The applicant did not identify any aging effects for the brass, copper, and stainless steel emergency containment filtration system components exposed to an external environment of containment air, as indicated in Table 3.3-6 of the LRA. The applicant's position was found to be acceptable because the staff agreed that there are no aging effects associated with the brass, copper, and stainless steel components exposed to containment air that could cause a component to lose its ability to perform an intended function during the period of extended operation.

The loss of material of carbon steel components in the external environment of containment air is due to general and pitting corrosion.

The loss of mechanical closure integrity due to aggressive chemical attack is an aging effect that requires management of mechanical closure carbon and low alloy steel bolting that is susceptible to potential borated water leaks.

Based on the description of the emergency containment filtration system components in the internal and external environments, and the materials used in the fabrication of the various components, the staff found that the applicant adequately identified the aging effects that are applicable for this system.

3.3.6.2.2 Aging Management Programs

To manage the aging effects for the emergency containment filter housings exposed to an internal environment of air/gas, the applicant identified the following AMP:

- periodic surveillance and preventive maintenance program

To manage the aging effects for the emergency containment filtration valves, and piping/fittings, exposed to an internal environment of treated water, the applicant identified the following AMP:

- chemistry control program

To manage the aging effects for the emergency containment filter housings exposed to an external environment of containment air, the applicant identified the following AMP:

- periodic surveillance and preventive maintenance program

To manage the aging effects for the emergency containment filter housings exposed to an external environment of borated water leaks, the applicant identified the following AMP:

- boric acid wastage surveillance program

To manage the aging effects for the emergency containment filtration bolting exposed to an external environment of borated water leaks, the applicant identified the following AMP:

- boric acid wastage surveillance program

The staff reviewed the information provided in the LRA for the AMPs used by the applicant to manage the aging of the emergency containment filtration system components, and determined that the AMPs identified above are acceptable to manage the applicable aging effects. Refer to Sections 3.1.1, 3.9.3, and 3.9.11 of this SER for the review of these AMPs.

3.3.6.3 Conclusion

The staff has reviewed the information in Sections 2.3.2.6 and 3.3 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the emergency containment filtration system will be adequately managed so that there is reasonable assurance that this system will perform its intended functions in accordance with the CLB throughout the period of extended operation.

3.3.7 Containment Post-Accident Monitoring and Control

3.3.7.1 Summary of Technical Information in the Application

The applicant describes its AMR of the containment post-accident monitoring and control system for license renewal in Section 2.3.2.7, "Containment Post-Accident Monitoring and Control," and Section 3.3 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging associated with the containment post-accident monitoring and control system will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The containment post-accident monitoring and control system includes the following subsystems: post-accident hydrogen monitoring, post-accident hydrogen control, containment pressure monitoring, post-accident sampling, and containment air particulate and gas monitoring. A description of these systems is provided below.

The post-accident hydrogen monitoring system provides indication of the hydrogen gas concentration in the containment atmosphere following a loss-of-coolant accident. The mechanical portions of the post-accident hydrogen monitoring system provide a flow path from the containment to the hydrogen monitors and then back to the containment.

The containment pressure monitoring system consists of redundant containment pressure signals that are provided to isolate the containment and initiate several reactor safeguard actions. The mechanical portions of the containment pressure monitoring system provide sensing lines from the containment to the containment pressure monitors.

The only mechanical portion of the post-accident sampling in the scope of license renewal is the sample cooler because it forms a part of the component cooling water pressure boundary. Component cooling water is described in UFSAR Section 9.3.

The post-accident hydrogen control system provides the means for achieving and maintaining containment post-accident hydrogen control. Post-accident hydrogen control is described in UFSAR Section 9.12.

The containment air particulate and gas monitoring system measures radioactivity in the containment air. The mechanical portions of containment air particulate and gas monitoring provide a flow path from the containment to the monitors and then back to the containment. The containment air particulate and gas monitoring system is described in UFSAR Section 11.2.3.

The containment post-accident monitoring and control components subject to an AMR include pumps and valves (pressure boundary only), orifices, piping, tubing and fittings. The intended functions for the containment post-accident monitoring and control components subject to an AMR include pressure boundary integrity and throttling. A complete list of the containment post-accident monitoring and control components requiring an AMR and the component intended functions are provided in Table 3.3-7 of the LRA.

3.3.7.2 Staff Evaluation

3.3.7.2.1 Effects of Aging

For the containment post-accident monitoring and control system, the applicant stated that the stainless steel hydrogen monitor pumps, filter housings, valves, piping, tubing, fittings, and orifices; carbon steel tubing and fittings; aluminum pump casings; brass piping and fittings and copper tubing and fittings are exposed to air/gas. As discussed in Table 3.3-7 of the LRA, for these items exposed to air/gas there is no aging effect. For the post-accident sampling system stainless steel cooler shells, covers and tube coils exposed only to treated water, loss of material is the applicable aging effect.

There are no aging effects for containment post-accident monitoring and control system components exposed to external environments on stainless steel, aluminum, brass or copper. For valves, piping, fittings, and tubing made of carbon steel, which are exposed to an "indoor-not air-conditioned," containment air environment, or borated water leaks, the applicable aging effect is loss of material. For carbon steel bolting that is exposed to borated water leaks, the aging effect is loss of mechanical closure integrity.

Based on the description of the containment post-accident monitoring and control system components in the internal and external environments, and the materials used in the fabrication of the various components, the staff finds that the applicant has adequately identified the aging effects that apply to this system.

3.3.7.2.2 Aging Management Programs

To manage the aging effects on the stainless steel components exposed to treated water, such as, the post-accident sampling system stainless steel cooler shells, covers and tube coils, the applicant identified the following AMP:

- chemistry control program

To manage aging effects for the carbon steel valves, piping, fittings and tubing exposed to "indoor-not air-conditioned" or containment air environment, the applicant identified the following AMP:

- systems and structures monitoring program

To manage the aging effects for the carbon steel valves, piping, fittings and tubing exposed to borated water leaks, the applicant identified the following AMPs:

- boric acid wastage surveillance program

To manage the aging effects for the carbon steel bolting exposed to borated water leaks, the applicant identified the following AMP:

- boric acid wastage surveillance program

The staff reviewed the information provided in the LRA for the AMPs used by the applicant to manage the aging effects of the containment post-accident monitoring and control system components, and determined that the AMPs identified above are acceptable to manage the applicable aging effects. Refer to Sections 3.1.3 and 3.9.3 of this SER for the review of these AMPs.

3.3.7.3 Conclusion

The staff has reviewed the information in Sections 2.3.2.7 and 3.3 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment post-accident monitoring and control system will be adequately managed so that there is reasonable assurance that this system will perform its intended functions in accordance with the CLB during the period of extended operation.

3.4 Auxiliary Systems

In LRA Section 3.4, "Auxiliary Systems," the applicant describes the AMR for the auxiliary systems. Appendices A, B, and C to the LRA also contain supplementary information related to the AMR of the auxiliary systems. The staff reviewed Section 3.4 and the applicable portions of Appendices A, B, and C to determine whether the applicant has provided sufficient information to demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB throughout the period of extended operation, in accordance with 10 CFR 54.21(a)(3) for the auxiliary system structures and components (SCs) that are determined to be within the scope of license renewal and subject to an AMR.

The Turkey Point auxiliary systems include the following 15 systems:

- intake cooling water (ICW)
- component cooling water (CCW)
- spent fuel pool cooling
- chemical and volume control
- primary water makeup
- sample systems
- waste disposal
- instrument air
- normal containment and control rod drive mechanism cooling

- auxiliary building ventilation
- control building ventilation
- emergency diesel generator building ventilation
- turbine building ventilation
- fire protection
- emergency diesel generators and support systems

In the LRA, Section 2.1, "Scoping and Screening Methodology," the applicant describes the method used to identify the SCs that are within the scope of license renewal and subject to an AMR. The applicant identifies and lists the auxiliary system SCs in Section 2.3.3 "Auxiliary Systems," of the LRA. The staff's evaluation of the scoping methodology and the auxiliary system SCs included within the scope of license renewal and subject to an AMR is documented in Sections 2.1 and 2.3.3 of this SER, respectively. In LRA Appendix A, "Updated Final Safety Analysis Report Supplement," the applicant provides a summary description of the programs and activities used to manage the effects of aging, as required in 10 CFR 54.21(d). The applicant provides a more detailed description of these AMPs for the staff to use in its evaluation in Appendix B to the LRA. In LRA Appendix C, the applicant describes the processes used to identify many of the applicable aging effects for the SCs that are subject to an AMR. In LRA Appendix D, the applicant states that no changes to the Turkey Point Technical Specifications (TS) have been identified. A discussion of each system follows.

3.4.1 Intake Cooling Water

The intake cooling water (ICW) system removes heat from the component cooling water (CCW) and the turbine plant cooling water. The ICW pumps supply salt water from the plant's intake area through two redundant piping headers to the tube side of the CCW and turbine plant cooling water heat exchangers.

3.4.1.1 Summary of Technical Information in the Application

The applicant described its AMR of the ICW system for license renewal in Section 2.3.3.1, "Intake Cooling Water," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the ICW system will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The applicant identifies the following SCs of the ICW system that are within the scope of license renewal and subject to an AMR:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events

- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are relied on during postulated fires and station blackout events

The applicant states that the components subject to an AMR include pumps and valves (pressure boundary only), strainers, orifices, piping, tubing and fittings. The intended functions of ICW components that are subject to an AMR are pressure boundary integrity, filtration, structural integrity, structural support, and throttling. A complete list of ICW components that require an AMR and their component intended functions appears in Table 3.4-1 of the LRA.

3.4.1.2 Staff Evaluation

The components in the ICW system are fabricated from carbon steel, stainless steel, cast iron, rubber, bronze, copper-nickel, monel, and aluminum-bronze exposed to an internal environment of raw water in the cooling canals. These components include ICW pumps and pump expansion joints; basket strainer (shells and internal screens); valves, piping, and fittings; orifices; and thermowells chemical injection nozzles (Units 3 only). The aging effects of these materials in the raw water environment are identified in Table 3.4-1, and are discussed in Section 6.2, "Raw Water," of Appendix C to the LRA. The raw water environment in the cooling canals is defined as salt water used as the ultimate heat sink. Applicable aging effects in this internal environment include loss of material (due to general corrosion, pitting corrosion, galvanic corrosion, crevice corrosion, microbiologically induced corrosion (MIC), and selective leaching) and cracking (due to embrittlement for rubber).

Components in the ICW system which are exposed externally to an outdoor environment are manufactured from the following materials: stainless steel, carbon steel, cast iron, rubber, bronze, copper nickel, monel, and aluminum-bronze. The outdoor environment consists of a moist, salt-ladened atmospheric air, temperature 30°F to 95°F, and exposure to weather (including precipitation and wind). The aging effects of these materials exposed externally to the outdoor environment are identified in Table 3.4-1 and are discussed in Section 7.0, "Outdoor," of Appendix C to the LRA. Applicable aging effects for these components exposed externally to the outdoor environment include loss of material due to general, pitting, galvanic, crevice and MIC; cracking due to stress corrosion and embrittlement (in the case of rubber).

A few components in the ICW system have external surfaces which may be exposed to borated water leaks. These components include the carbon steel basket strainer shells, as well as the cast and carbon steel iron valves, piping, and fittings. The aging effects associated with external exposure to borated water leaks are identified in Table 3.4-1 and are discussed in Section 7.5, "Borated Water Leaks," of Appendix C to the LRA. Applicable aging effects include loss of material and loss of mechanical closure due to aggressive chemical attack.

3.4.1.2.1 Aging Management Programs

To manage the aging effects of stainless steel, carbon steel, cast iron, bronze, copper-nickel, monel, rubber, and aluminum-bronze components exposed internally to a raw water environment in the cooling canals, the applicant relies on the following AMPs:

- periodic surveillance and preventive maintenance program
- ICW system inspection program
- systems and structures monitoring program

The periodic surveillance and preventive maintenance program provides for visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/refurbishment during the performance of periodic surveillance and preventive maintenance activities. The description of this program is provided in Appendix B, Section 3.2.11, "Periodic Surveillance and Preventive Maintenance Program," of the LRA. This program is credited for managing the aging effects of stainless steel, carbon steel and cast iron ICW pumps; rubber ICW pump expansion joints; and aluminum-bronze pump discharge valves that are internally exposed to the raw water environment. The periodic surveillance and preventive maintenance program is a current program that will be enhanced with regard to the scope of specific inspections and their documentation. The staff requested that the applicant provide applicable frequencies, bases and the most recent operating history supporting the adequacy of this program in managing the aging effects associated with the following components: stainless steel, carbon steel and cast iron ICW pumps; rubber ICW pump expansion joints; and aluminum-bronze pump discharge valves that are externally exposed to the raw water environment. In response to the staff's RAI, the applicant indicated that this program is used to manage internal and external aging effects of these components. In addition, the scheduled frequency of preventive maintenance activity for the replacement of the ICW pumps, discharge check valves, and expansion joints is 42 months. The applicant further stated that this frequency is based on the operating and maintenance history of these components at Turkey Point, and adjustments to this frequency may be made based on future plant-specific performance and/or industry experience.

On the basis of this information, the staff finds this program acceptable in managing the aging effects associated with these components in the ICW system. The staff's detailed evaluation of the periodic surveillance and preventive maintenance program is provided in Section 3.9.11 of this SER.

The ICW system inspection program was developed in response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." This program includes performance testing and evaluations, systematic inspections, leakage evaluations, and corrective actions to ensure that loss of material, cracking or biological fouling does not lead to loss of component intended function. The description of this program is provided in LRA Appendix B, Section 3.2.10, "Intake Cooling Water System Inspection Program." This program is credited for managing the aging effects of carbon steel basket strainer shell; stainless steel basket internal screen and cast iron, copper-nickel, and bronze valves, piping, and fittings exposed internally to the raw water environment. The staff finds this program adequate in managing the aging effects for these components in the ICW system. The staff's detailed evaluation of the ICW system inspection program is provided in Section 3.9.10 of this SER.

The systems and structures monitoring program provides for visual inspection and examination of accessible surfaces of specific systems, structures and components, including welds and bolting. The description of this program is provided in Appendix B, Section 3.2.15, "Systems and Structures Monitoring Program," of the LRA. This program is credited for managing the aging effects of cast iron, carbon steel, bronze, monel, and stainless steel valves, piping, tubing, and fittings; stainless steel orifices; monel chemical injection nozzles and stainless steel thermowells that are internally exposed to the raw water environment. For structures that are inaccessible for inspection through the systems and structures monitoring program, an inspection of structures with similar materials and environments may be indicative of aging effects. Several components in the ICW system credit this program for managing loss of material in the raw water environment. The staff requested the applicant provide the applicable frequencies, bases and the most recent operating history supporting the adequacy of this program for the following components in the ICW system: cast iron, carbon steel, bronze, monel, and stainless steel valves, piping, tubing, and fittings; stainless steel orifices; and stainless steel thermowells. In response to the staff's request, the applicant provided the following information: leakage inspection of the ICW orifices, thermowells and tubing/fittings was inadvertently omitted from the systems and structures monitoring program description of Appendix B, Section 3.2.15, of the LRA. In addition, the applicant responded that the leakage inspection is performed at least once per 18 months and that evaluations have been performed to show that throughwall leakage equivalent to a 1-inch diameter opening will not reduce ICW flow to the CCW heat exchangers below design requirements. The applicant provided the following reasons supporting the adequacy of this program in managing the aging effect of loss of material for these components:

- For above ground cement-lined cast iron piping, the maintenance history shows that localized failures of the cement lining have occurred. This results in small corrosion cells which will be detected through small throughwall leakage.

