5(4)-
11.	Steady state fluctuations	∞(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.

Rev. [LATER]

The techniques used to measure and predict the integrated fast neutron (E > 1 Mev) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron (E > 1 Mev) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the sample can be applied with confidence to the adjacent section of reactor vessel, the maximum vessel exposure will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal

neutron flux variation.

The maximum integrated fast neutron (E>1 Mev) exposure of the vessel was computed to be 5.1 x 10^{19} n/cm^2 for 40 years of operation at 2300 Mwt at 80 percent load factor. After flux reduction was imposed in 1984 and after thermal uprating performed in 1995, the maximum vessel exposure at the limiting circumferential vessel weld is predicted to be 4.5 x 10^{19} n/cm^2 at the end of the extended license terms (48 EFPY* approximately) (Reference 7). The predicted extended end of life RT(ndt) is less than the 10CFR50.61 screening criteria (Reference 6).

To evaluate the RT(ndt) shift of welds, heat affected zones and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4A.

* This value is approximate and will change from year to year based on the unit availability. Fluence prediction is acceptable in the ±20% range, so this value can easily vary within that limit.

4.2.13 REFERENCES

1. Westinghouse Electric Corporation, Report Number STC-TR-85-003 dated February 8, 1985, "Structural Evaluation - Pressurizer Surge Line and Spray Line for Pressurizer/RCS Differential Temperature of 320°F," PROPRIETARY.

2. Safety Evaluation, JPE-M-85-013, dated June 13, 1985, "Increased ΔT between Pressurizer and Reactor Coolant System to 320°F for PTP Unit 3."

3. NRC Letter, from G.E. Edison (NRC) to W.F. Conway (FPL), "Turkey Point Units 3 and 4 - Generic Letter 84-04, Asymmetric LOCA Loads," dated November 28, 1988.

4. NRC Letter, from R. P. Croteau (NRC) to J. H. Goldberg (FPL), "Turkey Point Units 3 and 4 - Approval to Utilize Leak-Before-Break Methodology for Reactor Coolant System Piping (TAC Nos. M91494 and M91495)," dated June 23, 1995.

5. Westinghouse WCAP-14237, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Turkey Point Units 3 and 4 Nuclear Power Plants," dated December 1994.

6. Westinghouse WCAP-14291, "Turkey Point Units 3 and 4 Uprating Engineering Report Volume 2," dated December 1995.

7. Westinghouse WCAP-15092, Revision 3, "Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," dated May 2000.

met is the more restrictive of a), the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90% of the material yield stress at the operating temperature; or b), the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving maximum stress, do not exceed 135% of the material yield stress at operating temperature. This use of these stress criteria for this abnormal operation is consistent with the ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, paragraph N 714.2 hydrotest stress criteria. The stresses and stress factors in the actual design tube sheet, obtained using the above stress criteria, are given in Table 4.3-3.

The tube sheet designed on the above basis meets code allowable stresses for a primary to secondary differential pressure of 1520 psi. The normal operating differential pressure is 1475 psi.

The tubes have been designed to the requirements (including stress limitations) of Section III for normal operation, assuming 2485 psi as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

No significant corrosion of the Inconel tubing is expected during the lifetime of the unit. The corrosion rate reported in Reference (4), (4) shows "worst case" rates of 15.9 mg/dm^2 in the 2000 hour test under steam generator operating conditions. Conversion of this rate to a 60-year unit life gives a corrosion loss of less than 2.25×10^{-3} inches which is insignificant compared to the nominal tube wall thickness of 0.050 inches.

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1100 psi. This pressure differential is less than the primary-secondary pressure differential capability

TABLE 4.4-2

SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE
TURKEY POINT UNITS 3 & 4

Capsule[4]	Capsule Location (Degree)	Updated Lead Factor	Removal EFPY[1]	Capsule Fluence (n/cm^2)
T_3[2]	270	2.60	1.15	7.39×10^{18}
T_4[2]	270	2.48	1.17	7.08×10^{18}
S_4[2]	280	1.60	3.41	1.43×10^{19}
S_3[2]	280	1.96	3.46	1.72×10^{19}
V_3[2]	290	0.75	8.06	1.53×10^{19}
X_3[3]	270	2.48	19.4 (29 years)	2.74×10^{19}
X_4[3]	270	2.48	24.0 (34 years)	3.85×10^{19}
Y_3	150	0.49	Standby	--
U_3	30	0.49	Standby	--
W_3	40	0.34	Standby	--
Z_3	230	0.34	Standby	--
V_4	290	0.79	Standby[5]	--
Y_4	150	0.49	Standby	--
U_4	30	0.49	Standby	--
W_4	40	0.34	Standby	--
Z_4	230	0.34	Standby	--

NOTES:

(1) Effective Full Power Years (EFPY) from plant startup.

(2) Plant specific evaluation.

(3) Since the vessel controlling material is the weld metal, and only Capsule V from Unit 4 and Capsules X from Units 3 and 4 contain weld specimens, Capsule X in Units 3 and 4 were moved to the 270° location to increase the lead factor.

(4) Unit designation shown in subscript.

(5) Standby end of life capsule, as needed.

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5.1.3 CONTAINMENT DESIGN ANALYSES

This section discusses analytical techniques, references and design philosophy for the containment building design/analyses. The results of the original analyses and the 1994 re-analysis are provided in Section 5.1.4 and Appendix 5H, respectively. The original design criteria, analyses, and construction drawings have been reviewed by Bechtel's consultants, T. Y. Lin, Kulka, Yang & Associate.

Original Analysis

The original containment structure analyses fall into two parts, axisymmetric and non-axisymetric. The axisymmetric analysis is performed through the use of a finite element computer program for the individual loads and is described in Section 5.1.3.1. The axisymmetric finite element approximation of the containment structure shell does not consider the buttresses, penetrations, brackets and anchors. These items of configuration, and lateral loads due to earthquakes or winds, and any concentrated loads, are considered in the non-axisymmetric analysis described in Section 5.1.3.2.

1994 Re-analysis

During the performance of the 20th year tendon surveillance of the Turkey Point Units 3 and 4 containment structure post-tensioning systems, a number of measured normalized tendon lift-off forces were below the predicted lower limit (PLL). Evaluation of the 20th year surveillance results concluded that the probable cause for the low tendon lift-off forces was due to an increased tendon wire steel relaxation loss caused by average tendon temperatures higher than originally considered. The evaluations also concluded that the containment post-tensioning system will provide sufficient prestress force to maintain Turkey Point licensing basis requirements through the 25th year tendon surveillance. The evaluations recommended that a structural re-analysis of the containment structure be performed to determine the minimum required prestress forces, and to establish that the containment structure will continue to meet the licensing basis requirements through the end of the licensed plant 40-year life (see Appendix 5H for additional detail).

A containment structure re-analysis was completed in 1994 and Safety Evaluation JPN-PTN-SECJ-94-027 (Reference 9) has been performed to document the results of this re-analysis.

The containment re-analysis used a three dimensional (3-D) finite element model of the containment structure. The 3-D model consisted of the cylindrical wall (including buttresses), ring girder, dome, base slab, and the major penetrations (equipment hatch and personnel hatch). The containment re-analysis did not include a new evaluation of the base slab since it was not affected by the post-tensioning system. The base slab was included in the 3-D model to provide a realistic boundary condition for the model.

Appendix 5H provides a summary of the containment re-analysis methodology, analytical techniques, references, and results.

The portions of Sections 5.1.3 and 5.1.4 relative to the original analysis of the containment structure which are affected by the 1994 re-analysis (see Appendix 5H) are annotated in the pertinent sections.

License Renewal Analysis

During the License Renewal process, the Turkey Point Units 3 and 4 containment tendons were analyzed for a 60-year life. The analysis concluded that the containment tendons will continue to meet the licensing basis requirements through the licensed plant 60-year life. (Subsection 16.3.4)

5.1.3.1 Axisymmetric Analysis (original analysis)

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Standard conventional analysis of frames and trusses can be considered to be examples of the finite element method. In the application of the method to an axisymmetric solid (e.g., a concrete containment structure) the continuous structure is replaced by a system of rings of triangular cross-section which are interconnected along circumferential joints. Based on energy principles, work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are the unknowns. The results of the solution of this set of equations is the deformation of the structure under the given loading

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Assuming that the jacking stress for the tendons is 0.80 f'_s or 192,000 psi and using the above prestress loss parameters, the following tabulation shows the magnitude of the design losses and the final effective prestress at end of 40 years for a typical dome, hoop, and vertical tendon.[5]

	Dome (Ksi)	Hoop (Ksi)	Vertical (Ksi)	Allowable (Ksi)
Temporary Jacking Stress	192	192	192	192
Friction Loss	19	21.3[1]	21	
Seating Loss	-	0	0	
Elastic Loss (average)	14.7	15.3	6.6	
Creep Loss	19.2	19.2	19.2[4]	
Shrinkage Loss	3.0	3.0	3.0	
Relaxation Loss[3]	12.5	12.5	12.5	
Final Effective Stress [2]	123.6	120.7	129.7	144.0

(1) Average of adjacent tendons

(2) This force does not include the effect of pressurization which increases the prestress force.

(3) See footnote (1) in listing at beginning of Section 5.1.4.4.

(4) To determine tendon surveillance lift-off acceptance criteria, the creep loss for the vertical tendons has been adjusted. For further details, see Reference 11 of safety evaluation JPN-PTN-SECJ-94-027 (Reference 9 on Page 5.1.3-38).

(5) The <u>40-year</u> prestress losses depicted in the tabulation were utilized to calculate <u>60-year</u> prestress losses for license renewal.

To provide assurance, of achievement of the desired level of Final Effective Prestress and that ACI 318-63 requirements are met, a written procedure was prepared for guidance of post-tensioning work. The procedures provided nominal values for end anchor forces in terms of pressure gage readings for calibrated jack-gage combinations. Force measurements were made at the end anchor, of course, since that is the only practical location for such measurements.

5.1.7.4 Tendon Surveillance

Provisions are made for an in-service tendon surveillance program, throughout the life of the plant that will maintain confidence in the integrity of the containment structure.
(See Subsection 16.2.1.4 for program description relating to license renewal.)

The following quantity of tendons have been provided over and above the structural requirements:

Horizontal - Three 120 degree tendons comprising one complete hoop system.

Vertical - Three tendons spaced approximately 120 degrees apart.

Dome - Three tendons spaced approximately 120 degrees apart.

Beginning with the twentieth year tendon surveillance, inspections and lift-off readings are performed on five horizontal, four vertical, and three dome tendons. The tendons chosen for surveillance are a random but representative sample.

The surveillance program for structural integrity and corrosion protection consists of the following operations to be performed during each inspection:

(a) Lift-off readings will be taken for all of the twelve tendons.

(b) One tendon of each directional group will be relaxed and one wire from each relaxed tendon will be removed as samples for inspection. Since these tendons are re-tensioned to their original lift-off forces these samples need not be replaced.

(c) After the inspection, the tendons will be retensioned to the stress level measured at the lift-off reading and then checked by a final lift-off reading.

(d) Should the inspection of one of the wires reveal any significant corrosion (pitting, or loss of area), further inspection of the other two sets will be made to determine the extent of the corrosion and its significance to the load-carrying capacity of the structure. Samples of corroded wire will be tested to failure to evaluate the effects of any corrosion on the tensile strength of the wire.

The inspection of the four vertical tendons in the wall is sufficient to indicate any tendon corrosion that could possibly appear longitudinally along the full height of the structure. Therefore, the twelve tendons arranged as described will provide adequate corrosion surveillance.

The anchorage details permit some degree of accessibility for inspection of all tendons in the containment structure. Corrective action will be taken if and when so indicated by the surveillance program, and an adequate containment structure will be maintained throughout the life of the plant.

The following steps are taken to protect the tendons and the reinforcing steel in the containment structure from corrosion due to stray current and moisture environment.

A tendon protection sheathing filler compound encloses the whole length of every tendon. This compound will not deteriorate during the -life of the unit. As its chemical composition is about 98% petroleum jelly, it will possess the normal stability of the linear hydrocarbons subjected to normal ambient temperature levels. The electrical resistivity of the compound is relatively high. This prevents the possibility of galvanic corrosion that would be detrimental to the tendons. Anodic corrosion centers that could develop on the surface of tendons surrounded by a good electrolyte material will not form in the presence of the protective sheathing filler.

All metallic components such as the tendon trumplate, reinforcing bars and liner plate are interconnected to form an electrically continuous cathodic structure, thereby avoiding inherent difficulties associated with isolation and interference of these members. This interconnection of the steel work with the liner plate ensures that cathodic protection currents will not be allowed to flow through any isolated member to cause electrolytic corrosion.

the combination of normal loads and design earthquake loading. Critical equipment needed for this purpose is required to operate within normal design limits.

In the case of the maximum hypothetical earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., shut the unit down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in Table 5A-1. No rupture of a Class I pipe is caused by the occurrence of the maximum hypothetical earthquake.

Careful design and thorough quality control during manufacture and construction and inspection during unit life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. Leak-Before-Break (LBB) criteria has been applied to the reactor coolant system piping based on fracture mechanics technology and material toughness. That evaluation, together with the leak detection system, demonstrates that the dynamic effects of postulated primary loop pipe ruptures may be eliminated from the design basis (Reference 5A-2). This Leak-Before-Break evaluation was approved by the NRC for use at Turkey Point (Reference 5A-5). This evaluation has been revised for the period of extended operation, as discussed in Subsection 16.3.8.

5A-1.3.2.2 Reactor Vessel Internals

5A-1.3.2.2.1 Reactor Vessel Internals Design Criteria

The internals and core are designed for normal operating conditions and subjected to load of mechanical, hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in the ASME Boiler and Pressure Vessel Code, Section III, Article 4, have been adopted as a guide for the design of the internals and core with the exception of those fabrication techniques and materials which are not covered by the Code. Earthquake stresses are combined in the most conservative way and are considered primary stresses.

to accommodate the forces exerted by the restrained liner plate, and that careful attention be paid to details at corners and connections to minimize the effects of discontinuities.

The most appropriate basis for establishing allowable liner plate strains is considered to be that portion of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically the following sections are adopted as guides in establishing the allowable strain limits:

Paragraph N 412 (m)	Thermal Stress
Paragraph N414.5	Peak Stress Intensity
	Table N 413
	Figure N 414, N 415 (A)
Paragraph N 412 (n)	
Paragraph N 415.1	

Implementation of the ASME Code requires that the liner material be prevented from experiencing significant distortion due to thermal load and that the stresses be considered from a fatigue standpoint. (Paragraph N412 (m) (2)).

The following fatigue loads are considered in the <u>60-year</u> design analysis of the liner plate (See Subsection 16.3.5 for additional details):

(a) Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 60 cycles for the unit life of 60 years.

(b) Thermal cycling due to the containment interior temperature variation during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500 cycles.

(c) Thermal cycling due to the MHA will be assumed to be one cycle.