- For carbon steel piping/fitting and valves on the discharge channel of the CCW heat exchangers, leakage does not impact the intended function because of the heat transfer capability of this component.

- For small instrument valves and piping/tubing/fittings and thermowells and orifices made of stainless steel, monel and bronze, leakage does not affect the system function due to the size of these components. In addition, operating maintenance history has shown that leakage from these components has not been significant at Turkey Point.

On the basis of the information provided by the applicant, the staff finds that this program is appropriate and adequate for managing the aging effects associated with components in the ICW system. The staff's detailed evaluation of the systems and structures monitoring program is provided in Section 3.1.3 of this SER.

To manage the aging effects of the stainless steel, carbon steel, cast iron, rubber, bronze, copper-nickel, monel, and aluminum-bronze components externally exposed to an outdoor environment, the applicant relies on the following AMPs:

- periodic surveillance and preventive maintenance program
- systems and structures monitoring program

The periodic surveillance and preventive maintenance program provides for visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/refurbishment during the performance of periodic surveillance and preventive maintenance activities. The description of this program is provided in Appendix B, Section 3.2.11, of the LRA. This program is credited for managing the aging effects of stainless steel, carbon steel, and cast iron intake cooling pumps, as well as rubber ICW pump expansion joints externally exposed to an outdoor environment. This program is discussed above.

The systems and structures monitoring program provides for visual inspection and examination of accessible surfaces of specific systems, structures and components, including welds and bolting. The description of this program is provided in Appendix B, Section 3.2.15, of the LRA. This program is credited for managing the aging effects of carbon steel basket strainers shell; stainless steel, carbon steel, and cast iron valves, piping, and fittings; stainless steel orifices; and stainless steel thermowells externally exposed to an outdoor environment. This program is discussed above.

To manage the aging effects of the carbon steel and cast iron components externally exposed to borated water leaks, the applicant relies on the following AMP:

- boric acid wastage surveillance program

The boric acid wastage surveillance program is an enhanced program which uses systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of the pressure boundary or structural integrity of components, supports or structures. The description of this program is provided in Appendix B, Section 3.2.3, "Boric Acid Wastage Surveillance Program," of the LRA. This program is credited for managing the aging effects of carbon steel basket strainers shell; carbon steel bolting; and cast iron and carbon steel valves, piping, and fittings externally exposed to borated water leaks. The boric acid wastage surveillance program provides for visual inspection of external surfaces for evidence of corrosion, cracking, leakage, fouling or coating damage. In RAI 3.4-3, dated February 2, 2001, the staff requested the applicant to provide more detail of the location of the bolts in the CCW heat exchanger room and the applicable frequencies, bases, and the most recent operating history supporting the adequacy of this program in managing the aging effects for these components. In its March 22, 2001, response to the staff's request, the applicant provided additional information that carbon and low alloy steel mechanical closures located near borated water systems are considered susceptible to aggressive chemical attack. In the ICW system, the bolted connections for piping, fittings and equipment (including valve bonnets) located in the CCW heat exchanger rooms are potentially exposed to leakage from the borated water systems. The applicant further stated that a review of the condition report and metallurgical report databases (1992 through 2000) did not identify any instance of bolting degradation due to boric acid corrosion in this system.

On the basis of the information provided, the staff finds that this program is appropriate and acceptable for managing the aging effects associated with these components. The staff's detailed evaluation of the boric acid wastage surveillance program is provided in Section 3.9.3 of this SER.

3.4.1.3 Conclusion

The staff has reviewed the information in Section 2.3.3.1 and Section 3.4 of the LRA and the applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the ICW system will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.2 Component Cooling Water

The CCW system removes heat from safety-related and non-safety-related components during normal and emergency operation. The component cooling water pumps circulate component cooling water through heat exchangers and coolers that are associated with other systems. The component cooling water heat exchangers transfer the heat from these systems to the intake cooling water.

3.4.2.1 Summary of Technical Information in the Application

The applicant described its AMR of the component cooling water system for license renewal in Section 2.3.3.2, "Component Cooling Water," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the component cooling water system will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The applicant identifies the following SCs of the CCW system that are within the scope of license renewal and subject to an AMR:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events
- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are part of the environmental qualification program
- SCs that are relied on during postulated fires and station blackout events

The applicant states that the components subject to an AMR include pumps and valves (pressure boundary only), heat exchangers, tanks, orifices, piping, tubing and fittings. The intended functions for component cooling water components subject to an AMR include pressure boundary integrity, heat transfer, and throttling. A complete list of component cooling water components that require AMR and their component intended functions appears in Table 3.4-2 of the LRA.

3.4.2.2 Staff Evaluation

The components in the CCW system are fabricated from stainless steel, carbon steel, cast iron, copper nickel, brass, aluminum-brass, and copper internally exposed to the treated water environment. The components exposed internally to the treated water environment are: stainless steel component cooling water head tanks; carbon steel component cooling water surge tanks; cast iron component cooling water pumps; carbon steel, copper-nickel, and aluminum-brass component cooling water heat exchanger internals; carbon steel and stainless steel valves, piping, fittings, and thermowells; brass valves, carbon steel and stainless steel rotometers; and stainless steel orifices. The aging effects of these materials in the treated water environment are identified in Table 3.4-2 and are discussed in Section 6.1, "Treated Water," of Appendix C to the LRA. Applicable internal aging effects in the treated water environment are loss of material due to general, pitting, and galvanic corrosion, MIC, and selective leaching; cracking due to stress corrosion, intergranular stress-corrosion, embrittlement, and high-cycle fatigue of stainless steel materials; and fouling due to biological and particulate fouling. Although cracking due to stress corrosion, intergranular stress corrosion, embrittlement, and high-cycle fatigue are applicable aging effects for stainless steel materials that are internally exposed to the treated water environment, these aging effects are not identified for any stainless steel component in Table 3.4-2 of the LRA. In RAI 3.4.2-1, dated February 2, 2001, the staff requested the applicant provide the bases for excluding these applicable aging effects for stainless steel components in the CCW system. In its March 22, 2001, response, the applicant provided the following information:

- The highest operating temperature in the CCW system is 140 °F, the threshold temperature for stress corrosion cracking (SCC) in a treated water environment. In addition, at this temperature, components of the CCW system are not susceptible to intergranular stress-corrosion cracking (IGSCC) or embrittlement. The applicant further stated that this conclusion is supported by plant operating experience which did not identify any instances of SCC or IGSCC in stainless steel CCW components.

- High cycle fatigue (such as vibration-induced fatigue) is fast acting, and typically occurs early in a component's life. The applicant did not find any instances of fatigue-induced cracking of stainless steel components in the CCW system.

On the basis of this information, the staff finds that stainless steel components in the CCW system are not subject to stress corrosion, intergranular stress corrosion, embrittlement, and high-cycle fatigue.

The CCW system also has components internally exposed to the raw water environment in the closed cooling canals. These components are the copper-nickel component cooling water heat exchanger tube sheets (tube side), channels, and channel door overlay, as well as the aluminum-brass component cooling water heat exchanger tubes (inside diameter). The aging effects of these materials in the raw water environment are identified in Table 3.4-2 and are discussed in Section 6.2 of Appendix C of the LRA. The raw water environment in the cooling canals is defined as salt water used as the ultimate heat sink. Applicable aging effects in this internal environment include loss of material due to pitting corrosion, crevice corrosion, and MIC.

The air/gas environment is an applicable internal environment for the stainless steel component cooling water head tanks; carbon steel pressure vessels (air reservoirs); stainless steel, carbon steel, and brass valves, piping, fittings, tubing, and filters; and stainless steel orifices. The applicant did not identify any aging effects of these components in the air/gas environment. The aging effects associated with exposure to the air/gas environment are identified in Table 3.4-2 and are discussed in Section 6.3, "Air/Gas," of Appendix C to the LRA. Several air/gas environment descriptions are provided for each of the air/gas environments found in the plant. Aging effects of components exposed to the air/gas environment is dependent, in part, on the type of air/gas environment, the operating temperature, and the water content. The staff requested the applicant provide the characteristic parameters of the air/gas environments applicable to the components found in the CCW system and to provide the bases by which the determination of no aging effects requiring management was concluded for all components exposed to the air/gas environment. This RAI is similar to information requested for stainless steel components exposed to an air/gas environment in the chemical and volume control system (RAI 3.4.4-1). The staff evaluated the information and on the basis of the applicant's response, stainless steel is not susceptible to loss of material in this environment and therefore, no AMP is required.

Components in the CCW system which are exposed externally to an outdoor environment are manufactured from the following materials: stainless steel, carbon steel, cast iron, brass and copper-nickel. The outdoor environment consists of a moist, salt-ladened atmospheric air, temperature 30 °F to 95 °F, and exposure to weather, including precipitation and wind. The aging effects associated with external exposure to an outdoor environment are identified in Table 3.4-2 and are discussed in Section 7.1, "Outdoor," of Appendix C to the LRA. Applicable aging effects for the remaining components exposed externally to the outdoor environment include loss of material due to general, and pitting corrosion.

Components in the CCW system which are exposed externally to an indoor environment - not air-conditioned are carbon steel component cooling water surge tanks; carbon steel pressure vessels (air reservoirs); carbon steel valves, pipings, and fittings; stainless steel valves, piping, fittings, filters, and thermowells; stainless steel orifices and rotometers; carbon steel rotometers; and brass valves. The indoor, not air-conditioned environment, is defined as atmospheric air with a maximum air temperature of 104 °F, humidity between 5% and 95%, and no exposure to weather. The aging effects associated with external exposure to an outdoor environment are identified in Table 3.4-2 and are discussed in Section 7.2, "Indoor - Not Air Conditioned," of Appendix C to the LRA. The applicable aging effect is loss of material due to general and pitting corrosion for carbon steel.

The CCW system also contains carbon steel valves, piping, and fittings that are externally exposed to containment air. The containment air environment is described as atmospheric air with a maximum temperature of 120 °F, humidity between 5% and 95%, radiation total integrated dose rate of 1 rad/hr, and no exposure to weather. The aging effects associated with external exposure to the containment air environment are identified in Table 3.4-2 and are discussed in Section 7.4, "Containment Air," of Appendix C to the LRA. Applicable aging effects for carbon steel components include loss of material due to general and pitting corrosion.

A few components in the CCW system have external surfaces which may be exposed to borated water leaks. These components include the carbon steel pressure vessels (air reservoirs); cast iron component cooling water pumps; carbon steel component cooling water heat exchanger shells, flanges, and doors; carbon steel valves, piping, fittings, rotometers, and bolting. The aging effects associated with external exposure to borated water leaks are identified in Table 3.4-2 and are discussed in Section 7.5 of Appendix C to the LRA. Borated water leaking from systems undergoes evaporation, which results in a highly concentrated solution of boric acid or deposits of boric acid crystals. Applicable aging effects include loss of material and loss of mechanical closure due to aggressive chemical attack. The staff finds that the aging effects identified by the applicant are acceptable.

3.4.2.2.1 Aging Management Programs

To manage the aging effects of stainless steel, carbon steel, cast iron, copper-nickel, aluminum-brass, and brass that are internally exposed to a treated water environment, the applicant relies on the following AMPs:

- chemistry control program
- galvanic susceptibility inspection program (carbon steel only)

The chemistry control program provides for sampling and analysis of treated water. The description of this program is provided in Appendix B, Section 3.2.4 "Chemistry Control Program," of the LRA. The staff finds this program appropriate in managing the aging effects associated with the treated water environment. The staff's detailed evaluation of this program is found in Section 3.1.1 of this SER.

The galvanic susceptibility inspection program is a new program which will provide for one-time inspections the results of which will be used to determine the need for additional actions. The description of this program is provided in Appendix B, Section 3.1.5, "Galvanic Corrosion Susceptibility Inspection Program," of the LRA. The staff requested that the applicant provide the bases for the determination of corrosion rates and for the techniques which will be used in this new program. The applicant stated in its response to the staff's RAI that plant experience with galvanic corrosion has been limited and typically has occurred in saltwater. In addition, the applicant stated that examination techniques that have previously been employed at Turkey Point include ultrasonic, radiographic, and visual inspections. The type of examination employed will be selected based on component geometry, material of construction, and accessibility, and will utilize accepted industry practices and standards (e.g., American Society of Mechanical Engineers standards). The applicant further stated that the corrosion rate will be estimated from the original thickness, if known, or from an unaffected zone and the service time of the component. On the basis of the information provided, the staff finds that this new program is appropriate and acceptable for managing components in the chemical and volume control system. The staff's detailed evaluation of this program is provided in Section 3.8.5 of this SER.

To manage the aging effects of the carbon steel and cast iron components externally exposed to containment air, outdoor and indoor - not air-conditioned environment, the applicant relies on the following AMP:

- systems and structures monitoring program

The systems and structures monitoring program provides for visual inspection and examination of accessible surfaces of specific systems, structures and components, including welds and bolting. The description of this program is provided in Appendix B, Section 3.2.15 of the LRA. The staff's detailed evaluation of these programs are provided in Sections 3.9.11 and 3.1.3 of this SER.

To manage the aging effects of the carbon steel and cast iron components externally exposed to borated water leaks, the applicant relies on the following AMP:

- boric acid wastage surveillance program

The boric acid wastage surveillance program is an enhanced program which uses systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of the pressure boundary or structural integrity of components, supports or structures. The description of this program is provided in Appendix B, Section 3.2.3 of the LRA. The staff's detailed evaluation of this program is provided in Section 3.9.3 to this SER.

3.4.2.3 Conclusion

The staff has reviewed the information in Section 2.3.3.2 and Section 3.4 of the LRA and the applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the CCW system will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB during the period of extended operation.

3.4.3 Spent Fuel Pool Cooling

The spent fuel pool (SFP) cooling system removes decay heat from the spent fuel pool and filters and demineralizes the water in the spent fuel pool. There are two SFPs and SFP cooling systems. Spent fuel pool cooling consists of three separate cooling, purification, and skimmer loops.

3.4.3.1 Summary of Technical Information in the Application

The applicant described its AMR of the SFP cooling system for license renewal in Section 2.3.3.3, "Spent Fuel Pool Cooling," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the SFP cooling system will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The applicant identifies the following SCs of the SFP cooling that are within the scope of license renewal and subject to an AMR:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events

- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are relied on during station blackout events

The applicant states that the components that are subject to an AMR include pumps and valves (pressure boundary only), heat exchangers, filters, demineralizers, orifices, piping, tubing, and fittings. The intended functions of SFP cooling components that are subject to an AMR include pressure boundary integrity, heat transfer, and throttling. A complete list of SFP cooling components that require AMR and the component intended functions appears in Table 3.4-3 of the LRA.

3.4.3.2 Staff Evaluation

The components in the SFP cooling system are fabricated from stainless steel, worthite (nickel-based alloy), and carbon steel exposed to an internal environment of treated water. These components include SFP cooling pumps (refueling water purification pumps), emergency SFP cooling pumps, SFP cooling heat exchanger internals, valves, piping, fittings, tubing, filters, demineralizers, flow elements, and orifices. The aging effects of these materials in the treated water environment are identified in Table 3.4-3 and are discussed in Section 6.1 of Appendix C of the LRA. The treated water environments are treated water-borated and treated water for this application. Applicable internal aging effects in the treated water environment are loss of material due to general, pitting, and galvanic corrosion, MIC, and fouling due to biological and particulate fouling.

Components in the SFP cooling system which are exposed externally to an indoor environment - not air-conditioned are stainless steel SFP cooling pumps; stainless steel refueling water purification pumps; worthite (nickel-based alloy) emergency SFP cooling pumps; carbon steel SFP cooling heat exchanger shells and covers; stainless steel valves, piping, fittings, filters, demineralizers, flow elements, and orifices. The "indoor environment-not air-conditioned," is defined as atmospheric air with a maximum air temperature of 104 °F, humidity between 5% and 95%, and no exposure to weather. The aging effects associated with external exposure to an outdoor environment are identified in Table 3.4-3, and are discussed in Section 7.2 of Appendix C to the LRA. Applicable aging effects include loss of material due to general and pitting corrosion for carbon steel.