(d) Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III fatigue considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the 60-year unit life.

The thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, of the ASME Boiler and Pressure Vessel Code. The allowable stresses in Figure N-415 (A) are for alternating stress intensity for carbon steel and temperatures not exceeding 700°F.

In accordance with ASME Code Paragraph N412 (m) 2, the liner plate is restrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700°F and also satisfies the criteria for limiting strains on the basis of fatigue consideration. Paragraph N412 (n) Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels, and it is a part of recognized design code. Figure N-415 (A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below 10 cycles, 10 cycles is being used for MHA instead of one cycle.

The maximum compressive strains are caused by accident pressure, thermal loading prestress, shrinkage and creep. The maximum strains do not exceed .0025 in/in and the liner plate always remains in a stable condition.

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Testing of Operational Sequence of Air Cleanup Systems

Criterion: A capability shall be provided to test initially under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability. (GDC 65)

Means are provided to test initially under conditions as close to design as is practical, the full operational sequence that would bring the Emergency Containment Filtering System into action, including transfer to the emergency diesel-generator power source.

6.3.6 MOTORS FOR EMERGENCY CONTAINMENT FANS

General

These totally enclosed fan cooled motors will have a useful life of ~~forty (40)~~ sixty (60) years under the normal containment service conditions as demonstrated by the appropriate EQ documentation package (See Appendix 8A). Internal heaters will dispel moisture condensation when motor is idle.

Insulation

The insulation will be a special Class B suitable for MHA conditions. The insulation system is described in Table 6.3-2.

Bearings

The bearings will be specially selected, conservatively rated ball bearings

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Environments in which radiation is the only parameter of concern are considered to be mild if the total radiation dose (includes 60-year normal dose plus the post accident dose) is 1.0E5 rads or less. This value is the threshold for evaluation and consideration based on EPRI NP-2129. However, certain solid state electronic components and components that utilize teflon are considered to be in a mild environment only if total radiation dose is 1.0E3 rads or less.

For additional detail on the identification of environmental conditions refer to Equipment Qualification Documentation Package (Doc Pac) 1001, "Generic Approach and Treatment of Issues."

8A.5 MAINTENANCE

The purpose of the Turkey Point Equipment Qualification Maintenance Program is the preservation of the qualification of systems, structures and components. In order to accomplish this task, the plants have developed approved Design Control, Procurement and Maintenance Procedures. In addition, the component specific documentation package contains the equipment's qualified life. The qualified life is developed based upon the qualification test report reviewed in conjunction with the environmental parameters associated with the area. After this review is completed a qualified life is established. Maintenance activities to be performed in addition to the vendor recommended maintenance are determined to ensure that qualification of each piece of equipment is maintained throughout its qualified life.

8A.6 RECORDS/QUALITY ASSURANCE

A documentation package is prepared for the qualification of each manufacturer's piece of equipment under the auspices of 10CFR50.49. This package contains the information, analysis and justifications necessary to demonstrate that the equipment is properly and validly qualified as defined in 10CFR50.49 for the environmental effects of 60 years of service plus a design basis accident.

This documentation package is developed from the criteria stipulated in Doc Pac 1001.

A complete listing of equipment under the auspices of 10CFR50.49 is maintained.

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TABLE 9.2-2

NOMINAL CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE [1]

Unit design life, years	~~40~~ 60
Seal water supply flow rate, gpm [2]	24
Seal water return flow rate, gpm	9
Normal letdown flow rate, gpm	60
Maximum letdown flow rate, gpm	120
Normal charging pump flow (one pump), gpm	69
Normal charging line flow, gpm	45
Maximum rate of boration with one transfer and one charging pump from an initial RCS concentration of 1800 ppm, ppm/min	5.4
Equivalent cooldown rate to above rate of boration, °F/min	1.5
Maximum rate of boron dilution with two charging pumps from an initial RCS concentration of 2500 ppm, ppm/hour	350
Two-pump rate of boration, using refueling water, from initial RCS concentration of 10 ppm, ppm/min	6.2
Equivalent cooldown rate to above rate of boration, °F/min	1.7
Temperature of reactor coolant entering system at full power (design), °F	555.0
Temperature of coolant return to reactor coolant system at full power (design), °F	493.0
Normal coolant discharge temperature to holdup tanks, °F	127.0
Amount of 3.0 weight percent boron solution required to meet cold shutdown requirements, at end of life with peak xenon (including consideration for one stuck rod) gallons	7500

NOTES :

1. Reactor coolant water quality is given in Table 4.2-2.
2. Volumetric flow rates in gpm are based on 130°F and 2350 psig.

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TABLE 11.1-1

WASTE DISPOSAL SYSTEM
PERFORMANCE DATA
(Two Units)

Plant Design Life	~~40~~ 60 years
Normal process capacity, liquids	Table 11.1-3
Evaporator load factor	Table 11.1-4
Annual liquid discharge	
Volume	Table 11.1-4
Activity	
Tritium	Table 11.1-5
Other	Table 11.1-5
Annual gaseous discharge	
Activity	Table 11.1-6

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The neutron absorber rack design includes a poison verification view-hole in the cell wall so that the presence of poison material may be visually confirmed at any time over the life of the racks. Upon completion of rack fabrication, such an inspection was performed. This visual inspection, coupled with the Westinghouse quality assurance program controls and the use of qualified Boraflex neutron absorbing material, satisfies an initial verification test to assure that the proper quantity and placement of material was achieved during fabrication of the racks. This precludes the necessity for on-site poison verification.

As discussed in Section 4.7.2, irradiation tests have been previously performed to test the stability and structural integrity of Boraflex in boric acid solution under irradiation[7]. These tests have concluded that there is no evidence of deterioration of the suitability of the Boraflex poison material through a cumulative irradiation in excess of 1 x 10^{11} rads gamma radiation. As more data on the service life performance of Boraflex becomes available in the nuclear industry in the coming years through both experimentation and operating experience, FPL will evaluate this information and will take action accordingly. (See Subsection 16.2.2 for a program description relating to License Renewal.)

[NEW]
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[NEW CHAPTER 16]

16.0 AGING MANAGEMENT PROGRAMS AND TIME-LIMITED AGING ANALYSES ACTIVITIES

The integrated plant assessment for license renewal identified existing and new aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This chapter describes these programs and their planned implementation.

FPL has established and implemented a Quality Assurance Program to provide assurance that the design, procurement, modification and operation of nuclear power plants conform to applicable regulatory requirements. The FPL Quality Assurance Program, described in the FPL Topical Quality Assurance Report, is in compliance with the requirements of 10 CFR 50, Appendix B. The FPL Quality Assurance Program meets the requirements provided by regulatory guidance and industry standards as listed in Appendix C of the FPL Topical Quality Assurance Report. Corrective Actions, Confirmatory Actions, and Administrative Controls apply to all aging management programs credited for license renewal and performed, or in the case of new programs, to be performed, in accordance with the FPL Quality Assurance Program.

This chapter also discusses the evaluation results for each of the plant-specific time-limited aging analyses performed for license renewal. The evaluations have demonstrated that the analyses remain valid for the period of extended operation; the analyses have been projected to the end of the period of extended operation; or that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

No 10 CFR 50.12 exemptions involving a time-limited aging analysis as defined in 10 CFR 54.3 were identified for Turkey Point.

16.1 NEW PROGRAMS

16.1.1 AUXILIARY FEEDWATER PUMP OIL COOLERS INSPECTION

The cast iron parts of the auxiliary feedwater pumps lube oil coolers and turbine governor controller oil coolers, which are wetted internally by auxiliary feedwater, are potentially susceptible to graphitic corrosion (i.e., selective leaching) and other types of corrosion. A one-time visual inspection will be performed on one of the cast iron bonnets of the auxiliary feedwater pump lube oil coolers to assess the extent of loss of material due to corrosion. The results of this inspection will be evaluated to determine the need for additional inspections/programmatic corrective actions. This inspection and evaluation will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.2 AUXILIARY FEEDWATER STEAM PIPING INSPECTION PROGRAM

The Auxiliary Feedwater Steam Piping Inspection Program manages the aging effects of loss of material due to general and pitting corrosion on the internal and external surfaces of carbon steel auxiliary feedwater steam supply lines. Periodic volumetric examinations of representative auxiliary feedwater steam supply components will be performed to ensure that minimum required wall thickness is maintained. Examinations will be performed on piping/fittings and other components using volumetric techniques, such as ultrasonic or computed radiography. The inspections will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.3 EMERGENCY CONTAINMENT COOLERS INSPECTION

A one-time volumetric examination of a sample of emergency containment coolers (ECC) tubes will be performed to determine the extent of loss of material due to erosion in the ECC tubes. The results of this inspection will be evaluated to determine the need for additional inspections/programmatic corrective actions. This inspection and evaluation will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.4 FIELD ERECTED TANKS INTERNAL INSPECTION

A one-time visual inspection to determine the extent of corrosion on the internal surfaces of the field erected tanks for both units -- including the Condensate Storage Tanks, the Demineralized Water Storage Tank, and the Refueling Water Storage

Tanks -- will be performed. The results of these inspections will be evaluated to determine the need for additional inspections/programmatic corrective actions. These inspections will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.5 GALVANIC CORROSION SUSCEPTIBILITY INSPECTION PROGRAM

The Galvanic Corrosion Susceptibility Inspection Program manages the aging effect of loss of material due to galvanic corrosion on the internal surfaces of susceptible piping and components. The program involves selected, one-time inspections on the internal surfaces of piping and components with the greatest susceptibility to galvanic corrosion. Loss of material is expected mainly in carbon steel components directly coupled to stainless steel components in raw water systems, however, baseline examinations in select systems will be performed and evaluated to establish if the corrosion mechanism is active. Based on the results of these inspections, the need for followup examinations or programmatic corrective actions will be established. The program will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.1.6 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel Internals Inspection Program consists of two types of examinations, visual (VT-1 and, if necessary, enhanced VT-1) and ultrasonic testing. The visual examination manages the aging effect of cracking due to irradiation assisted stress corrosion (IASCC) and reduction in fracture toughness due to irradiation and thermal embrittlement. The ultrasonic testing examination manages the aging effect of loss of mechanical closure integrity of reactor vessel internals bolting. The program, including an evaluation of program scope with regard to dimensional changes due to void swelling, will be in place prior to the end of the initial operating license terms for Turkey Point Units 3 and 4, and the actual visual and ultrasonic examinations, one inspection per unit, will be performed during the period of extended operation.

A report will be submitted to the NRC prior to the end of the initial operating license for Unit 3 summarizing the understanding of aging effects applicable to the reactor vessel internals. The report will contain a description of the Turkey Point inspection plan, including methods for detection and sizing of cracks and acceptance criteria.

The inspections will correspond with the ASME Section XI reactor vessel inspections. In order to develop a baseline for the extended period of operation,

FPL plans to perform the first of these reactor vessel internals inspections early in the renewal period on the unit leading in fluence at that time. The second inspection will be conducted on the other unit at the next 10-year inspection interval during the license renewal term. This will act as a status examination and should provide confidence in the structural integrity for the final ten years of service.

16.1.7 SMALL BORE CLASS 1 PIPING INSPECTION

A volumetric inspection of a sample of small bore Class 1 piping and nozzles will be performed to determine if cracking is an aging effect requiring management during the period of extended operation. This one-time inspection will address Class 1 piping less than 4 inches in diameter. Based on the results of these inspections, the need for additional inspections or programmatic corrective actions will be established. The inspection will be performed prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. A report describing the details of the inspection plan will be submitted to the NRC prior to the implementation of this inspection.

16.1.8 CONTAINMENT CABLE INSPECTION PROGRAM

The Containment Cable Inspection Program manages potential aging of non-EQ cable, connections, and penetrations. This aging management program consists of periodic visual inspection of accessible non-EQ cables, connections and penetrations within the scope of license renewal located in the containment structures that may be installed in adverse localized environments. The inspections will be implemented before the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2 EXISTING PROGRAMS

16.2.1 ASME SECTION XI INSERVICE INSPECTION PROGRAMS

16.2.1.1 ASME SECTION XI, SUBSECTIONS IWB, IWC, AND IWD INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program inspections identify and correct degradation in Class 1, 2, and 3 components and piping. The program manages the aging effects of loss of material, cracking, and loss of mechanical closure integrity. The program provides inspection and examination of accessible components, including welds, pump casings, valve bodies, steam generator tubing, and pressure-retaining bolting. This program will be enhanced to require VT-1 examinations of the core support lugs during the period of extended operation.

16.2.1.2 ASME SECTION XI, SUBSECTION IWE INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsection IWE Inservice Inspection Program inspections identify and correct degradation of pressure retaining components and their integral attachments and the metallic liner of Class CC pressure-retaining components and their integral attachments. The program manages the aging effects of loss of material and loss of pressure retention. The program provides inspection and examination of containment surfaces, seals, gaskets and moisture barriers, pressure-retaining bolting, and pressure retaining components in accordance with the requirements of ASME Section XI, Subsection IWE.

16.2.1.3 ASME SECTION XI, SUBSECTION IWF INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsection IWF Inservice Inspection Program inspections identify and correct degradation of ASME Class 1, 2, and 3 component supports. This program manages the aging effect of loss of material. The scope of the program provides for inspection and examination of accessible surface areas of the component supports in accordance with the requirements of ASME Section XI, Subsection IWF.

16.2.1.4 ASME SECTION XI, SUBSECTION IWL INSERVICE INSPECTION PROGRAM

ASME Section XI, Subsection IWL Inservice Inspection Program inspections assess the quality and structural performance of the Containment structure post-tensioning

system components. The program manages the aging effects of loss of material and confirms the results of the Containment tendon loss of prestress Time-Limited Aging Analysis (see Subsection 16.3.4). The program includes inspection of tendon and anchorage hardware surfaces and measurement of tendon force and elongation. The program also includes inspection of Containment reinforced concrete above groundwater for evidence of concrete degradation.

16.2.2 BORAFLEX SURVEILLANCE PROGRAM

The Boraflex Surveillance Program manages the aging effect of change in material properties (including shrinkage, gap formation, and dissolution) for the Boraflex material in the spent fuel storage racks.

The program includes periodic areal density testing (or other approved testing methods if available) of the encapsulated Boraflex material in the spent fuel storage racks to provide detailed information on the condition of the panels. The frequency of the Boraflex condition monitoring surveillance ensures timely detection of degradation impacting intended function and is consistent with regulatory commitments.

16.2.3 BORIC ACID WASTAGE SURVEILLANCE PROGRAM

The Boric Acid Wastage Surveillance Program manages the aging effects of loss of material and mechanical closure integrity due to aggressive chemical attack resulting from borated water leaks. The program addresses the Reactor Coolant System and structures and components containing, or exposed to, borated water. This program utilizes systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of pressure boundary or structural integrity of components, supports, or structures, including electrical equipment in proximity to borated water systems. This program includes commitments to NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants."