Components in the SFP cooling system that are exposed externally to an outdoor environment are the stainless steel refueling water purification pumps, valves, piping, fittings, tubing, filters and demineralizers. The outdoor environment consists of a moist, salt-ladened atmospheric air, temperature 30 °F to 95 °F, and exposure to weather, including precipitation and wind. The aging effects of these materials which are externally exposed to the outdoor environment are identified in Table 3.4-3 and discussed in Section 7.0 to Appendix C of the LRA. There are no applicable aging effects for these stainless steel components, which are externally exposed to the outdoor environment.

A few components in the SFP cooling system have external surfaces that may be exposed to borated water leaks. These components include carbon steel bolting and the carbon steel SFP cooling heat exchanger shells and covers. Borated water leaking from systems undergoes evaporation, which results in a highly concentrated solution of boric acid or deposits of boric acid crystals. The aging effects associated with external exposure to borated water leaks are identified in Table 3.4-3 and are discussed in Section 7.5 of Appendix C of the LRA. Applicable aging effects include loss of material and loss of mechanical closure due to aggressive chemical attack.

The SFP cooling system has stainless steel piping and fittings which are encased or embedded in concrete. There are no applicable aging effects requiring management for these components.

3.4.3.2.1 Aging Management Programs

To manage the aging effects of stainless steel, worthite (nickel-based alloy), and carbon steel components that are internally exposed to a treated water environment, and submerged stainless steel piping and fittings in the same environment, the applicant relies on the following AMPs:

- chemistry control program
- galvanic susceptibility inspection program (carbon steel only)

The chemistry control program provides for sampling and analysis of treated water. The description of this program is provided in Appendix B, Section 3.2.4 of the LRA. The staff finds this program appropriate in managing the aging effects associated with the treated water environment. The staff's detailed evaluation of this program is found in Section 3.1.1 of this SER.

The galvanic susceptibility inspection program is a new program which will provide for one-time inspections, the results of which will be used to determine the need for additional actions. The description of this program is provided in Appendix B, Section 3.1.5. The staff requested that the applicant provide the bases for the determination of corrosion rates and for the techniques which will be used in this new program. The applicant stated in its response to the staff's RAI that plant experience with galvanic corrosion has been limited and typically has occurred in saltwater. In addition, the applicant stated that examination techniques that have previously been employed at Turkey Point include ultrasonic, radiographic, and visual inspections. The type of examination employed will be selected based on component geometry, material of construction, and accessibility, and will utilize accepted industry practices and standards (e.g., American Society of Mechanical Engineers standards). The applicant further stated that the corrosion rate will be estimated from the original thickness, if known, or from an unaffected zone and the service time of the component.

On the basis of the information provided, the staff finds that this new program is appropriate and acceptable for managing components in the SFP cooling system. The staff's detailed evaluation of this program is provided in Section 3.8.5 of this SER.

To manage the aging effects of carbon steel exposed externally to an indoor - not air-conditioned environment, the applicant relies on the following AMP:

- systems and structures monitoring program

The systems and structures monitoring program provides for visual inspection and examination of accessible surfaces of specific systems, structures and components, including welds and bolting. The description of this program is provided in Section 3.2.15 of Appendix B to the LRA. The staff's detailed evaluation of this program is provided in Section 3.1.3 of this SER.

To manage the aging effects of the carbon steel components externally exposed to borated water leaks, the applicant relies on the following AMP:

- boric acid wastage surveillance program

The boric acid wastage surveillance program is an enhanced program which uses systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of the pressure boundary or structural integrity of components, supports or structures. The description of this program is provided in Appendix B, Section 3.2.3 of the LRA. This program is credited for managing the aging effects of carbon steel SFP cooling heat exchanger shells and covers and carbon steel bolting externally exposed to borated water leaks. The boric acid wastage surveillance program provides for visual inspection of external surfaces for evidence of corrosion, cracking, leakage, fouling or coating damage. The staff requested the applicant provide more detail of the location of the bolts in the SFP cooling water system and the applicable frequencies, bases and the most recent operating history supporting the adequacy of this program in managing the aging effects for these components. In response to the staff's request, the applicant provided the following additional information: carbon and low alloy steel mechanical closures located near borated water systems are considered susceptible to aggressive chemical attack. In the SFP cooling water system, all bolted connections for piping, fittings and equipment (including valve bonnets), regardless of location, are potentially exposed to leakage from the borated water systems. The applicant further stated that a review of the condition report and metallurgical report databases (1992 through 2000) did not identify any instance of bolting degradation due to boric acid corrosion in this system.

On the basis of the information provided, the staff finds that this program is appropriate and acceptable for managing the aging effects associated with these components. The staff's detailed evaluation of the boric acid wastage surveillance program is provided in Section 3.9.3 of this SER.

3.4.3.3 Conclusion

The staff has reviewed the information in Section 2.3.3.3 and Section 3.4 of the LRA and the applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the SFP cooling system will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.4 Chemical and Volume Control

The chemical and volume control system provides for continuous feed and bleed for the reactor coolant system to maintain proper water level and to adjust boron concentration. This system includes the boron addition and supply system, which provides makeup, transfer boric acid solution, and maintains reactor water purity.

3.4.4.1 Summary of Technical Information in the Application

The applicant described its AMR of the chemical and volume control system for license renewal in Section 2.3.3.4 "Chemical and Volume Control," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the chemical and volume control system will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The applicant identifies the following SCs of the chemical and volume control system that are within the scope of license renewal and subject to an AMR:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events

- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are part of the environmental qualification program

- SCs that are relied on during postulated fires and station blackout events

The applicant states that the components subject to an AMR include pumps and valves (pressure boundary only), tanks, heat exchangers, orifices, piping, tubing, and fittings. The intended functions for chemical and volume control components subject to an AMR are pressure boundary integrity, heat transfer, and throttling. A complete list of chemical and volume control system (CVCS) components that require AMR and the component intended functions appears in Table 3.4-4 of the LRA.

3.4.4.2 Staff Evaluation

The components exposed internally to the treated water environment are: stainless steel boric acid storage and batching tanks, volume control tanks, holdup tanks, boric acid storage tank pumps, charging pump suction stabilizers and pulsation dampeners, non-regenerative heat exchanger internals, valves, piping, fittings, thermowells charging pumps, demineralizers and filters, orifices and flow meters; copper charging pump oil cooler tubes; cast iron charging pump oil cooler bonnets and tubes; and carbon steel non-regenerative heat exchanger shells and tube supports. The aging effects of these materials in the treated water environment are identified in Table 3.4-4 and are discussed in Section 6.1 of Appendix C of the LRA. The treated water environment is defined as either borated water or water with corrosion and/or biocides added. Applicable internal aging effects in the treated water environment are loss of material due to general, pitting, and galvanic corrosion, MIC, and selective leaching; cracking due to stress corrosion, intergranular stress corrosion and high-cycle fatigue of stainless steel materials; and fouling due to biological and particulate fouling.

The outside diameter of the copper charging pump oil cooler tubes is exposed to a lubricating oil environment. The aging effects associated with this component are identified in Table 3.4-4 and are discussed in Section 6.5, "Lubricating Oil," of Appendix C to the LRA. There are no applicable aging effects for the copper tubes.

The air/gas environment is an applicable internal environment for the stainless steel boric acid storage and batching tanks; stainless steel volume control and holdup tanks; stainless steel valves, piping, tubing, and fittings; and brass valves. The applicant did not identify any aging effects of these components in the air/gas environment. The aging effects associated with exposure to the air/gas environment are identified in Table 3.4-4 and are discussed in Section 6.3 of Appendix C to the LRA. Several air/gas environment descriptions are provided for each of the air/gas environments found in the plant. Aging effects of components exposed to the air/gas environment is dependent, in part, on the type of air/gas environment, the operating temperature, and the water content. The staff requested the applicant to provide the characteristic parameters of the air/gas environments applicable to the components found in the chemical and volume control system and to provide the bases by which the determination of no aging effects requiring management was concluded for all components exposed to the air/gas environment. In response to the staff's request, the applicant provided the following information:

- The volume control tanks internal gas space surfaces and associated valves, piping/fittings, and tubing/fittings are made up of stainless steel and exposed to a non-wetted hydrogen environment with traces of nitrogen, oxygen, and helium at a temperature of 100 °F to 130 °F.

- The holdup tanks gas space surfaces are constructed from stainless steel and are exposed to a non-wetted nitrogen environment with traces of hydrogen, helium, and oxygen at a temperature of 50 °F to 130 °F.

- The boric acid storage and boric acid batching tanks gas space surfaces and associated valves and tubing/fittings are constructed from stainless steel and exposed to a non-wetted indoor not-air conditioned environment at a maximum temperature of 104 °F.

Since stainless steel is not susceptible to loss of material in any of these environments, the applicant concluded that no AMP is required. In addition, plant operating history supports this conclusion. Based on this additional information, the staff finds that there are no applicable aging effects for stainless steel components the chemical and volume control system exposed to these environments. Therefore, there is no need for an AMP for these components.

Components in the chemical and volume control system which are exposed externally to an indoor environment not air conditioned are: stainless steel boric acid storage and batching tanks, volume control and holdup tanks, boric acid storage tank pumps, charging pumps, charging pump suction stabilizers and discharge dampeners, valves, piping fittings, thermowells, tubings, fittings, filters, demineralizers, orifices, and flow meters; brass solenoid valves; cast iron charging pump oil cooler bonnets; and carbon steel non-regenerative heat exchanger shells. The indoor environment - not air-conditioned, is defined as atmospheric air with a maximum air temperature of 104 °F, humidity between 5% and 95%, and no exposure to weather. The aging effects associated with external exposure to an outdoor environment are identified in Table 3.4-4 and are discussed in Section 7.2 of Appendix C to the LRA.

The applicable aging effect for carbon steel and cast iron components exposed to a non-air conditioned indoor environment is loss of material due to general and pitting corrosion. There are no applicable aging effects for stainless steel components exposed to a non-air-conditioned indoor environment except for components that were previously heat traced. Cracking due to stress corrosion is an applicable aging effect for stainless steel components exposed to a non-air conditioned indoor environment and previously heat-traced.

The chemical and volume control system also contains stainless steel valves, piping, fittings, thermowells, tubing, and orifices, as well as brass instrument solenoid valves that are externally exposed to containment air. The containment air environment is described as atmospheric air with a maximum temperature of 120 °F, humidity between 5% and 95%, radiation total integrated dose rate of 1 rad/hr, and no exposure to weather. The aging effects associated with external exposure to the containment air environment are identified in Table 3.4-4 and are discussed in Section 7.4 of Appendix C to the LRA. There are no applicable aging effects for these components in the containment air environment.

A few components in the chemical and volume control system have external surfaces which may be exposed to borated water leaks. These components include the cast iron charging pump oil cooler bonnets, carbon steel non-regenerative heat exchanger shells, and carbon steel bolting. The aging effects associated with external exposure to borated water leaks are identified in Table 3.4-4 and are discussed in Section 7.5 of Appendix C of the LRA. Borated water leaking from systems undergoes evaporation, which results in a highly concentrated solution of boric acid or deposits of boric acid crystals. Applicable aging effects include loss of material and loss of mechanical closure due to aggressive chemical attack.

3.4.4.2.1 Aging Management Programs

To manage the aging effects of stainless steel, carbon steel, cast iron, and copper exposed internally to a treated water environment, the applicant relies on the following AMPs:

- chemistry control program
- periodic surveillance and preventive maintenance program
- galvanic susceptibility inspection program (carbon steel, cast iron, copper)

The chemistry control program provides for sampling and analysis of treated water. The description of this program is provided in Appendix B, Section 3.2.4 of the LRA. The staff's detailed evaluation of this program is provided in Section 3.1.1 of the SER.

The periodic surveillance and preventive maintenance program provides visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/refurbishment during performance of periodic surveillance and preventive maintenance activities. The description of this program is provided in Appendix B, Section 3.2.11.

The galvanic susceptibility inspection program is a new program which will provide for a one-time inspection, the results of which will be used to determine the need for additional actions. The description of this program is provided in Appendix B, Section 3.1.5 of the LRA. The staff requested that the applicant provide the bases for the determination of corrosion rates and for the techniques which will be used in this new program. The applicant stated in its response to the staff's RAI that plant experience with galvanic corrosion has been limited and typically has occurred in saltwater. In addition, the applicant stated that examination techniques that have previously been employed at Turkey Point include ultrasonic, radiographic, and visual inspections. The type of examination employed will be selected based on component geometry, material of construction, and accessibility, and will utilize accepted industry practices and standards (e.g., American Society of Mechanical Engineers standards). The applicant further stated that the corrosion rate will be estimated from the original thickness, if known, or from an unaffected zone and the service time of the component.

On the basis of the information provided, the staff finds that this new program is appropriate and acceptable for managing components in the chemical and volume control system. The staff's detailed evaluation of this program is provided in Section 3.8.5 of the SER.

To manage the aging effects of the stainless steel, cast iron, and carbon steel components externally exposed to an "indoor-not air-conditioned environment," the applicant relies on the following AMPs:

- periodic surveillance and preventive maintenance program
- systems and structures monitoring program

The periodic surveillance and preventive maintenance program provides for visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/refurbishment during the performance of periodic surveillance and preventive maintenance activities. The description of this program is provided in Appendix B, Section 3.2.1.1 of the LRA. Cracking has been identified as a potential aging effect for stainless steel components which have been previously heat-traced. The staff requested the applicant to provide the justification of crediting a sampling program of visual inspections for detecting cracking in these stainless steel components. In addition, the staff requested additional information on the most recent inspection of these stainless steel components, the baseline inspection of these components, if applicable, and the plant history of previously heat-traced components. The applicant responded to the staff's RAI by stating that all safety-related components in the chemical and volume control system which were previously heat traced are visually inspected for leakage on a periodic basis. In addition, plant operating experience has shown that leakage in a previously heat-traced component occurred due to SCC resulting from halogen contaminants. Corrective actions resulting from this experience included inspections and replacement, as needed. The applicant stated that the most recent visual leakage inspection did not reveal any throughwall leakage, and there are no other stainless steel components at Turkey Point, presently in service, where previously existing heat tracing was removed.

On the basis of this information, the staff finds that this program will adequately manage the aging effects associated with previously heat-traced components. The staff's detailed evaluation of this program is provided in Section 3.9.11 of this SER.

The systems and structures monitoring program provides for visual inspection and examination of accessible surfaces of specific systems, structures and components, including welds and bolting. The description of this program is provided in Appendix B, Section 3.2.15 of the LRA. The staff's detailed evaluation of this program is provided in Section 3.1.3 of this SER.

To manage the aging effects of the carbon steel and cast iron components externally exposed to borated water leaks, the applicant relies on the following AMP:

- boric acid wastage surveillance program

The boric acid wastage surveillance program is an enhanced program which uses systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of the pressure boundary or structural integrity of components, supports or structures. The description of this program is provided in Appendix B, Section 3.2.3 of the LRA. The staff's detailed evaluation of this program is provided in Section 3.9.3 of this SER.

3.4.4.3 Conclusion

The staff has reviewed the information in Sections 2.3.3.4 and 3.4 of the LRA and the applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the chemical and volume control system will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.5 Primary Water Makeup

The primary water makeup system provides demineralized and deaerated water for makeup to various systems throughout the plant.