Some systems outside Containment (i.e., Spent Fuel Pool Cooling and portions of Waste Disposal associated with containment integrity) are currently inspected under other existing programs. The scope of the Boric Acid Wastage Surveillance Program will be enhanced to include these systems and components prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.4 CHEMISTRY CONTROL PROGRAM

The Chemistry Control Program manages loss of material, cracking, and fouling aging effects for primary and secondary systems, structures, and components. The aging effects are minimized or prevented by controlling the chemical species that cause the underlying mechanism(s) that results in these aging effects. Alternatively, chemical agents, such as corrosion inhibitors and biocides, are introduced to prevent certain aging mechanisms. The program includes sampling activities and analysis. The program provides assurance that an elevated level of contaminants and oxygen does not exist in the systems, structures, and components covered by the program, and thus prevents and minimizes the occurrences of aging effects.

16.2.5 CONTAINMENT SPRAY SYSTEM PIPING INSPECTION PROGRAM

The Containment Spray System Piping Inspection Program manages the aging effect loss of material due to general, crevice, and pitting corrosion on the internal surfaces of carbon steel piping and fittings, and valves wetted by boric acid in the Containment Spray System spray headers. Periodic ultrasonic examinations of selected locations are used to determine wall thickness and are evaluated to ensure that minimum thickness requirements are maintained.

16.2.6 ENVIRONMENTAL QUALIFICATION PROGRAM

The Environmental Qualification Program evaluations of electrical equipment are identified as Time-Limited Aging Analyses. Equipment covered by the Environmental Qualification Program has been evaluated to determine if the existing Environmental Qualification aging analyses can be projected to the end of the period of extended operation by reanalysis or additional analysis. Qualification into the license renewal period is treated the same as equipment initially qualified for 40 years or less. When analysis cannot justify a qualified life in excess of the license renewal period, then the component parts will be replaced, refurbished, or requalified prior to exceeding the qualified life in accordance with the Environmental Qualification Program.

16.2.7 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program is designed to track design cycles to ensure that Reactor Coolant System components remain within their design fatigue limits. Design cycle limits for Turkey Point Units 3 and 4 are provided in Table 4.1-8. The specific fatigue analyses validated by the Fatigue Monitoring Program are

associated with the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines. Administrative procedures provide the methodology for logging design cycles. Guidance is provided in the event design cycle limits are approached.

16.2.8 FIRE PROTECTION PROGRAM

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

The scope of the Fire Protection Program will be enhanced to include inspection of additional components prior to the end of the initial operating license terms for Turkey Point Units 3 and 4. Additionally, Turkey Point will perform testing of wet pipe sprinkler heads following the guidance of NFPA 25 commencing in the year 2022.

16.2.9 FLOW ACCELERATED CORROSION PROGRAM

The Flow Accelerated Corrosion Program manages the aging effect of loss of material due to flow accelerated corrosion. The Flow Accelerated Corrosion Program predicts, detects, monitors, and mitigates flow accelerated corrosion wear in high energy carbon steel piping associated with the Main Steam and Turbine Generators, and Feedwater and Blowdown Systems, and is based on industry guidelines and experience. The program includes analysis and baseline inspections; determination, evaluation, and corrective actions for affected components; and follow-up inspections.

This program will be enhanced to address internal and external loss of material of steam trap lines due to flow accelerated corrosion and general corrosion, respectively, prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.10 INTAKE COOLING WATER SYSTEM INSPECTION PROGRAM

The Intake Cooling Water System Inspection Program manages the aging effects of loss of material due to various corrosion mechanisms, stress corrosion cracking,

and biological fouling for Intake Cooling Water System components. The program includes inspections, performance testing, evaluations, and corrective actions that are performed as the result of FPL commitments to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

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

16.2.11 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program manages the aging effects of loss of material, cracking, fouling buildup, loss of seal, and embrittlement for systems, structures, and components. The scope of the program provides for visual inspection and examination of selected surfaces of specific components and structural components. The program also includes leak inspection of limited portions of the Chemical and Volume Control Systems. Additionally, the program provides for replacement/refurbishment of selected components on a specified frequency, as appropriate.

Specific enhancements to the scope and documentation of some inspections performed under this program will be implemented prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.12 REACTOR VESSEL HEAD ALLOY 600 PENETRATION INSPECTION PROGRAM

The Reactor Vessel Head Alloy 600 Penetration Inspection Program encompasses the Turkey Point Units 3 and 4 reactor vessel head Alloy 600 penetrations that are part of the Reactor Coolant System pressure boundary. This program manages the aging effect of cracking due to primary water stress corrosion (PWSCC). The program includes a one-time volumetric examination of selected Unit 4 reactor vessel head penetrations to detect crack initiation. Visual examination of the Unit 3 and Unit 4 reactor vessel head external surfaces during outages and the Boric Acid Wastage Surveillance Program are also utilized to manage cracking. Turkey Point will continue to participate in industry programs to ensure that PWSCC of the reactor vessel head penetrations is managed for the period of extended operation.

16.2.13 REACTOR VESSEL INTEGRITY PROGRAM

The Reactor Vessel Integrity Program manages reactor vessel irradiation embrittlement and encompasses the following subprograms:

- Reactor Vessel Surveillance Capsule Removal and Evaluation
- Fluence and Uncertainty Calculations
- Monitoring Effective Full Power Years
- Pressure-Temperature Limit Curves

Program documentation will be enhanced to integrate all aspects of the Reactor Vessel Integrity Program prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.13.1 REACTOR VESSEL SURVEILLANCE CAPSULE REMOVAL AND EVALUATION

This subprogram manages the aging effect of reduction in fracture toughness of the Turkey Point Units 3 and 4 reactor vessel materials (beltline forgings and circumferential welds) due to neutron irradiation embrittlement by performing Charpy V-notch and tensile tests on the reactor vessel irradiated specimens. The Reactor Vessel Surveillance Capsule Removal and Evaluation subprogram is a NRC-approved program that meets the requirements of 10 CFR 50, Appendix H. The surveillance capsule withdrawal schedule is specified in Table 4.4-2.

16.2.13.2 FLUENCE AND UNCERTAINTY CALCULATIONS

This subprogram provides an accurate prediction of the Turkey Point Units 3 and 4 reactor vessel accumulated fast neutron fluence values at the reactor vessel beltline forgings and circumferential welds.

16.2.13.3 MONITORING EFFECTIVE FULL POWER YEARS

This subprogram accurately monitors and tabulates the accumulated operating time experienced by the reactor vessels to ensure that the Turkey Point Units 3 and 4 pressure-temperature limit curves and end-of-life reference temperatures are not exceeded.

16.2.13.4 PRESSURE-TEMPERATURE LIMIT CURVES

This subprogram provides pressure-temperature limit curves for the Turkey Point Units 3 and 4 reactor vessels to establish the Reactor Coolant System operating limits. The pressure-temperature limit curves are included in the Technical Specifications.

16.2.14 STEAM GENERATOR INTEGRITY PROGRAM

The Steam Generator Integrity Program ensures steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The program manages the aging effects of cracking and loss of material and includes the following essential elements:

- Inspection of steam generator tubing and tube plugs
- Steam generator secondary-side integrity inspections
- Tube integrity assessments
- Assessment of degradation mechanisms
- Primary-to-secondary leakage monitoring
- Primary and secondary chemistry control
- Sludge lancing
- Maintenance and repairs
- Foreign material exclusion

16.2.15 SYSTEMS AND STRUCTURES MONITORING PROGRAM

The Systems and Structures Monitoring Program manages the aging effects of loss of material, cracking, fouling, loss of seal, and change in material properties. The program provides for periodic visual inspection and examination for degradation of accessible surfaces of specific systems, structures, and components, and corrective actions as required based on these inspections.

This program will be enhanced by restructuring it to address inspection requirements to manage certain aging effects in accordance with 10 CFR 54, modifying the scope of specific inspections, and improving documentation requirements prior to the end of the initial operating license terms for Turkey Point Units 3 and 4.

16.2.16 THIMBLE TUBE INSPECTION PROGRAM

The Thimble Tube Inspection Program manages the aging effect of material loss due to fretting wear. This program consists of an eddy current test inspection of thimble tube N-05 on Unit 3. Eddy current testing of thimble tubes was initiated in response to NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," and inspections have been performed on all in-service thimble tubes for Units 3 and 4. This inspection will be performed prior to the end of the initial operating license term for Turkey Point Unit 3.

16.3 TIME-LIMITED AGING ANALYSIS ACTIVITIES

16.3.1 REACTOR VESSEL IRRADIATION EMBRITTLEMENT

The Turkey Point Units 3 and 4 reactor vessels are described in Chapters 3.0 and 4.0. Time-limited aging analyses (TLAAs) applicable to the reactor vessels are:

- pressurized thermal shock
- upper-shelf energy
- pressure-temperature limits

The Reactor Vessel Integrity Program, described in Subsection 16.2.13, manages reactor vessel irradiation embrittlement utilizing subprograms to monitor, calculate, and evaluate the time-dependent parameters used in the aging analyses for pressurized thermal shock, Charpy upper-shelf energy, and pressure-temperature limit curves to ensure continuing vessel integrity through the period of extended operation.

16.3.1.1 PRESSURIZED THERMAL SHOCK

The requirements in 10 CFR 50.61 provide rules for protection against pressurized thermal shock events for pressurized water reactors. Licensees are required to perform an assessment of the projected values of the maximum nil ductility reference temperature (RTPTS) whenever a significant change occurs in projected values of RTPTS, or upon request for a change in the expiration date for the operation of the facility.

The calculated RTPTS values at the end of the extended period of operation (48 effective full power years) for the Turkey Point Units 3 and 4 reactor vessels are less than the 10 CFR 50.61(b)(2) screening criteria of 270°F for intermediate and lower shells and 300°F for the circumferential welds. Based upon the revised calculations, additional measures will not be required for the Turkey Point reactor vessels during the license renewal period.

The analysis associated with pressurized thermal shock has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.1.2 UPPER-SHELF ENERGY

The requirements on reactor vessel Charpy upper-shelf energy are included in 10 CFR 50, Appendix G. Specifically, 10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing.

A fracture mechanics evaluation was performed in accordance with Appendix K of ASME Section XI to demonstrate continued acceptable equivalent margins of safety against fracture through 48 effective full power years.The analysis associated with upper-shelf energy has been projected to the end of the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.1.3 PRESSURE-TEMPERATURE LIMITS

The requirements in 10 CFR 50, Appendix G, ensure that heatup and cooldown of the reactor pressure vessel are accomplished within established pressure-temperature limits. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure is reduced. Operation of the Reactor Coolant System is also limited by the net positive suction curves for the reactor coolant pumps. These curves specify the minimum pressure required to operate the reactor coolant pumps. Therefore, in order to heatup and cooldown, the reactor coolant temperature and pressure must be maintained within an operating window established between the Appendix G pressure-temperature limits and the net positive suction curves.

To address the period of extended operation, the 48 effective full power year projected fluences and the Turkey Point-specific reactor vessel material properties were used to determine the limiting material and calculate pressure-temperature limits for heatup and cooldown. The limiting material at all temperatures for the period of extended operation is the circumferential girth weld. As discussed in the NRC Safety Evaluation for Technical Specification Amendments 208/202 Turkey Point 32 EFPY Pressure-Temperature Curves, future submittals will ensure the consideration of the Chemistry Factor in accordance with Regulatory Guide 1.99, Revision 2 and that reactor pressure vessel circumferential weld (heat number 72442) is tracked and considered.

A license amendment to incorporate the pressure-temperature limit curves projected to 48 effective full power years will be submitted to the NRC for review and approval prior to exceeding the licensed operating period for these curves.

The analysis associated with reactor vessel pressure-temperature limit curves has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

16.3.2 METAL FATIGUE

The thermal and mechanical fatigue analyses of plant mechanical components have been identified as time-limited aging analyses for Turkey Point. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the Turkey Point UFSAR.

16.3.2.1 ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, CLASS 1 COMPONENTS

The reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the Turkey Point Units 3 and 4 Nuclear Steam Supply System components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various Nuclear Steam Supply System components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycle set applicable to the Class 1 components was assembled. The actual frequency of occurrence for the design cycles was determined and compared to the design cycle set. The severity of the

actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis, the design cycle profiles envelop actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation.

The analyses associated with verifying the structural integrity of the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

As a confirmatory program, the monitoring of plant transients performed as a part of the Fatigue Monitoring Program, as described in Subsection 16.2.7, will assure that the design cycle limits are not exceeded.

16.3.2.2 REACTOR VESSEL UNDERCLAD CRACKING

In early 1971, an anomaly identified as grain boundary separation, perpendicular to the direction of the cladding weld overlay, was identified in the heat-affected zone of reactor vessel base metal. A generic fracture mechanics evaluation demonstrated that the growth of underclad cracks during a 40-year plant life is insignificant.

The evaluation was extended to 60 years using fracture mechanics evaluations based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service life. The 60-year evaluation shows insignificant growth of the underclad cracks.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

16.3.2.3 REACTOR COOLANT PUMP FLYWHEEL

During normal operation, the reactor coolant pump flywheel possesses sufficient kinetic energy to potentially produce high-energy missiles in the unlikely event of failure. Conditions which may result in overspeed of the reactor coolant pump increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack initiation in the flywheel bore keyway. An evaluation of the

probability of failure over the extended period of operation was performed. It demonstrates that the flywheel design has a high structural reliability with a very high flaw tolerance and negligible flaw crack extension over a 60-year service life.

The analysis associated with the structural integrity of the reactor coolant pump flywheel has been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

16.3.2.4 ANSI B31.1 PIPING

The Reactor Coolant System primary loop piping and balance-of-plant piping are designed to the requirements of ANSI B31.1, Power Piping. The exceptions are the Units 3 and 4 pressurizer surge lines and the Unit 4 Emergency Diesel Generator safety-related piping.

The pressurizer surge lines have been designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 1.

The Unit 4 Emergency Diesel Generator safety-related piping has been designed to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 3, which is essentially the same as ANSI B31.1 design requirements. The evaluation of the Unit 4 Emergency Diesel Generator safety-related piping fatigue is, therefore, included in the discussion below.

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and vary operating temperatures only during plant heatup and cooldown, during plant transients, or during periodic testing. The results of the evaluation for ANSI B31.1 piping systems demonstrate that the number of assumed thermal cycles would not be exceeded in 60 years of plant operation.

The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

16.3.2.5 ENVIRONMENTALLY ASSISTED FATIGUE

The Turkey Point approach to address reactor water environmental effects accomplishes two objectives. First, the TLAA on fatigue design has been resolved by confirming that the original transient design limits remain valid for the 60-year operating period. Confirmation by fatigue monitoring will ensure these transient design limits are not exceeded. Second, reactor water environmental effects on fatigue life are examined using the most recent data from laboratory simulation of the reactor coolant environment.