3.4.5.1 Summary of Technical Information in the Application

The applicant described its AMR of the primary water makeup system for license renewal in Section 2.3.3.5, "Primary Water Makeup," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the primary water makeup system will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The applicant identifies the following SCs of the primary water makeup system that are within the scope of license renewal and subject to an AMR:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events

- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are relied on during postulated fires and station blackout events

The applicant states that the components that are subject to an AMR include valves (pressure boundary only), piping, tubing and fittings. The intended function for primary water makeup components that are subject to an AMR is pressure boundary integrity. A complete list of primary water makeup components that require AMR and the component intended functions appears in Table 3.4-5.

3.4.5.2 Staff Evaluation

The components in the primary water makeup system are fabricated from stainless steel and carbon steel exposed to an internal environment of treated water. These components are stainless steel valves, piping, and fittings. The aging effects of these materials in the treated water environment are identified in Table 3.4-5 and are discussed in Section 6.1 of Appendix C of the LRA. Applicable internal aging effects in the treated water environment are loss of material due to pitting corrosion. Components in the primary water makeup system, which are exposed externally to an indoor not air conditioned environment are stainless steel valves, piping, and fittings. The indoor environment not air conditioned, is defined as atmospheric air with a maximum air temperature of 104 °F, humidity between 5% and 95%, and no exposure to weather. The aging effects associated with external exposure to an outdoor environment are identified in Table 3.4-5 and discussed in Section 7.2 of Appendix C to the LRA. There are no applicable aging effects for these components.

Carbon steel bolts in the primary water makeup system have external surfaces which may be exposed to borated water leaks. The aging effects associated with external exposure to borated water leaks are identified in Table 3.4-5 and discussed in Section 7.5 of Appendix C to the LRA. Borated water leaking from systems undergoes evaporation, which results in a highly concentrated solution of boric acid or deposits of boric acid crystals. Applicable aging effects are loss of mechanical closure due to aggressive chemical attack.

3.4.5.2.1 Aging Management Programs

To manage the aging effects of stainless steel components exposed internally to a treated water environment, the applicant relies on the following AMP:

- chemistry control program

The chemistry control program provides for sampling and analysis of treated water. The description of this program is provided in Appendix B, Section 3.2.4 of the LRA. The staff finds this program appropriate in managing the aging effects associated with the treated water environment. The staff's detailed evaluation of this program is found in Section 3.1.1 of this SER.

To manage the aging effects of the carbon steel bolts externally exposed to borated water leaks, the applicant relies on the following AMP:

- boric acid wastage surveillance program

The boric acid wastage surveillance program is an enhanced program which uses systematic inspections, leakage evaluations, and corrective actions to ensure that boric acid corrosion does not lead to degradation of the pressure boundary or structural integrity of components, supports or structures. The description of this program is provided in Appendix B, Section 3.2.3 of the LRA. This program is credited for managing the aging effects of carbon steel externally exposed to borated water leaks. The boric acid wastage surveillance program provides for visual inspection of external surfaces for evidence of corrosion, cracking, leakage, fouling or coating damage. The staff requested the applicant to provide more detail of the location of the bolts in the primary water makeup system and the applicable frequencies, bases and the most recent operating history supporting the adequacy of this program in managing the aging effects for these components. In response to the staff's request, the applicant provided the following additional information: carbon and low alloy steel mechanical closures located near borated water systems are considered susceptible to aggressive chemical attack. In the primary water makeup system, the bolted connections for piping, fittings, and equipment (including valve bonnets) located in the auxiliary and containment buildings are potentially exposed to leakage from the borated water systems. The applicant further stated that a review of the condition report and metallurgical report databases (1992 through 2000) did not identify any instance of bolting degradation due to boric acid corrosion in this system.

On the basis of the information provided, the staff finds that this program is appropriate and acceptable for managing the aging effects associated with these components. The staff's detailed evaluation of the boric acid wastage surveillance program is provided in Section 3.9.3 of this SER.

3.4.5.3 Conclusion

The staff has reviewed the information in Section 2.3.3.5 and Section 3.4 of the LRA and the applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the primary water makeup system will be adequately managed so that there is reasonable assurance that these systems will perform its intended functions in accordance with the CLB throughout the period of extended operation.

3.4.6 Sample Systems

3.4.6.1 Summary of Technical Information in the Application

The applicant described its AMR of the sample systems in Section 2.3.3.6, "Sample Systems," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the sample systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Turkey Point Unit 3 and 4 sample systems each consist of two subsystems, namely the sample system-nuclear steam supply system and sample system-secondary. Both subsystems are designed to operate manually, on an intermittent basis. Samples can be obtained under conditions ranging from full power to cold shutdown.

The sample system-nuclear steam supply system permits remote sampling of fluids of the primary plant systems. The subsystem is used to evaluate fluid chemistry in the reactor coolant, emergency core cooling, and chemical and volume control systems.

The sample system-secondary permits remote sampling of fluids of the secondary systems. The subsystem is used to evaluate fluid chemistry in the feedwater, condensate/condenser hotwell, steam generator blowdown, main steam, and heater drain systems.

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

Sample systems components subject to an AMR include: valves and coolers (pressure boundary only), piping, tubing, and fittings. The intended functions for sample system components that are subject to an AMR include pressure boundary integrity and throttling. A complete list of sample systems components that require an AMR and the component intended functions appears in Table 3.4-6 of the LRA.

3.4.6.2 Staff Evaluation

The aging effects requiring management in the sample systems are loss of material for carbon steel and stainless steel components, and cracking for certain stainless steel components. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

The applicant supplied references pertaining to Turkey Point plant-specific as well as industry-wide experience to support its identification of applicable aging effects for all auxiliary systems identified above. The aging effects were identified based upon the description of internal and external environments and material of construction of the system components. The applicant has included all aging effects that are consistent with published literature and industry experience and, thus, the applicable aging effects for sample systems have been properly identified.

3.4.6.2.1 Aging Management Programs

The applicant also identified three AMPs for controlling the effects of aging on the sample system: chemistry control program, system and structures monitoring program, and the boric acid wastage surveillance program. The programs were developed from industry-wide data, industry-developed methodologies, NRC documents, and the applicant's own experience. The applicant concluded that these programs would manage the aging effects in such a way that the intended function of the components in the sample systems will be maintained throughout the period of extended operation, consistent with the CLB, under all design conditions. The staff's detailed evaluations of the programs are found in Sections 3.1.1, 3.1.3 and 3.9.3 of this SER.

3.4.6.3 Conclusion

The staff reviewed the information in Sections 2.3.3.6, "Sample Systems," and 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that aging effects associated with the sample systems will be adequately managed so that there is a reasonable assurance that the system will perform the intended functions in accordance with the CLB throughout the period of extended operation.

3.4.7 Waste Disposal

3.4.7.1 Summary of Technical Information in the Application

The applicant described its AMR of the waste disposal systems for license renewal in Section 2.3.3.7, "Waste Disposal," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the waste disposal systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Waste disposal collects and processes potentially radioactive reactor plant wastes prior to release or removal from the plant site. The system is common to Units 3 and 4, except for the components associated with each containment. Waste disposal consists of three subsystems, including the liquid, solid, and gaseous waste disposal systems.

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

Waste disposal components subject to an AMR include pumps, valves and heat exchangers (pressure boundary only), piping, tubing, and fittings. The intended function for waste disposal components subject to an AMR is pressure boundary integrity. A complete list of waste disposal components that require an AMR and the component intended functions appears in Table 3.4-7 of the LRA.

3.4.7.2 Staff Evaluation

The aging effects requiring management in the waste disposal systems are loss of material for carbon steel and stainless steel components and admiralty brass heat exchanger tubing, and fouling for stainless steel drain piping. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

The applicant supplied references pertaining to Turkey Point plant-specific as well as industry-wide experience to support its identification of applicable aging effects for all auxiliary systems identified above. The aging effects were identified based upon the description of internal and external environments and material of construction of the system components. The applicant has included all aging effects that are consistent with published literature and industry experience and, thus, the applicable aging effects for waste disposal systems have been properly identified.

3.4.7.2.1 Aging Management Programs

The applicant also identified five AMPs for controlling the effects of aging on the waste disposal systems: chemistry control program, system and structures monitoring program, galvanic corrosion susceptibility inspection program, periodic surveillance and preventive maintenance program and the boric acid wastage surveillance program. The programs were developed from industry-wide data, industry-developed methodologies, NRC documents, and the applicant's own experience. The applicant concluded that these programs will manage the aging effects in such a way that the intended function(s) of the components in the waste disposal systems will be maintained during the period of extended operation, consistent with the CLB, under all design conditions. The staff's detailed evaluations of the programs are found in Sections 3.1.1, 3.1.3, 3.8.5, 3.9.3, and 3.9.11 of this SER.

3.4.7.3 Conclusion

The staff reviewed the information in Sections 2.3.3.7 and 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that aging effects associated with the waste disposal system will be adequately managed so that there is a reasonable assurance that the system will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.8 Instrument Air

3.4.8.1 Summary of Technical Information in the Application

The applicant described its AMR of the instrument air for license renewal in Section 2.3.3.8, "Instrument Air," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the instrument air systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Instrument air provides a reliable source of dry, oil-free air for instrumentation and controls and pneumatic valves. Instrument air provides motive power and control air to safety-related and non-safety-related components. Instrument air contains both electric driven and diesel driven air compressors.

Safety-related air-operated valves, normally supplied by instrument air, which are required to operate following design-basis events are provided with backup sources of either air or nitrogen. These backup sources are considered safety-related and were screened with the particular valves they serve.

The flow diagrams listed in Table 2.3-5 of the LRA show the evaluation boundaries for the portions of instrument air that are within the scope of license renewal. Instrument air components that are subject to an AMR include valves (pressure boundary only), flasks/tanks, filters, strainers, heat exchangers, orifices, piping, tubing, and fittings. The intended functions for instrument air components subject to an AMR include pressure boundary integrity, heat transfer, filtration, and throttling. A complete list of instrument air components that require an AMR and the component intended functions appears in Table 3.4-8 of the LRA.

3.4.8.2 Staff Evaluation

The aging effects requiring management in the instrument air system are loss of material for carbon steel, stainless steel, and copper alloy components, as well as fouling for aluminum heat exchanger fins. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

The applicant supplied references pertaining to Turkey Point plant-specific as well as industry-wide experience to support its identification of applicable aging effects for all auxiliary systems identified above. The aging effects were identified based upon the description of internal and external environments and material of construction of the system components. The applicant has included all aging effects that are consistent with published literature and industry experience and, thus, the applicable aging effects for instrument air systems have been properly identified.

3.4.8.2.1 Aging Management Programs

The applicant also identified four AMPs for controlling the effects of aging on the instrument air system: galvanic corrosion susceptibility inspection program, periodic surveillance and preventive maintenance program, system and structures monitoring program, and the boric acid wastage surveillance program. The programs were developed from industry-wide data, industry-developed methodologies, NRC documents, and the applicant's own experience. The applicant concluded that these programs will manage the aging effects in such a way that the intended function(s) of the components in the instrument air systems will be maintained during the period of extended operation, consistent with the CLB, under all design conditions. The staff's detailed evaluations of the programs are found in Sections 3.1.3, 3.8.5, 3.9.3, and 3.9.11 of this SER.

3.4.8.3 Conclusion

The staff reviewed the information in Sections 2.3.3.8 and 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that aging effects associated with the instrument air system will be adequately managed so that there is a reasonable assurance that the system will perform its intended functions in accordance with the CLB throughout the period of extended operation.

3.4.9 Normal Containment and Control Rod Drive Mechanism Cooling

3.4.9.1 Summary of Technical Information in the Application

The applicant described its AMR of the normal containment and control rod drive mechanism cooling for license renewal in Section 2.3.3.11, "Normal Containment and Control Rod Drive Mechanism Cooling," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the normal containment and control rod drive mechanism (CRDM) cooling systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Normal containment and control rod drive mechanism cooling provides air circulation and cooling to maintain containment bulk ambient temperature below design limits and to remove heat from the CRDM.

The flow diagrams listed in Table 2.3-5 of the LRA show the evaluation boundaries for the portions of normal containment and CRDM cooling that are within the scope of license renewal.

Normal containment and CRDM cooling components subject to an AMR include heat exchangers, coolers, ductwork, tubing, and fittings. The intended functions for normal containment and CRDM cooling components that are subject to an AMR include pressure boundary integrity, heat transfer, and structural support. A complete list of normal containment and CRDM cooling components that require an AMR and the component intended functions appears in Table 3.4-9 of the LRA.

3.4.9.2 Staff Evaluation

The aging effects requiring management in the normal containment and CRDM cooling are loss of material for carbon steel components; cracking for neoprene and coated canvas flexible connectors; and loss of material and fouling for admiralty brass, stainless steel, and aluminum heat exchanger tubing and fins. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

The applicant supplied references pertaining to Turkey Point plant-specific as well as industry-wide experience to support its identification of applicable aging effects for all auxiliary systems identified above. The aging effects were identified based upon the description of internal and external environments and material of construction of the system components. The applicant has included all aging effects that are consistent with published literature and industry experience and, thus, the applicable aging effects for normal containment and CRDM cooling systems have been properly identified.

3.4.9.2.1 Aging Management Programs

The applicant also identified five AMPs for controlling the effects of aging on the normal containment and CRDM cooling systems: chemistry control program, system and structures monitoring program, galvanic corrosion susceptibility inspection program, periodic surveillance and preventive maintenance program and the boric acid wastage surveillance program. The programs were developed from industry-wide data, industry-developed methodologies, NRC documents, and the applicant's own experience. The applicant concluded that these programs would manage the aging effects in such a way that the intended function of the components in the normal containment and CRDM cooling systems will be maintained during the period of extended operation, consistent with the CLB, under all design conditions. The staff's detailed evaluations of the programs are found in Sections 3.1, 3.8 and 3.9 of this SER.

3.4.9.3 Conclusion

The staff reviewed the information in Sections 2.3.3.9, "Normal Containment and CRDM Cooling," and 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that aging effects associated with the normal containment and CRDM cooling will be adequately managed so that there is reasonable assurance that the system will perform the intended functions in accordance with the CLB during the period of extended operation.

3.4.10 Auxiliary Building Ventilation and Electrical Equipment Room Ventilation

3.4.10.1 Summary of Technical Information in the Application

The applicant described its AMR of the auxiliary building ventilation and electrical equipment ventilation systems for license renewal in Section 2.3.3.10, "Auxiliary Building Ventilation," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the auxiliary building ventilation systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Auxiliary building ventilation provides adequate heat removal to ensure proper operation of safety-related equipment in the auxiliary building. Auxiliary building ventilation includes electrical equipment room ventilation.

Auxiliary building ventilation is common to both units. The system provides clean air to the operating areas of the auxiliary building and exhausts air from the equipment rooms and open areas of the auxiliary building. Electrical equipment room ventilation is the same for Turkey Point, Units 3 and 4. Electrical equipment room ventilation provides cooling for the electrical equipment room under normal and emergency conditions. During normal operations, non-safety-related chillers maintain the desired room temperature. In the event of a failure of the non-safety-related system or a loss of offsite power, safety-related air conditioners will perform the same function. The flow diagrams listed in Table 2.3-5 of the LRA show the evaluation boundaries for the portions of auxiliary building ventilation and electrical equipment room ventilation that are within the scope of license renewal. Auxiliary building ventilation and electrical equipment room ventilation components subject to an AMR include air handlers (pressure boundary only), damper housings, supply and exhaust fan housings filters, ductwork, tubing, and fitting. The intended function for auxiliary building ventilation and electrical equipment room ventilation components subject to an AMR is pressure boundary integrity. A complete list of auxiliary building ventilation and electrical equipment room ventilation components that require AMR and the component intended functions appears in Table 3.4-10 of the LRA.