As a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term, Idaho National Engineering Laboratories (INEL) evaluated, in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995, fatigue-sensitive component locations at plants designed by all four U. S. Nuclear Steam Supply System (NSSS) vendors. The pressurized water reactor (PWR) calculations, especially the early-vintage Westinghouse PWR calculations, are directly relevant to Turkey Point. The description of the "Older Vintage Westinghouse Plant" evaluated in NUREG/CR-6260 matches Turkey Point with respect to design code. In addition, the transient cycles considered in the evaluation match or bound Turkey Point design.

NUREG/CR-6260 calculated fatigue usage factors for critical fatigue-sensitive component locations for the early-vintage Westinghouse plant utilizing the interim fatigue curves provided in NUREG/CR-5999, "Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments," August 1993. The results of NUREG/CR-6260 analyses were then utilized to scale up the Turkey Point plant-specific usage factors for the same locations to account for environmental effects. Generic industry studies performed by EPRI and NEI were also considered in this aspect of the evaluation, as well as environmental data that have been collected and published subsequent to the generic industry studies.

For the pressurizer surge line, FPL will 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 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 NRC approved inspection program.

Since actual projected cycle counts were utilized in the reactor pressure vessel outlet nozzles and the reactor pressure vessel shell at the core support pads evaluations, FPL will either:

1. Modify the Fatigue Monitoring Program to limit transient accumulations to the values used in the evaluations, or
2. Perform a more refined evaluation for the reactor pressure vessel outlet nozzles and the reactor pressure vessel shell at the core support pads to show acceptable CUFs for 60 years, or
3. Track CUF values in addition to cycle counts to ensure CUF values remain acceptable.

Since actual projected cycle counts were utilized in the pressurizer spray nozzle evaluations, FPL will either:

1. Modify the Fatigue Monitoring Program to limit transient accumulations to the values used in the evaluations, or
2. Perform a more refined evaluation for the pressurizer spray nozzle to show acceptable CUFs for 60 years, or
3. Track CUF values in addition to cycle counts to ensure CUF values remain acceptable.

16.3.3 ENVIRONMENTAL QUALIFICATION

The thermal, radiation, and wear cycle aging analyses of plant electrical and I&C components have been identified as time-limited aging analyses for Turkey Point. In particular, the environmental qualification evaluations of electrical equipment with a 40-year qualified life or greater have been determined to be time-limited aging analyses.

Equipment included in the Turkey Point Environmental Qualification Program has been evaluated to determine if existing environmental qualification aging analyses

can be projected to the end of the period of extended operation by reanalysis or additional analysis. Qualification into the license renewal period is treated the same as for equipment currently qualified at Turkey Point for 40 years or less. When aging analysis cannot justify a qualified life into the license renewal period, then the component or parts will be replaced prior to exceeding their qualified lives in accordance with the Environmental Qualification Program, as described in Subsection 16.2.6.

Age-related service conditions that are applicable to the environmentally qualified equipment (i.e., 60 years of exposure versus 40 years) were evaluated for the period of extended operation to verify that the current environmental qualification analyses are bounding. The evaluations considered radiation, thermal, and wear cycle aging effects.

Therefore, the analyses associated with the environmental qualification of electrical equipment remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i), or have been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.4 CONTAINMENT TENDON LOSS OF PRESTRESS

The Turkey Point Units 3 and 4 containment buildings are post-tensioned, reinforced concrete structures composed of vertical cylinder walls and a shallow dome, supported on a conventional reinforced concrete base slab. The cylinder walls are provided with vertical tendons and horizontal hoop tendons. The dome is provided with three groups of tendons oriented 120-degrees apart.

The prestress of containment tendons decreases over time as a result of seating of anchorage losses, elastic shortening of concrete, creep of concrete, shrinkage of concrete, relaxation of prestressing steel, and friction losses. New upper limit curves, lower limit curves, and trend lines of measured prestressing forces have been established for all tendons through the period of extended operation. The predicted final effective preload at the end of 60 years exceeds the minimum required preload for all containment tendons. Consequently, the post-tensioning system will continue to perform its intended function throughout the period of extended operation.

The analyses associated with containment tendon loss of prestress have been projected to the end of the period of extended operation, in accordance with the

requirements of 10 CFR 54.21(c)(1)(ii).

As a confirmatory program, the Containment structure post-tensioning system surveillance performed as a part of the ASME Section XI, Subsection IWL Inservice Inspection Program, as described in Subsection 16.2.1.4, will continue to be performed in accordance with the requirements of plant Technical Specifications.

16.3.5 CONTAINMENT LINER PLATE FATIGUE

The interior surface of each Containment is lined with welded steel plate to provide an essentially leak-tight barrier. Design criteria are applied to the liner to assure that the specified allowed leak rate is not exceeded under the design basis accident conditions. The fatigue loads, as described in Appendix 5B, Section B.2.1, were considered in the design of the liner plates and are considered time-limited aging analyses for the purposes of license renewal. Each of these has been evaluated for the period of extended operation.

The number of thermal cycles due to annual outdoor temperature variations was increased from 40 to 60 for the extended period of operation. The effect of this increase is insignificant in comparison to the assumed 500 thermal cycles due to Containment interior temperature varying during heatup and cooldown of the Reactor Coolant System. The 500 thermal cycles includes a margin of 300 thermal cycles above the 200 Reactor Coolant System allowable design heatup and cooldown cycles, which is sufficient margin to accommodate the additional 20 cycles of annual outdoor temperature variation. Therefore, this loading condition is considered valid for the period of extended operation as it is enveloped by the evaluation for 500 thermal cycles.

The assumed 500 thermal cycles was evaluated based on the more limiting heatup and cooldown design cycles (transients) for the Reactor Coolant System. The Reactor Coolant System was designed to withstand 200 heatup and cooldown thermal cycles. The evaluation determined that the originally projected number of maximum Reactor Coolant System design cycles is conservative enough to envelop the projected cycles for the extended period of operation. Therefore, the original containment liner plate fatigue analysis for 500 heatup and cooldown cycles is considered valid for the period of extended operation.

The assumed value of one for thermal cycling due to the maximum hypothetical accident remains valid. No maximum hypothetical accident has occurred and none is expected, therefore, this assumption is considered valid for the period of extended

operation.

The design of the containment penetrations has been reviewed. The design meets the general requirements of the 1965 Edition of ASME Boiler and Pressure Vessel Code, Section III. The main steam piping, feedwater piping, blowdown piping, and letdown piping are the only piping penetrating the containment wall and liner plate that contribute significant thermal loading on the liner plate. The projected number of actual operating cycles for these piping systems through 60 years of operation was determined to be less than the original design limits.

The analyses associated with the containment liner plate 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).

16.3.6 BOTTOM MOUNTED INSTRUMENTATION THIMBLE TUBE WEAR

As discussed in NRC Information Notice No. 87-44, Supplement 1, "Thimble Tube Thinning in Westinghouse Reactors," thimble tubes have experienced thinning as a result of flow-induced vibration. Thimble tube wear results in degradation of the Reactor Coolant System pressure boundary and could potentially create a non-isolable leak of reactor coolant. Therefore, the NRC staff requested that licensees perform the actions described in NRC Bulletin No. 88-09, "Thimble Tube Thinning in Westinghouse Reactors." In response to this bulletin, FPL established a program for inspection and assessment of thimble tube thinning. Turkey Point commitments to the NRC for two eddy current inspections of the thimble tubes for each unit were completed in May 1990 for Unit 4, and in December 1992 for Unit 3. The results demonstrated that the thimble tubes were acceptable for operation and that no appreciable thinning had occurred between the two inspections. Based on the results of the inspections and the analyses performed, only the Unit 3 thimble tube N-05 will require further evaluation for the extended period of operation.