3.4.10.2 Staff Evaluation

The aging effects requiring management in the auxiliary building ventilation system are loss of material for carbon steel components and cracking for coated canvas flexible connectors. The aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity.

The applicant supplied references pertaining to Turkey Point plant-specific as well as industry-wide experience to support its identification of applicable aging effects for all auxiliary systems identified above. The aging effects were identified based upon the description of internal and external environments and material of construction of the system components. The applicant has included all aging effects that are consistent with published literature and industry experience and, thus, the applicable aging effects for auxiliary building and electrical equipment room ventilation have been properly identified.

3.4.10.2.1 Aging Management Programs

The applicant also identified two AMPs for controlling the effects of aging on the auxiliary building and electrical equipment room ventilation: system and structures monitoring program and the boric acid wastage surveillance program. The programs were developed from industry-wide data, industry-developed methodologies, NRC documents, and the applicant's own experience. The applicant concluded that these programs will manage the aging effects in such a way that the intended function(s) of the components in the auxiliary building and electrical equipment room ventilation will be maintained throughout the period of extended operation, consistent with the CLB, under all design conditions. The staff's detailed evaluations of the programs are found in Sections 3.1.3, and 3.9.3 of this SER.

3.4.10.3 Conclusion

The staff reviewed the information in Sections 2.3.3.10 and 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that aging effects associated with the auxiliary building ventilation system will be adequately managed so that there is a reasonable assurance that the system will perform its intended functions in accordance with the CLB throughout the period of extended operation.

3.4.11 Control Building Ventilation

3.4.11.1 Summary of Technical Information in the Application

The applicant described its AMR of the control building ventilation systems for license renewal in Section 2.3.3.11 and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the control building ventilation systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Control building ventilation provides a temperature controlled environment to ensure proper operation of equipment in the control building. Control building ventilation is composed of three subsystems: control room ventilation; computer/cable spreading room ventilation; and DC equipment/inverter room ventilation. These subsystems are common for Turkey Point, Units 3 and 4.

Control room ventilation circulates air from the control room and the control room offices through roughing filters to the air handling units. Conditioned air is returned and distributed throughout the control room. Control room ventilation maintains the habitability of the control room following design-basis events. Control room ventilation is described in UFSAR Section 9.9.1.

Computer/cable spreading room ventilation maintains the temperature and humidity requirements of the vital electrical equipment installed in the computer and cable spreading rooms. It also provides sufficient ventilation for intermittent occupancy by operations and maintenance personnel. Computer/cable spreading room ventilation is described in UFSAR Section 9.9.3.

DC equipment/inverter room ventilation provides cooling to the rooms that house the safety-related battery banks, battery chargers, inverters, and DC load centers. DC equipment/inverter room ventilation is described in UFSAR Section 9.9.2.

Control building ventilation components subject to an AMR include air handling units and valves (pressure boundary only), heat exchangers, ductwork, piping, tubing, and fittings. The intended functions for control building ventilation components subject to an AMR include pressure boundary integrity, throttling, and heat transfer. A complete list of control building ventilation components that require an AMR and the component intended functions appears in Table 3.4-11 of the LRA. The AMR for control building ventilation is discussed in Section 3.4 of the LRA.

The control building ventilation system contains various components (e.g., cable spreading room and computer room chilled water surge tanks, cable spreading room and computer room chilled water pumps, cable spreading room and computer room chilled water boxes, wye strainers, thermowells, valves, piping/fittings, level gauges, flow elements, air separators, valves, tubing/fittings, cable spreading room and computer room air handling unit headers, and cable spreading room and computer room air handling unit tubes) fabricated from carbon steel, stainless steel, and/or copper and exposed to treated water. The applicant evaluated the aging effects for carbon steel, stainless steel, and/or copper exposed to treated water in Sections 5 and 6 of Appendices C.5 and C.6 to the LRA and identified several forms of corrosion that may result in loss of material (e.g., general corrosion, pitting, galvanic corrosion, erosion/corrosion, and MIC).

The control building ventilation system contains various components (e.g., cable spreading room and computer room chilled water surge tanks, valves, piping/fittings, level gauges, air separators, cable spreading room and computer room air handling unit air boxes in air handlers, ductwork, and ductwork flexible connectors) fabricated from carbon steel, and/or coated canvas exposed to an air/gas environment. The applicant identified loss of material as the aging effect requiring management for carbon steel, and cracking for coated canvas. The applicant evaluated the aging effects for carbon steel exposed to an air/gas environment in Sections 4, 5, and 6 of Appendix C to the LRA.

3.4.11.2 Staff Evaluation

3.4.11.2.1 Effects of Aging

The control building ventilation system contains various components (e.g., cable spreading room and computer room chilled water surge tanks, cable spreading room and computer room chilled water pumps, cable spreading room and computer room chilled water boxes, wye strainers, thermowells, flow elements, air separators, valves, and tubing/fittings) that are fabricated from carbon steel and exposed to outdoor air environment. The applicant identified loss of material as an aging effect requiring management in an external environment and evaluated the aging effects for carbon steel exposed to an outdoor air environment in Section 7 of the application. The applicant identified several forms of corrosion that may result in loss of material (e.g., general corrosion, pitting, galvanic corrosion, crevice corrosion, and MIC) in Sections 5 and 7 of Appendix C to the LRA.

The control building ventilation system contains ductwork fabricated from galvanized steel exposed to an air/gas environment. The applicant concluded that there are no aging effects requiring management for this material in this environment. The staff agrees there are no aging effects for galvanized steel exposed to an air/gas environment.

The control room ventilation system contains ductwork flexible connectors constructed of coated canvas exposed to an air/gas environment. The applicant identified cracking as the aging effect requiring management for this material in this environment as discussed in Section 6 of Appendix C to the LRA.

The control room ventilation system contains various components (e.g., valves, piping/fittings, thermowells, flow elements, control room ventilation air handling unit housings, control room ventilation recirculation filter housing, inverter room and battery room air handling unit housing, cable room and computer room air handling unit housings, and bolting) fabricated from carbon steel, carbon steel-galvanized and stainless steel exposed to air conditioned air. The applicant did not identify any aging effects requiring management for these components in this environment. The staff agrees that there are no aging effects required for these components in an air conditioned air environment.

The control room ventilation system contains various components (e.g., cable spreading room and room air handling unit headers, cable spreading room and room air handling unit tubes, cable spreading room and room air handling unit air boxes in air handlers, cable spreading room and room air handling unit tube fins) constructed from stainless steel, copper, carbon steel, and aluminum exposed to air conditioned air wetted with condensation. The applicant identified loss of material as the aging effect requiring management for these components in this environment as discussed in Sections 5 and 7 of Appendix C to the LRA.

3.4.11.2.2 Aging Management Programs

To manage corrosion-induced aging effects for carbon steel, stainless steel, and copper exposed to a treated water environment, the applicant relies on the following AMPs:

- chemistry control program
- galvanic corrosion susceptibility 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. The staff's detailed review of this program is described in Section 3.1.1, "Chemistry Control Program," of this SER.

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.

On the basis of 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. The staff's detailed review of this program is described in Section 3.8.5 of this SER.

To manage corrosion-induced aging effects for carbon steel exposed to an air/gas environment, the applicant relies on the following AMP:

- 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. The staff's detailed review of this program is described in Section 3.1.3 of this SER.

To manage corrosion-induced aging effects for carbon steel exposed to an outdoor environment, the applicant relies on the following AMP:

- systems and structures monitoring program
- galvanic corrosion susceptibility inspection program

These programs have been previously discussed.

To manage corrosion-induced aging effects for stainless steel, copper, carbon steel, and aluminum exposed to an air/gas environment wetted with condensation, the applicant relies on the following AMP:

- 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 replacement/refurbishment of selected components is 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. The staff's detailed discussion of this program is found in Section 3.9.11 of this SER.

To manage corrosion-induced aging effects for ductwork flexible connectors exposed to an air/gas environment, the applicant relies on the systems and structures monitoring program. This program is discussed above.

3.4.11.3 Conclusion

The staff has reviewed the information in Section 2.3.3.11, "Control Building Ventilation," and Section 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the control building ventilation systems will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.12 Emergency Diesel Generator Building Ventilation

3.4.12.1 Summary of Technical Information in the Application

The applicant described its AMR of the emergency diesel generator building ventilation systems for license renewal in Section 2.3.3.12, "Emergency Diesel Generator Building Ventilation," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the emergency diesel generator building ventilation systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Emergency diesel generator building ventilation is required to provide cooling functions for the emergency diesel generators and associated equipment. Emergency diesel generator building ventilation is different for Turkey Point, Units 3 and 4. Emergency diesel generator building ventilation is necessary to ensure proper operation of the emergency diesel generators and other safety-related electrical equipment.

Unit 3 emergency diesel generator building ventilation consists of one wall-mounted exhaust fan and associated ductwork for each emergency diesel generator. The fan operates to maintain cooling in the room when its associated emergency diesel generator is running. Unit 4 emergency diesel generator building ventilation includes the following subsystems: emergency diesel generator room ventilation; diesel control room ventilation; and 3d and 4d switchgear room ventilation. Unit 4 emergency diesel generator building ventilation is described in UFSAR Section 8.2.2.1.1.3.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of emergency diesel generator building ventilation that are within the scope of license renewal. Note: there is no flow diagram for Unit 3 emergency diesel generator building ventilation, however, all components associated with this system are in the scope of license renewal. Emergency diesel generator building ventilation is in the scope of license renewal because it contains:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events
- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Emergency diesel generator building ventilation components subject to an AMR include filters (pressure boundary only), ductwork, tubing, and fittings. The intended function for emergency diesel generator building ventilation components subject to an AMR is pressure boundary integrity. A complete list of emergency diesel generator building ventilation components that require an AMR and the component intended functions is provided in Table 3.4-12 of the LRA. The AMR for emergency diesel generator building ventilation is discussed in Section 3.4 of the LRA.

3.4.12.2 Staff Evaluation

3.4.12.2.1 Effects of Aging

The emergency diesel generator building ventilation contains components (e.g., ductwork and filter housings) fabricated from galvanized carbon steel exposed to a not air conditioned, indoor environment and an air conditioned, indoor environment. The emergency diesel generator building ventilation contains components (e.g., ductwork and filter housings) fabricated from galvanized carbon steel exposed to an air/gas environment. The applicant did not identify any aging effects requiring management for these components in this environment and the staff agrees with this assessment.

The emergency diesel generator building ventilation contains bolting (mechanical closures) fabricated from carbon steel exposed to a not air conditioned, indoor environment and an air conditioned, indoor environment. The applicant did not identify any aging effects requiring management for these components in this environment and the staff agrees with this assessment.

3.4.12.2.2 Aging Management Programs

There are no AMPs for the emergency diesel generator building ventilation because there are no aging effects requiring aging management and the staff agrees with this assessment.

3.4.12.3 Conclusion

The staff has reviewed the information in Section 2.3.3.12 and Section 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that there are no aging effects associated with the emergency diesel generator building ventilation systems requiring aging management throughout the period of extended operation.

3.4.13 Turbine Building Ventilation

3.4.13.1 Summary of Technical Information in the Application

The applicant described its AMR of the turbine building ventilation systems for license renewal in Section 2.3.3.13, "Turbine Building Ventilation," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the turbine building ventilation systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Turbine building ventilation provides a temperature controlled environment to ensure proper operation of equipment in the turbine building. Turbine building ventilation consists of two subsystems, including the load center and switchgear rooms ventilation, and the steam generator feed pump ventilation.

Load center and switchgear rooms ventilation provides a temperature controlled environment for the safety-related 4160V switchgear and 480V load centers, located in the rooms, during normal and emergency conditions. Load center and switchgear rooms ventilation is described in UFSAR Section 9.16. Steam generator feed pump ventilation provides cooling to the steam generator feed pump room. The steam generator feed pump ventilation is non-safety-related, performs no safety-related functions, and is not within the scope of license renewal.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of turbine building ventilation that are within the scope of license renewal. Turbine building ventilation is within the scope of license renewal because it contains the following types of SCs:

- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires and station blackout events in the turbine building

Ventilation components that are subject to an AMR include pumps, valves, and air handling units (pressure boundary only), as well as heat exchangers, piping, tubing, and fittings. The intended functions for turbine building ventilation components that are subject to an AMR include pressure boundary integrity, throttling, and heat transfer. A complete list of turbine building ventilation components that require an AMR and the component intended functions appears in Table 3.4-13 of the LRA. The AMR for turbine building ventilation is discussed in Section 3.4 of the LRA.

3.4.13.2 Staff Evaluation

3.4.13.2.1 Effects of Aging

The turbine building ventilation system contains various components (e.g., chilled water surge tanks, chilled water air separators, chilled water pumps, chiller water boxes, valves, piping/fittings, wye strainers, flexible hoses, flow elements, air handling unit headers, and air handling unit heat exchanger tubes) fabricated from carbon steel, stainless steel, or copper and exposed to treated water. The applicant evaluated the aging effects for these materials and environments in Section 5.1 of Appendix C of the LRA and identified several forms of corrosion that may result in loss of material (e.g., general corrosion, pitting, crevice corrosion, galvanic corrosion, and selective leaching). The applicant also identified fouling as an aging effect for the air handling unit heat exchanger tubes as discussed in Section 5.3 of the LRA.

The turbine building ventilation system contains various components (e.g., valves, piping/fittings, level gauges, air handling unit housings, air handling unit air boxes) fabricated from stainless steel and carbon steel exposed to an air/gas (wetted with condensation) environment. The applicant evaluated the aging effects for these materials and environments in Section 5.1 of Appendix C of the LRA. The applicant identified several forms of corrosion that may result in loss of material (e.g., general corrosion, pitting, crevice corrosion, and galvanic corrosion).

The turbine building ventilation system contains various components (e.g., chilled water surge tanks, chilled water air separator, chilled water pumps, chiller water boxes, valves, piping, wye strainers, thermowells, flexible hoses, level gauges, flow elements, and bolting) fabricated from carbon steel and stainless steel and exposed to outside air. The applicant evaluated the aging effects for these materials and environments in Section 5.1 of Appendix C of the LRA. The applicant identified several forms of corrosion that may result in loss of material (e.g., general corrosion, pitting, crevice corrosion, and galvanic corrosion).

The turbine building ventilation system contains various components (e.g., valves, piping/fittings, test wells, flexible hoses, flow elements, air handling unit housings, and bolting) fabricated from carbon steel, galvanized carbon steel, and stainless steel exposed to air conditioned, indoor air. The applicant evaluated the aging effects for these materials and environment in Section 4.2.3 and 7.3 of Appendix C of the LRA and identified no corrosion-related aging effects because sufficient moisture is not present. The staff agrees that there will be no aging effects requiring management of these materials in this environment.

The turbine building ventilation system contains various components (e.g., air handling unit headers, air handling unit heat exchanger tubes, air handling unit air boxes, and air handling unit heat exchanger fins) fabricated from carbon steel, copper, and aluminum exposed to air conditioned inside air wetted with condensation. The applicant evaluated the aging effects for these materials and environment in Sections 4.2.3 and 7.3 of Appendix C of the LRA and identified several forms of corrosion that may result in loss of material (e.g., general corrosion, pitting, crevice corrosion, and galvanic corrosion.)

3.4.13.2.2 Aging Management Programs

To manage corrosion-induced aging effects for carbon steel, copper, and stainless steel exposed to a treated water environment, the applicant relies on the following AMPs:

- chemistry control program
- galvanic corrosion susceptibility inspection 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 result 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.

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. On the basis of 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. The staff's detailed evaluations of the programs are found in Sections 3.1.1, and 3.8.5 of this SER.

To manage corrosion-induced aging effects for carbon steel exposed to an air/gas environment, the applicant relies on the following AMP:

- 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 Part 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. The staff's detailed evaluation of this program is found in Section 3.1.3 this SER.

To manage corrosion-induced aging effects for carbon steel exposed to an outside air environment, the applicant relies on the following AMPs:

- systems and structures monitoring program
- galvanic corrosion susceptibility inspection program

These programs are discussed above.

To manage corrosion-induced aging effects for carbon steel, copper, and aluminum exposed to an air conditioned inside air wetted with condensation environment, the applicant relies on the following AMP:

- 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. The staff's detailed evaluation of this program is found in Sections 3.9.11 of this SER.

3.4.13.3 Conclusion

The staff has reviewed the information in Section 2.3.3.13 and Section 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the turbine building ventilation systems will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.14 Fire Protection

3.4.14.1 Summary of Technical Information in the Application

The applicant described its AMR of the fire protection system for license renewal in Section 2.3.3.14, "Fire Protection" and Section 3.4, "Auxiliary Systems," of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the fire protection systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

Fire protection protects plant equipment in the event of a fire, to ensure safe plant shutdown, and minimizes the risk of a radioactive release to the environment. Fire protection consists of fire water supply including sprinklers, Halon suppression, fire dampers, RCP oil collection, alternate shutdown, safe shutdown, and fire detection and protection. Individual components that constitute alternate shutdown and safe shutdown were screened with their respective systems. Fire detection and protection was screened with electrical and instrumentation and controls (see Section 2.5). Fire protection is described in UFSAR Appendix 9.6A. The majority of fire protection is common to Units 3 and 4.

The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of fire protection that are within the scope of license renewal. Fire protection is within the scope of license renewal because it contains the following types of SCs:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events
- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions
- SCs that are relied on during postulated fires

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

3.4.14.2 Staff Evaluation

The fire protection system contains various components [e.g., basket strainers (body), basket strainers (elements) orifices, valves, piping/fittings, sprinkler heads, tubing/fittings, flexible hoses, tanks, pumps and flow restriction orifices] that are fabricated from either cast iron, stainless steel, carbon steel, galvanized carbon steel, or copper alloys exposed to raw city water. The applicant evaluated the aging effects for these materials and environment in Sections 5 and 6.2 of Appendix C to the LRA. The applicant identified several forms of corrosion that may result in loss of material. Loss of material due to general corrosion is an aging effect requiring management for cast iron and carbon steel in raw water environments. Loss of material due to pitting corrosion is an aging effect requiring management for aluminum bronze, carbon steel, cast iron, monel, and stainless steel in raw water environments. Loss of material due to galvanic corrosion is an aging effect requiring management for carbon steel and cast iron in raw water environments when coupled with materials having higher electrical potential. Loss of material due to crevice corrosion and MIC are aging effects requiring management for carbon steel, cast iron, copper-nickel, and stainless steel in raw water environments. Loss of material due to selective leaching is an aging effect requiring management for aluminum bronze and gray cast iron in raw water environments.

The fire protection system contains components (e.g., valves, piping/fittings, sprinkler heads, and flame arrestors) fabricated from carbon steel, stainless steel, galvanized carbon steel, cast iron, and copper alloy components exposed to an air/gas environment. The applicant evaluated the aging effects for these materials in Sections 4.1.3 and 6.3 of Appendix C to the LRA. Loss of material due to general corrosion is an aging effect requiring management for carbon steel and cast iron in an atmospheric air/gas environment. The applicant did not identify any aging effects for stainless steel, galvanized steel and copper alloy components.

The fire protection system contains valves, piping/fittings, tubing/fittings, tanks, oil collection enclosures and flexible hoses constructed from carbon steel and stainless steel exposed to lubrication oil or air/gas with an oil film. The applicant evaluated the aging effects for these material in Section 6.5 of Appendix C to the LRA and identified no corrosion-related aging effects because of the presence of the oil film. The staff agrees that there are no aging effects requiring aging management.

The fire protection system contains various components (e.g., raw water tanks, diesel-driven fire pump fuel oil tank, electric fire pump, basket strainer, bodies, valves, piping/fittings, flexible hoses, sprinkler heads, flow restriction orifices, and flame arrestors) fabricated from carbon steel, cast iron, galvanized carbon steel, copper alloy, and stainless steel exposed to an outdoor environment. The applicant evaluated the aging effects for these materials and environment in Section 7.0 of the LRA. Loss of material due to general corrosion is an aging effect requiring management for low alloy steel, carbon steel, and cast iron in outdoor environments. Loss of material due to pitting corrosion is an aging effect requiring management for low alloy steel, carbon steel, and cast iron in outdoor environments.

The fire protection system contains a reactor coolant pump oil collection tank, valve, piping/fittings, tubing/fittings, enclosures and drip pans, and flexible hoses fabricated from carbon steel and stainless steel exposed to containment air. The applicant evaluated the aging effects in Section 7.4.3.1 of Appendix C to the LRA and identified several forms of corrosion that may result in loss of material. Loss of material due to general corrosion is an aging effect requiring management for carbon steel in containment environments when wetted. Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel in containment environments. Corrosion is an aging effect requiring management for carbon steel when wetted in containment environments.

The fire protection system contains a reactor coolant pump oil collection tank, valves, piping/fittings, and bolting fabricated from carbon steel exposed to borated water leaks. The applicant evaluated the aging effects for these components and environment in Section 7.5 of Appendix C of the LRA. The applicant identified severe chemical attack that could lead to loss of material for these components. In addition, severe chemical attack of the bolting in bolted connections could lead to loss of mechanical closure integrity.

The fire protection system contains a diesel fire pump heat exchanger shell and cover (radiator), valves, piping and fittings, expansion joints, tubing/fittings, and flexible hoses fabricated using carbon steel, cast iron, copper alloy, stainless steel, and rubber exposed to not air conditioned indoor air. The applicant evaluated the aging effects for these materials and environment in Section 7.2 of Appendix C to the LRA. Loss of material due to general corrosion is an aging effect requiring management for carbon steel and cast iron in non-air conditioned indoor environments. Loss of material due to pitting corrosion is an aging effect requiring management for carbon steel and cast iron in non-air conditioned indoor environments. The applicant identified cracking as the aging effect for rubber in this environment. The applicant did not identify any aging effects for stainless steel and copper alloys.

The fire protection system contains valves, piping, and fittings fabricated from cast iron and carbon steel and exposed to a buried environment. The applicant evaluated the aging effects for these materials and this environment in Section 7 of Appendix C of the LRA. Loss of material due to general corrosion is an aging effect requiring management for carbon steel and cast iron in buried environments. Loss of material due to crevice and pitting corrosion, MIC and selective leaching is an aging effect requiring management for carbon steel and cast iron in buried environments.

In RAI 2.3.3.14, the staff identified that neither the Halon suppression system components nor the Halon suppression system as a whole appeared to be included in Tables 3.4-14 or 3.6-12 of the LRA. The components which appeared to be missing from the table include, but are not limited to, Halon cylinders, Halon nozzles, nitrogen cylinders, Halon piping, pilot heads, pilot lines, pilot valve bodies, and auxiliaries. The staff requested that these components be included in the scope of license renewal. The staff also requested that the applicant provide a discussion if these components should be subject to an AMR and justification for those components that are not subject to an AMR.

The applicant responded that Halon Suppression is included as part of Fire Protection in Subsection 2.3.3.14 of the LRA. All Halon Suppression components, as depicted on drawing 0-FP-08, were determined to perform or support license renewal system intended functions and are within the scope of license renewal. Except for nitrogen and Halon cylinders, Halon nozzles, and flexible hoses, all components of Halon Suppression were included in an aging management review.

Nitrogen cylinders are monitored routinely and replaced based on condition replacement criteria, therefore, nitrogen cylinders are considered short-lived and do not require an aging management review.

Halon cylinders and flexible hoses are also monitored and/or inspected on a specified frequency, however, the Halon cylinders and flexible hoses are not normally replaced. Therefore, the Halon cylinders and flexible hoses are not short-lived and should have been included in an aging management review. Additionally, the Halon nozzles were inadvertently omitted from Table 3.4-14 of the LRA.

In the response to RAI 2.3.314-14, the applicant identified that the fire protection system contains additional component (e.g., Halon cylinders, flexible hoses and Halon nozzles) fabricated of carbon steel, wire reinforced rubber and aluminum, exposed to an internal air/gas environment. The applicable external environments for these components are outdoor air for the Halon cylinders and flexible hoses and indoor-air conditioned air for the Halon nozzles. The applicant evaluated the aging effects of these materials and concluded that cracking due to embrittlement is an aging effect requiring management for wire reinforced rubber in an internal atmospheric air/gas environment. The applicant did not identify any aging effects for carbon steel and aluminum components exposed to an internal air/gas environment. The applicant concluded that loss of material due to general and pitting corrosion is an aging effect requiring management for carbon steel Halon cylinders exposed externally to an outdoor air environment. Cracking due to embrittlement is an aging effect requiring management for wire reinforced rubber exposed externally to an outdoor air environment. The applicant did not identify any aging effects for aluminum Halon nozzles exposed to an external indoor-air conditioned environment.

Additionally, in the response to RAI 2.3.3.14-4, the applicant identified that Table 3.4-14 of the LRA should have included the aging management review results for valves and piping/fittings exposed to an external environment of "indoor-air conditioned." These components are fabricated of copper alloys and galvanized carbon steel. The applicant did not identify any aging effects for these components.

In RAI 2.3.3.14-6, the staff indicated that Fire Protection License Renewal Boundary Drawing 0-FP-03, showed the fire water jockey pumps in the scope of license renewal. The pump casings were not included in the list of components identified in the scope of license renewal (Table 3.4-14). The staff requested that the applicant clarify this apparent discrepancy between the drawings and the LRA.

The applicant responded that the fire water jockey pumps were screened within the scope of LR and require and AMR. These pumps were inadvertently omitted from LRA Table 3.4-14. The response to RAI 2.3.3.14-6 indicated that the jockey pumps are fabricated of cast iron and are exposed internally to raw water - city water and externally to an outdoor air environment. The applicant concluded that loss of material due to general corrosion, crevice corrosion, pitting corrosion, MIC, selective leaching and galvanic corrosion is an aging effect requiring management for cast iron exposed to an internal environment of raw water - city water. Additionally loss of material due to general and pitting corrosion in an aging effect requiring aging management for cast iron exposed to an external outdoor air environment.

3.4.14.2.1 Aging Management Programs

To manage corrosion-induced aging effects for cast iron, stainless steel, carbon steel, galvanized steel, and copper alloys exposed to raw city water, the applicant relies on the following AMPs:

- fire protection program
- galvanic corrosion susceptibility inspection 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. UFSAR 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. The staff's detailed evaluations of these programs are found in Sections 3.9.8 of this SER.

The galvanic corrosion susceptibility inspection program manages the aging effects of loss of material due to galvanic corrosion on the internal surfaces of susceptible piping and components. The program involves selected, one-time inspections of 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. On the basis of 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. The staff's detailed review of this program is described in Section 3.8.5 of this SER.

To manage corrosion-induced aging effects for cast iron and carbon steel exposed to air/gas environment, the applicant relies on the AMP for the fire protection program. The fire protection program is described above and in Section 3.9.8 of this SER.

To manage corrosion-induced aging effects for cast iron and carbon steel exposed to outdoor environment, the applicant relies on the AMP for the fire protection program. The fire protection program is described above and in Section 3.9.8 of this SER.

To manage corrosion-induced aging effects for carbon steel exposed to a containment air environment, the applicant relies on the following AMP:

- 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.

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. The staff's detailed evaluation of this program is found in Section 3.9.11 of this SER.

To manage corrosion-induced aging effects for carbon steel exposed to borated water leaks, the applicant relies on the following AMP:

- 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., SFP 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. The staff's detailed evaluation of this program is found in Section 3.9.3 of this SER.

To manage corrosion-induced aging effects for carbon steel and cast iron exposed to not-air conditioned indoor air environment, the applicant relies on the following AMP:

- fire protection program

The fire protection program is described above.

To manage corrosion-induced aging effects for carbon steel and cast iron exposed to a buried environment, the applicant relies on the following AMP:

- fire protection program

The fire protection program is described above.

3.4.14.3 Conclusion

The staff has reviewed the information in Sections 2.3.3.14 and 3.4 of the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the fire protection systems will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.15 Emergency Diesel Generators and Support Systems

3.4.15.1 Summary of Technical Information in the Application

The applicant described its AMR of the emergency diesel generators and support systems for license renewal in Section 2.3.3.15, "Emergency Diesel Generators and Support Systems," and Section 3.4 of the LRA. The staff reviewed these sections of the LRA to determine whether the applicant has demonstrated that the effects of aging on the emergency diesel generators and support systems will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

The emergency diesel generators provide AC power to the onsite electrical distribution system to ensure the capability for a safe and orderly shutdown. The following emergency diesel generators support systems are necessary to ensure proper operation of the emergency diesel generators:

- air intake and exhaust
- air start
- fuel oil
- cooling water
- lube oil

The emergency diesel generators are described in UFSAR Section 8.2.2.1.1.1 and the emergency diesel generators support systems are described in Section 9.15. The Unit 3 emergency diesel generator fuel oil storage tank is a free-standing steel tank. The Unit 4 emergency diesel generator fuel oil storage tank is a concrete structure with a steel liner that is an integral part of the Unit 4 emergency diesel generator building. The flow diagrams listed in Table 2.3-5 show the evaluation boundaries for the portions of emergency diesel generators and support systems (EDGASS) that are within the scope of license renewal. Emergency diesel generators and support systems are in the scope of license renewal because they contain the following types of SCs:

- SCs that are safety-related and are relied upon to remain functional during and following design-basis events

- SCs that are non-safety-related whose failure could prevent satisfactory accomplishment of the safety-related functions

- SCs that are relied on during postulated fires, anticipated transients without scram, and station blackout events

Emergency diesel generators and support systems components subject to an AMR include two diesel oil storage tanks, pumps and valves (pressure boundary only), tanks, heat exchangers, flame arrestors, filters, strainers, piping, tubing, and fittings. The intended functions for emergency diesel generators and support systems components subject to an AMR include pressure boundary integrity, filtration, heat transfer, throttling, and fire spread prevention. A complete list of the emergency diesel generators and support systems components that require an AMR and component intended functions appears in Table 3.4-15 of the LRA. The AMR for the emergency diesel generators and support systems are discussed in Section 3.4 of the LRA.

3.4.15.2 Staff Evaluation

The emergency diesel generators and support systems contain various components (e.g., exhaust piping/fittings, silencers, air filter assemblies, expansion joints, tubes/fittings, air start accumulators, air start motors, air start system lubricators, valves, governor bypasses, filters, flexible hose, Unit 4 diesel oil storage tank liner, emergency diesel generator fuel oil pumps, diesel oil skid tanks, sight glasses, flexible couplings, air start piping/fittings, day tanks, Unit 3 diesel oil storage tanks and flame arrestors) fabricated either from carbon steel, galvanized steel, stainless steel, copper alloy, cast iron, aluminum, rubber, or rubber braided hoses that are exposed to an air/gas environment. The applicant evaluated the aging effects for these materials and environments in Sections 5 and 6 of Appendix C to the LRA and identified several forms of corrosion that may result in loss of material or cracking.

The EDGASS contains various components (e.g., exhaust piping/fittings, bolting, Unit 3 diesel oil storage tank, Unit 3 emergency diesel generator fuel oil pumps, various valves, piping/fittings silencers, tubing/fittings, and flame arrestors) that are fabricated from carbon steel, cast iron, and stainless steel exposed to outside air. The applicant evaluated the aging effects for these materials and environments in Sections 5 and 6 of Appendix C to the LRA, and identified several forms of corrosion that may result in loss of material.