In order to ensure thimble tube reliability, an inspection of Unit 3 thimble tube N-05 will be conducted under the Thimble Tube Inspection Program, described in Subsection 16.2.16. This aging management program will ensure that thimble tube thinning will be adequately managed for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

16.3.7 EMERGENCY CONTAINMENT COOLER TUBE WEAR

The component cooling water flow rate through the emergency containment coolers

could exceed the nominal design flow during certain plant conditions. High flow rates can produce increased wear on the inside surface of the emergency containment cooler tubes. The effect of increased wear was previously evaluated and the tube wall nominal thickness was determined to exceed the minimum required wall thickness during the existing operating period of 40 years. In order to ensure emergency containment cooler tube reliability, a one-time inspection for minimum tube wall thickness will be conducted on a sample of cooler tubes prior to the end of the existing operating period to further assess the actual tube wall thinning. The inspection will be conducted in accordance with the Emergency Containment Coolers Inspection, described in Subsection 16.1.3.

The Emergency Containment Coolers Inspection will ensure that the aging effect of emergency containment cooler tube wear will be adequately managed for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

16.3.8 LEAK-BEFORE-BREAK FOR REACTOR COOLANT SYSTEM PIPING

A plant-specific Leak-Before-Break (LBB) analysis was performed for Turkey Point Units 3 and 4 in 1994. The LBB analysis was performed to show that any potential leaks that develop in the Reactor Coolant System loop piping can be detected by plant monitoring systems before a postulated crack causing the leak would grow to unstable proportions during the 40-year plant life. As documented in the June 23, 1995, NRC letter to FPL (Appendix 5A, Reference 5A-5), the NRC approved the Turkey Point LBB analysis. The NRC safety evaluation concluded that the LBB analysis was consistent with the criteria in NUREG-1061, Volume 3, and the draft Standard Review Plan, Section 3.6.3; therefore, the analysis complied with GDC-4.

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

flaw sizes.

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

The Reactor Coolant System primary loop piping Leak-Before-Break analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

16.3.9 CRANE LOAD CYCLE LIMIT

The crane load cycle limit was identified as a time-limited aging analysis for the cranes within the scope of license renewal. They include the polar cranes, reactor cavity manipulator cranes, spent fuel pool bridge cranes, spent fuel cask crane, turbine gantry crane, and intake structure bridge crane.

The load cycles for these cranes were evaluated for the period of extended operation. For each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed quantity of cycles. All the cranes in the scope of license renewal will continue to perform their intended function throughout the period of extended operation.

Therefore, the analyses associated with crane design, including fatigue, are valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

ATTACHMENT 4
CLARIFICATION OF VISUAL INSPECTION OF REACTOR VESSEL INTERNALS

RAI 3.8.6-2:

Since stress corrosion cracks tend to be very tight, and the surfaces on which the cracking can occur may be rough, as-wrought or as-welded surfaces, what steps will be taken in the selection of examination technique, and what performance demonstration(s) will be used, to ensure that the features of interest (morphology and size) will be detectable with the visual examination proposed?

FPL Supplemental Response To RAI 3.8.6-2

In FPL letter L-2001-65 dated April 19, 2001, FPL provided clarification of the planned inspections of the reactor vessel internals. Based on discussions with NRC staff, FPL provides the following supplemental response to RAI 3.8.6-2.

As stated in LRA Appendix B, Section 3.1.6 (page B-23), FPL will perform ultrasonic testing of the baffle/former bolting to identify if irradiation assisted stress corrosion cracking (IASCC) is occurring which could lead to loss of mechanical closure integrity. The baffle/former bolting is selected for this examination as the leading location for IASCC because it is subject to more limiting fluences and higher stresses than other potentially susceptible parts of the reactor internals addressed under the scope of the Reactor Vessel Internals Inspection Program. If IASCC is identified by the ultrasonic examination of the baffle/former bolting, then FPL will perform an enhanced VT-1 inspection capable of detecting 0.5 mil wire against a gray background of the accessible areas of the lower core plates and fuel pins, lower support columns, core barrels, baffle/former assemblies, thermal shields, and lower support forgings.

ATTACHMENT 5
SUPPLEMENTAL RESPONSE TO RAI 3.6.2.1-2
CRACKING OF ABOVE GROUNDWATER REINFORCED CONCRETE STRUCTURES

RAI 3.6.2.1-2:

In Section 3.6.2, for reinforced concrete components in structures other than containments, which are above groundwater elevation, you provided no aging management program. Most of the licensees use their systems and structures monitoring program to monitor these components. Please explain how these components will be monitored for aging effects at Turkey Point.

FPL Supplemental Response To RAI 3.6.2.1-2:

In FPL letter L-2001-61 dated March 30, 2001, FPL clarified that the analysis of possible aging effects for reinforced concrete components in structures other than containment is summarized in LRA Section 3.6.2.3. The response further stated that the analysis concludes that there are no aging effects that could cause a loss of intended function for reinforced concrete components above groundwater. Therefore, no aging management programs are required for these components. However, based upon discussions with the NRC Staff, FPL proposed in its response to modify the ASME Section XI, Subsection IWL Inservice Inspection Program described in the LRA to manage aging of Containment reinforced concrete above groundwater. These detailed inspections would serve as an indicator of potential aging for above groundwater reinforced concrete components in structures other than the Containments.

Based on further discussions with NRC staff on October 22, 2001, FPL provides the following supplemental response to RAI 3.6.2.1-2:

In response to the NRC Staff's request that an inspection program be credited for managing degradation of above groundwater concrete structures and structural components, FPL has revised the Systems and Structures Monitoring Program described in Appendix B to the Application (Section 3.2-15, beginning at page B-83) to include periodic visual inspection of accessible reinforced concrete structures and structural components for degradation. The third sentence of attribute "Parameters Monitored or Inspected" of Subsection 3.2-15 in LRA Appendix B (pages B-84 and B-85) is revised to read as follows:

> "External surfaces of concrete are monitored through visual examination for exposed rebar, extensive rust bleeding, cracks that exhibit rust bleeding, and cracking, loss of material, and change in material properties of reinforced concrete, and cracking of block walls and building roof seals."

Also, the attribute "Detection of Aging Effects" of Subsection 3.2-15 in LRA Appendix B (page B-85) is revised to reference to ACI 201.1R, "Guide for Making a Condition Survey of Concrete in Service," and the attribute "Acceptance Criteria" is revised to reference to Chapter 4 of ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures."

To summarize, although the aging management reviews performed by FPL on above groundwater reinforced concrete did not identify any aging effects requiring management, FPL will inspect accessible surfaces of above groundwater reinforced concrete structures and structural components associated with the structures listed in Subsection 3.2-15 in LRA Appendix B (pages B-83 and B-84) for concrete degradation. The ASME Section XI, Subsection IWL Inservice Inspection Program will be utilized to perform these inspections for the Containment structures, and the Systems and Structures Monitoring Program will be utilized to perform these inspections for the other structures and structural components (including internal containment reinforced concrete structures) listed on LRA pages B-83 and B-84.

FPL has reviewed the UFSAR program summaries in LRA Appendix A Subsections 16.2.1.4 (page A-35) 16.2.15 (page A-41) for the ASME Section XI, IWL Inservice Inspection Program and Systems Structures Monitoring Program, respectively. Based on the above, a revision has been made to Subsection 16.2.1.4 and is included in the revised UFSAR Supplement in Attachment 3 to this letter. No change to Subsection 16.2.15 was determined to be required. Subsection 16.2.15 as written already addresses the aging effects noted above and meets the intent of 10 CFR 54.21(d).