The EDGASS contains various components (e.g., exhaust piping/fittings, silencers, air filter assemblies, expansion joints, flexible couplings, tubing/fittings, bolting, air start accumulators, air start motors, air start system lubricators, governor bypasses, flexible hoses, diesel oil day tanks, diesel oil skid tanks, Unit 3 emergency diesel generator fuel oil pumps, sight glasses, filters, diesel generator cooling water expansion tanks, diesel generator cooling water pumps, diesel generator cooling water immersion heaters, radiator water boxes, radiator tubes, orifices, diesel lube oil pumps, heat exchanger shells, and heat exchanger channel heads) that are fabricated from carbon steel, galvanized steel, cast iron, stainless steel, aluminum alloy, copper, copper alloy, and rubber exposed to not-air conditioned indoor air. The applicant evaluated the aging effects for these materials and environment in Sections 5 and 6 of Appendix C to the LRA and identified loss of material for carbon steel, copper alloys, and cast iron and cracking for stainless steel and rubber as the aging effects. No aging effects were identified for galvanized steel, copper alloy or aluminum alloy components. In several cases, the applicant identified no aging effects for carbon steel and stainless steel components when exposed to not-air conditioned indoor air. In addition, in Table 3.4-15 (page 3.4-87) of the LRA, the applicant shows that stainless steel exposed to not-air conditioned indoor air has the aging effect of loss of material. These aging effects are not consistent and needed to be explained. The applicant provided an explanation for these inconsistencies in letter L-2001-50, dated March 22, 2001. Carbon steel exposed to exhaust air/gas has the potential aging effect of loss of material due to general corrosion, crevice corrosion, and pitting because exhaust gases contain moisture and other potential contaminates the staff found the explanation adequate. Stainless steel exposed to ambient air/gas has no aging effect. Stainless steel, carbon steel, galvanized carbon steel, aluminum, and copper alloys exposed to a compressed air/gas environment has no aging effect. Carbon steel exposed to an air/gas environment in an enclosed area with diesel fuel oil vapor has no aging effect since the fuel oil vapor will prevent corrosion. Stainless steel expansion joints exposed to exhaust gas/air are subject to cracking due to fatigue. The cracking is minor and is managed by periodic inspection. The carbon steel in the diesel oil storage tank is subject to temperature fluctuations that may result in condensation on the inside of the tank. However, due to the size of the tank, the oil vapor may not provide protection against corrosion. Therefore, the applicant listed loss of material as a potential aging effect for the Unit 3 diesel oil storage tank.

The EDGASS contains various components (e.g., Unit 3 diesel oil storage tank, Unit 4 diesel oil storage tank liner, Unit 3 emergency diesel generator fuel oil pumps, diesel oil day tanks, diesel oil skid tanks, valves, piping/fittings, sight glasses, flex hoses, filters, and tubing) that are fabricated from carbon steel, cast iron, copper, and stainless steel exposed to fuel oil. The applicant evaluated the aging effects for these materials and environment in Sections 5 and 6 of Appendix C to the LRA and identified MIC that may lead to loss of material.

The EDGASS contains various components (e.g., diesel generator cooling water expansion tanks, diesel generator cooling water pumps, diesel generator cooling water immersion heaters, radiator water boxes, radiator tubes, valves, piping/fittings, tubing/fittings, flexible hoses, orifices, sight glasses, and heat exchanger channel heads) that are fabricated from carbon steel, cast iron, stainless steel, red brass, and copper alloy exposed to a treated water environment. The applicant evaluated the aging effects for these materials and environment in Sections 5 and 6 of Appendix C to the LRA, and identified several forms of corrosion that could lead to loss of material (e.g., general corrosion, pitting, and MIC for carbon steel and cast iron, and pitting for copper alloy and stainless steel).

The EDGASS contains the Unit 4 diesel oil storage tank liner constructed of carbon steel exposed to an embedded/encased environment. The applicant evaluated the aging effects for carbon steel in an embedded/encased environment in Section 7.7 of Appendix C to the LRA, and did not identify any aging effects.

The EDGASS contains various components (e.g., diesel generator lube oil pumps, heat exchanger shells, heat exchanger tubing, valves, piping/fittings, flexible hoses, sight glasses, filters, tubing/fittings, and orifices) that are constructed from carbon steel, cast iron, red brass, and stainless steel exposed to lubricating oil. The applicant evaluated the aging effects for these materials exposed to lubricating oil in Section 6.5 of Appendix C of the application and did not identify any aging effects requiring management.

3.4.15.2.2 Aging Management Programs

Emergency Diesel Generators and Support Systems

To manage corrosion-induced aging effects for carbon steel, stainless steel, and rubber exposed to an air/gas environment, the applicant relies on the following AMP:

- periodic surveillance and preventive maintenance program

The periodic surveillance and preventive maintenance program is credited for managing the aging effects of loss of material, cracking, fouling, loss of seal, and embrittlement for structures, systems, and components within the scope of license renewal. This program provides for visual inspection of selected surfaces of specific components and structural components, or alternatively their replacement/ refurbishment during the performance of periodic surveillance and preventive maintenance activities. The program also includes leak inspections of limited portions of the chemical and volume control systems. The staff's detailed evaluation of this program is found in Section 3.9.11 of this SER.

To manage corrosion-induced aging effects for carbon steel exposed to outdoor air, the applicant relies on the following AMP:

- periodic surveillance and preventive maintenance program
- systems and structures monitoring program

This program is described above.

To manage corrosion-induced aging effects for carbon steel, cast iron, stainless steel, and rubber exposed to not-air conditioned indoor air, the applicant relies on the following AMPs:

- periodic surveillance and preventive maintenance program
- systems and structures monitoring program

The periodic surveillance and preventive maintenance program has been previously discussed.

The systems and structures monitoring program manages the aging effects of loss of material, cracking, fouling, loss of seal, and change in material properties for selected systems, structures, and components within the scope of license renewal. The program provides for visual inspection and examination of accessible surfaces of specific systems, structures, and components, including welds and bolting. Aging management of structural components that are inaccessible for inspection is accomplished by inspecting accessible structural components with similar materials and environments for aging effects that may be indicative of aging effects for inaccessible structural components. For example, rust bleeding on an accessible surface of a concrete structure may be indicative of corrosion of inaccessible reinforcing steel embedded in the concrete.

This program will be enhanced by restructuring it to address inspection requirements to manage certain aging effects in accordance with 10 CFR Part 54, modifying the scope of specific inspections, and improving documentation requirements. The staff's detailed evaluation of this program is found in Section 3.1.3 of this SER.

To manage corrosion-induced aging effects for carbon steel, cast iron, stainless steel, and copper and exposed to fuel oil, the applicant relies on the following AMPs:

- chemistry control program
- periodic surveillance and preventive maintenance program

The chemistry control program is credited for managing the aging effects of loss of material, cracking, and fouling buildup for the internal surfaces of primary and secondary systems and structures. The program includes sampling activities and analysis for treated water—primary, treated water—borated, treated water—secondary, treated water, and fuel oil.

The periodic surveillance and preventive maintenance program has been previously described.

To manage corrosion-induced aging effects for carbon steel, cast iron, stainless steel, and copper alloy exposed to treated water, the applicant relies on the following AMPs:

- chemistry control program
- Unit 3 - periodic surveillance and preventive maintenance program
- Unit 4 - galvanic corrosion susceptibility inspection program

The chemistry control program and the periodic surveillance and preventive maintenance program have already been described. The staff's detailed evaluations of these programs are found in Sections 3.1.1 and 3.8.5 of this SER.

The galvanic corrosion susceptibility inspection program will manage the potential effects of loss of material due to galvanic corrosion on the internal surfaces of susceptible piping and components. Carbon steel components directly coupled to stainless steel components in raw water systems at Turkey Point are the most susceptible to galvanic corrosion. However, baseline examinations in select systems will be performed and evaluated to establish if the corrosion mechanism is active. The program will involve selected one-time inspections, the results of which will be utilized to determine the need for additional actions. The staff's detailed evaluation of this program is found in Section 3.8.5 of this SER.

3.4.15.3 Conclusion

The staff has reviewed the information in Sections 2.3.3.15 and 3.4 of the LRA and applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the emergency diesel generators and support systems will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the CLB throughout the period of extended operation.

3.4.16 General

3.4.16.1 Thermal Fatigue

The applicant did not identify cracking due to thermal fatigue as an aging effect requiring management in Section 3.4.2 for the auxiliary system components. However, the applicant identified thermal fatigue for piping systems designed to the requirements of ANSI B31.1 as a time limited aging analysis (TLAA) in Section 4.3.4 of the LRA. The staff's evaluation of that TLAA is in Section 4.3 of this SER, and aging effect due to thermal fatigue, as it applies to auxiliary system components, will not be discussed further in this section of the SER.

3.4.16.2 Mechanical Closure Integrity

In Section 5.4 of Appendix C of the LRA, the applicant stated that the loss of mechanical closure integrity is an aging effect associated with bolted mechanical closures that can result from the loss of preload due to cyclic loading, gasket creep, thermal or other effects, cracking, or loss of bolting material. The applicant further stated that the effects of the mechanisms associated with loss of preload are the same as that of a degraded gasket; that is, the potential for external leakage of the internal fluid at the mechanical joint. However, the applicant stated that with the exception of the situation where a gasket/seal is utilized to provide a radiological boundary/barrier, the aging mechanisms associated with loss of preload are not considered to require management for non-Class 1 components during the period of extended operation. Furthermore, the applicant stated that it utilizes the proper bolt torquing procedures to prevent loss of preload. Furthermore, leakage of auxiliary systems mechanical joints due to loss of preload has not been a significant issue at Turkey Point. The applicant concluded that there are no aging effects associated with loss of preload resulting from settling, relaxation after cyclic loading, gasket creep, and temperature effects in the auxiliary systems during the period of extended operation. On the basis of the information provided by the applicant, the staff agrees that loss of preload is not a significant issue at Turkey Point for mechanical joints in the auxiliary systems.

Loss of bolting material can result in a loss of components pressure boundary integrity. Most carbon steel bolting is in a dry environment and coated with a lubricant; thus, general corrosion of bolting has not been a major concern in the industry. Corrosion of fasteners has only been a concern, when leakage of a joint occurs, specifically, when bolting is exposed to boric acid. Loss of mechanical closure integrity due to boric acid corrosion was considered a potential aging effect for components in proximity, to borated water systems.

Susceptibility to cracking due to SCC of bolting material is controlled by yield strength and minimizing contaminants. Therefore, no AMP was required for cracking of bolting.

3.4.16.3 Ventilation Systems Flexible Connectors

Several ventilation systems included in Section 3.4 of the LRA contain flexible connectors (rubber, neoprene, or coated canvas materials). The ductwork in the heating, ventilation, and air conditioning (HVAC) system typically includes isolators (such as flexible connectors between ducts and fans) to prevent transmission of vibration and dynamic loading to the rest of the system. Those isolators may degrade (e.g., hardening and cracking) because of relative motion between vibrating equipment, warm moist air, temperature changes, oxygen, and radiation. In Section 5.2 of Appendix A to the LRA, the applicant stated that embrittlement is an aging mechanism that could cause cracking of rubber, neoprene, or coated canvas materials. To manage that aging effect, the applicant relies on the visual inspection included in two AMPs, periodic surveillance and preventive maintenance program, and systems and structures monitoring program described in LRA Appendix B, Sections 3.2.11 and 3.2.15, respectively. Both programs do not provide a description of the inspection schedule (frequency). In a letter dated February 2, 2001, the staff requested the applicant to describe the frequency of the subject visual inspection. Also, the applicant is requested to demonstrate the adequacy of that inspection frequency and method to ensure that aging degradation will be detected before there is loss of intended functions. The applicant responded to this RAI in a letter dated March 22,

2001. The applicant stated that the ductwork flexible connectors for HVAC system within the scope of license renewal are visually inspected on a 5-year frequency, except for the flexible connectors for the normal containment coolers. The visual inspection of the flexible connectors for the normal containment coolers is included as part of an 18-month preventive maintenance task for these coolers. The applicant further stated that these inspection frequencies are appropriate, based on the environment (ambient air) that the connectors are exposed to and the operating history of these components at Turkey Point. The applicant also stated that the frequency of these inspections may be adjusted as necessary, based on future inspection results and industry experience. The staff concurs with this response. The staff's detailed evaluations of periodic surveillance and preventive maintenance program and systems and structures monitoring program are discussed separately in Sections 3.9.11 and 3.1.3 of this SER.

3.4.16.4 Scoping Issues Related to Aging Management Programs for Auxiliary Systems

The scoping requirements of 10 CFR 54.4(a)(2) include all non-safety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). In Section 2.1.1.3 of the LRA, the applicant stated that Turkey Point, Units 3 and 4, were not originally licensed for "seismic II over I." However, "seismic II over I" was considered for license renewal scoping. In a letter dated February 2, 2001, the staff requested that the applicant clarify whether the scope of the auxiliary systems discussed in Section 3.4 of the LRA includes any seismic II over I piping. In addition, the applicant requested to clarify how the AMPs for those piping systems including their supports have been addressed. Specifically, the staff asked whether the same AMPs discussed in tables included in LRA Section 3.4 also apply to those seismic II over I piping components. The applicant responded to this RAI in letters dated March 22, and May 3, 2001. The applicant stated those piping supports for non-safety-related systems with the potential of "seismic II over I" interactions with safety-related components were identified as within the scope of license renewal. Piping for these non-safety-related systems, however, was not identified as within the scope of license renewal. The applicant also stated that the Turkey Point CLB does not require the assumption of collapse and/or deformation of non-safety-related piping under seismic loading. However, non-safety-related piping system and their supports were designed, manufactured, and installed in accordance with recognized conventional practice. The applicable codes and standards were established based on conservative criteria resulting in design stresses well within the yield strength of the materials during maximum postulated loading conditions. In addition, non-safety-related piping systems are maintained in good, essentially leak tight, operating condition, especially in the areas where safety-related components are located. System engineer walkdowns and operator rounds are performed in accordance with plant administrative, engineering, operation, and Maintenance Rule procedures. Current procedures require system engineers to perform walkdowns at least quarterly (and in some cases monthly) of their assigned systems. Operator rounds are performed at least daily, and are specifically designed to route operators through most areas of the plant to observe system operating conditions. Although not anticipated, significant degradation of non-safety-related piping would be promptly identified and resolved through FPL's 10 CFR Part 50, Appendix B corrective action program. Furthermore, the applicant stated that safety-related systems are protected from water spray, jet impingement, and pipe whip effects (due to postulated failures of non-safety-related piping) by the use of pipe whip restraints and internal barriers, as described in Turkey Point UFSAR Section 5.4.

The staff reviewed the information described above and disagreed 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 of 10 CFR 54.4(a)(2). The staff considers the seismic II over I piping segments to be within the scope of license renewal. 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. Furthermore, 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 segments are within the scope of license renewal. This issue is also discussed in Section 2.1 of this SER and is identified as open item 2.1.2-1.

By letter dated November 1, 2001, the applicant provided additional information to supplement the March 22, 2001, and May 2, 2001, responses. The applicant reiterated those SSCs, including mitigative design features, included within the scope of license renewal as a result of their initial evaluation. The applicant also addressed staff's concerns regarding the potential for age-related degradation of non-safety-related piping segments which could affect safety-related SSCs by performing a supplemental review to establish what additional non-safety-related piping should be included in the scope of license renewal. As a result of this supplemental review, the applicant brought additional non-safety-related piping segments into the scope of license renewal. On the basis of the additional information provided by the applicant, the staff concludes that all SSCs that meet the 54.4(a)(2) Scoping criteria, have been included within the scope of license renewal. The staff's evaluation of the applicant's Scoping criteria and results is discussed in Section 2.1.2.1 of this SER.

In the letter dated November 1, 2001, the applicant also provided information regarding the management of aging effects associated with those additional non-safety-related piping segments that are brought into the scope of license renewal. These contain carbon seel piping, fittings and valves in auxiliary steam, in the condensate downstream of the #4 feedwater heaters to the main feedwater pump suction line, and the #6 to #5 feedwater heater drains. The inside of these carbon steel components are exposed to treated, secondary side water and the effects of aging is loss of material. The applicant is using the chemistry control program and the flow-accelerated corrosion program to manage the effects of aging. The chemistry control program is reviewed in Section 3.2.4 and the flow-accelerated corrosion program is reviewed in Section 3.2.9 of this SER. The staff agrees that these programs are the applicable programs for managing loss of material since both of these programs follow EPRI Guidelines that have been endorsed by the staff. The applicant did not identify any effects of aging for the outside surface of these components, and the staff agrees with this conclusion.

On the basis of the additional information provided by the applicant in response to Open Item 2.1.2-1, the staff concludes that the aging management of seismic II/I piping systems is adequate and provides reasonable assurance that safety-related structures, systems, and components will be adequately protected from the consequences of a failure in the seismic II/I piping systems. Therefore, the Open Item 2.1.2-1 is closed.

3.5 Steam and Power Conversion Systems

The applicant has described its AMR of the steam and power conversion systems (SPCSs) for license renewal in Sections 2.3.4, "Steam and Power Conversion Systems," and 3.5, "Steam and Power Conversion Systems," of its LRA. The staff has reviewed these sections of the application to determine whether the applicant has provided adequate information to meet the requirements of 10 CFR 54.21(a)(3) for managing the aging effects of the SPCSs for license renewal.

3.5.1 Summary of Technical Information in the Application

The LRA has identified three systems that will require aging management to meet the requirements of 10 CFR 54.21(a)(3) for management of aging effects. The three systems are main steam and turbine generators, feedwater and blowdown, and auxiliary feedwater and condensate storage. A brief description of the systems is provided in the LRA and is given below.

3.5.1.1 Main Steam and Turbine Generators

Main steam transports saturated steam from the steam generators to the main turbine and other secondary steam system components. Main steam provides the principal heat sink for the reactor coolant system protecting the reactor coolant system and the steam generators from overpressurization, provides isolation of the steam generators during a postulated steam line break, and provides steam supply to the auxiliary feedwater pump turbines.

Turbine generators convert the steam input from main steam to the plants' electrical output, provide first-stage pressure input to the reactor protection system, and provide isolation under certain postulated steam line break scenarios. Main steam and turbine generators are described in UFSAR Section 10.2.2.

The flow diagrams listed in the LRA, Table 2.3-6 show the evaluation boundaries for the mechanical portions of main steam and turbine generators that are within the scope of license renewal. As described in the LRA, the initial scoping was performed on the basis of functions.

Main steam and turbine generators components that are subject to an AMR include valves (pressure boundary only), steam traps, flow elements, piping, tubing, and fittings. The intended functions for main steam and turbine generators components that are subject to an AMR are pressure boundary integrity and throttling. A complete list of main steam and turbine generators components that require an AMR and the component intended functions appears in Table 3.5-1. The AMR for main steam and turbine generators is discussed in Section 3.5 of LRA.

3.5.1.2 Feedwater and Blowdown

Feedwater and blowdown provide sufficient water flow to the steam generators to maintain an adequate heat sink for the reactor coolant system, provide for feedwater and blowdown isolation following a postulated loss-of-coolant accident or steam line break event, and assist in maintaining steam generator water chemistry. Feedwater and blowdown consists of three subsystems, including main feedwater, steam generator blowdown, and standby steam generator feedwater. Main feedwater supplies preheated, high-pressure feedwater to the steam generators at a rate equal to main steam and the steam generator blowdown flows. The feedwater flow rate is controlled by the steam generator level control system, which determines the desired feedwater flow by comparing the feed flow, steam flow, and the steam generator level. Main feedwater system is described in UFSAR Section 10.2.2.

Steam generator blowdown assists in maintaining required steam generator chemistry by providing a means for removal of foreign matter that concentrates in the evaporator section of the steam generator. Steam generator blowdown is fed by three independent blowdown lines (one per steam generator), which tie to a common blowdown flask. Steam generator blowdown is continuously monitored for radioactivity during plant operation. Steam generator blowdown is described in UFSAR Section 10.2.4.3.

Standby steam generator feedwater is common to Turkey Point, Units 3 and 4. Standby steam generator feedwater supplies steam generator feedwater during normal startup, shutdown, and hot standby conditions. Standby steam generator feedwater delivers sufficient feedwater to maintain one unit at hot standby, while providing makeup for maximum blowdown. The standby steam generator feedwater pumps take suction from the demineralized water storage tank and discharge to a common header upstream of the feedwater regulating valves. Standby steam generator feedwater is described in UFSAR Section 9.11. The flow diagrams listed in Table 2.3-6 show the evaluation boundaries for the portions of feedwater and blowdown that are within the scope of license renewal.

Feedwater and blowdown components that are subject to an AMR include the demineralized water storage tank, pumps and valves (pressure boundary only), orifices, piping, tubing, and fittings. The intended functions for the feedwater and blowdown system components that are subject to an AMR are pressure boundary integrity and throttling. A complete list of feedwater and blowdown components that require AMR and the component intended functions, is provided in Table 3.5-2. The aging management review for feedwater and blowdown is discussed in Section 3.5 of the LRA.

3.5.1.3 Auxiliary Feedwater and Condensate Storage

The auxiliary feedwater system supplies feedwater to the steam generators when normal feedwater sources are not available, provides for auxiliary feedwater steam and feedwater isolation during a postulated steam generator tube rupture event, and provides for auxiliary feedwater isolation to the faulted steam generator and limits feedwater flow to the steam generators to limit positive reactivity insertion during a postulated steam line break event. The auxiliary feedwater system is a shared system between Turkey Point, Units 3 and 4.

The auxiliary feedwater system contains three steam turbine-driven pumps. The pumps can be supplied steam from the steam generators in either unit. The pumps take suction from either condensate storage tank and discharge to one of two redundant headers. Each header can supply each steam generator. The auxiliary feedwater system is normally maintained in standby with one pump aligned to one discharge header and two pumps aligned to the other header. Upon initiation, all three pumps start to supply the affected steam generator with feedwater. The auxiliary feedwater system is described in UFSAR Section 9.11.

The condensate storage tank stores water for use by the auxiliary feedwater system to support safe shutdown of the plant. The condensate storage tank on each unit supplies water using three auxiliary feedwater pumps. The tank outlet piping is cross-connected between the units so that either tank can supply auxiliary feedwater to the steam generators. The condensate storage system is described in UFSAR Section 9.11.3.

The flow diagrams listed in Table 2.3-6 show the evaluation boundaries for the portions of the auxiliary feedwater system and the condensate storage system that are within the scope of license renewal.

3.5.1.4 Aging Management Programs

The LRA has identified eight aging management programs that will manage the aging effects associated with the steam and power conversion systems. These programs are auxiliary feedwater pump oil coolers inspection, auxiliary feedwater steam piping inspection program, boric acid wastage surveillance program, chemistry control program, field-erected tanks internal inspection, flow-accelerated corrosion program, galvanic corrosion susceptibility inspection program, and systems and structures monitoring program. A detailed description concerning each of the above listed programs is included in Appendix B to the LRA.

The steam and power conversion systems (SPCSs) are exposed to internal environments of treated water, lubricating oil, and air/gas, as well as external environments of outdoor, containment air, underground, and potential borated water leaks. The only parts of the SPCS components that are considered to be inaccessible for inspection are those that are buried underground. On February 8, 2001, in response to a staff RAI dated January 10, 2001, the applicant indicated that sections of the standby steam generator feedpumps suction and recirculation piping are buried underground as shown on drawing 0-FW-01. The underground sections of this piping are made of stainless steel and externally coated and wrapped in plastic to protect the coating against backfill damage. Although the pipe is buried, it is above the ground water level and therefore not exposed to ground water chemicals. Additionally, the area where the pipe is buried is paved or covered by a concrete slab, making it unlikely that the surface of the pipe will be exposed to a water environment. As part of the AMR process, the applicant reviewed the plant's operating experience, and confirmed that there has been no external corrosion of buried stainless steel piping at Turkey Point. The applicant concluded that this piping is adequately protected against potential external aging mechanisms, and that there are no external aging effects requiring management. The staff concurs with the applicant's conclusion that the AMR program is adequate to protect these buried piping sections at Turkey Point against potential aging effects.

3.5.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff has reviewed the information included in Sections 2.3.4 and 3.5 of the LRA. The purpose of the review was to ascertain whether the applicant has adequately demonstrated that the effects of aging will be adequately managed so that the intended function of the systems will be maintained consistent with the CLB throughout the period of extended operation.

3.5.2.1 Effects of Aging

The components of the SPCSs are constructed from carbon steel low alloy steel, cast iron, brass, and stainless steel. They are exposed to an external environment of outdoor, containment air, buried, and potential borated water leaks. Internally, the components in the SPCSs are exposed to environments of treated water, steam, lubricating oil, and air/gas. Section 5 of Appendix C to the LRA provides a discussion of the potential aging effects based on materials and environments. Aging effects are considered to require management if the effects could cause a component to lose its ability to perform an intended function during the period of extended operation.

The following aging effects were identified in the systems carrying treated water and steam: loss of material, cracking and loss of mechanical enclosure integrity. Tables 3.5-1, 3.5-2, and 3.5-3 of the LRA list the components, component function, material, environment, applicable aging effects and applicable aging management programs. In Tables 3.5-1 and 3.5-2 for carbon steel bolting, the effect of humidity in the external environment is not considered to cause aging that leads to loss of material due to general corrosion and loss of preload. The applicant relies on the boric acid wastage surveillance program to manage the aging effects of mechanical bolting in piping connections and closures to ensure that boric acid corrosion does not lead to degradation of the pressure boundary. When external leakage involves borated water, the aging effect of concern is loss of carbon or low alloy steel bolting material due to aggressive chemical attack (i.e., boric acid corrosion).

Therefore, the LRA addresses loss of mechanical closure integrity resulting from the external environment of "borated water leaks" and credits the boric acid wastage surveillance program for management of this effect on carbon and low alloy steel bolting. This is acceptable to the staff.

The applicant has provided references to Turkey Point plant-specific as well as industry-wide experience to support its identification of applicable aging effects for steam and power conversion systems. The staff concludes that, on the basis of the description of the internal and external environments and material of fabrication for these systems, the applicant has included aging effects that are consistent with published literature and industry experience and, thus, are acceptable to the staff.

3.5.2.2 Aging Management Programs

The applicant has identified the following eight aging management programs for controlling the effects of aging in the SPCSs:

- auxiliary feedwater pump oil coolers inspection
- auxiliary feedwater steam piping inspection program
- boric acid wastage surveillance program
- chemistry control program
- field-erected tanks internal inspection
- flow-accelerated corrosion program
- galvanic corrosion susceptibility inspection program
- systems and structures monitoring program

The programs were developed from industry-wide data, industry-developed methodologies, NRC documents, and the applicant's own experience. The applicant concluded that these programs would manage the aging effects in such a way that the intended function of the components in the SPCSs will be maintained during the period of extended operation, consistent with the current licensing basis (CLB), under all design conditions.

The staff has evaluated the FPL aging management programs in order to determine if they contain the essential elements needed to provide adequate aging management of the components in the SPCSs so that the components will perform their intended functions in accordance with the CLB during the period of extended operation. In Appendix B to the LRA, the applicant discusses the attributes that each aging management program is required to address. Those attributes are (1) scope of program including the specific structure, component or commodity for the identified aging effect, (2) preventive actions to mitigate or prevent aging degradation, (3) parameters monitored or inspected which are linked to the degradation of the particular intended function, (4) method of detection of the aging effects, (5) monitoring and trending for timely corrective actions, (6) acceptance criteria, (7) corrective actions including root cause determination and prevention of recurrence, (8) confirmation process, (9) administrative controls, and, (10) operating experience including past corrective actions resulting in program enhancements or additional programs. The application indicated that the corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled corrective actions program pursuant to 10 CFR Part 50, Appendix B, and covers all structures and components that are subject to an aging management review. The staff's evaluation of the applicant's corrective actions program is provided separately in Section 3.1.2 of this SER. The QA program satisfies the elements of corrective actions, confirmation process, and administrative controls.

On the basis of the information provided, the staff finds that the following 8 AMPs are appropriate and acceptable for managing the aging effects associated with these components:

- auxiliary feedwater pump oil coolers inspection
- auxiliary feedwater steam piping inspection program

- boric acid wastage surveillance program
- chemistry control program

- field-erected tanks internal inspection
- flow-accelerated corrosion program
- galvanic corrosion susceptibility inspection program
- systems and structures monitoring program

The eight AMPs are discussed in Sections 3.1.1, 3.1.3, 3.8.1, 3.8.2, 3.8.4, 3.8.5, 3.9.3, and 3.9.9 of this SER.

3.5.3 Conclusion

The staff has reviewed the information in LRA Sections 2.3.4, "Steam and Power Conversion Systems," and 3.5, "Steam and Power Conversion Systems," as well as applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that aging effects associated with the subject systems will be adequately managed so that there is a reasonable assurance that the subject systems will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6 Structures and Structural Components

3.6.1 Containments

3.6.1.1 Containment Structure Concrete Components

3.6.1.1.1 Summary of Technical Information in the Application

Containment structure concrete components are described in Section 3.6.1.1 of the LRA. The containment structure provides radiation shielding, protects the reactor vessel and other safety-related systems, equipment, and components against missiles and environmental conditions, and serves as the last engineered barrier to the release of radioactivity. The containment structure concrete components identified by the licensee are the dome, cylinder wall, floor, and foundation mat. These components are made of concrete and reinforced by steel bars. The dome and cylinder wall were further reinforced with a post-tensioning steel system. The containment structure concrete components were designed and constructed in accordance with the American Concrete Institute (ACI) and the American Society for Testing and Materials (ASTM) standards.

Containment structure concrete components are exposed to several different environments depending on their location. Below-grade containment structure concrete components can be either above or below the groundwater elevation. Containment structure concrete components that are below grade and above the groundwater elevation are exposed to a soil/fill environment. Containment structure concrete components that are below the groundwater elevation are exposed to a soil/fill and groundwater environment. Above-grade external surfaces of the containment structure are exposed to both indoor (without air conditioned) and outdoor environments. Internal components of the containment structure are exposed to the containment air environment.

The aging effects identified by the applicant that could cause loss of intended function(s) for containment structure concrete components are loss of material, cracking, and change in material properties. The aging management program used by the applicant to manage these aging effects is the systems and structures monitoring program.

3.6.1.1.2 Staff Evaluation

In addition to Section 3.6.1.1 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results — Structures" and the applicable aging management program descriptions provided in Appendix B to the LRA to determine whether the aging effects for the containment structure concrete components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.1.1.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for containment structure concrete components are loss of material, cracking, and change in material properties. Loss of material is manifested in containment structure concrete components as scaling, spalling, pitting and erosion. Cracking is manifested in containment structure concrete components as complete or incomplete separation of the concrete into two or more parts. Change in material properties is manifested in concrete as increased permeability, increased porosity, reduction in pH value, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength.

For the loss of material aging effect, the applicant identified the following plausible aging mechanisms: (1) freeze-thaw, (2) abrasion and cavitation, (3) elevated temperature, (4) aggressive chemical attack, and (5) corrosion of reinforcing and embedded steel. Of these aging mechanisms, the applicant stated that only aggressive chemical attack and corrosion of reinforcing and embedded steel are applicable for containment structure concrete components exposed to groundwater. As such, the applicant committed to manage loss of material only for reinforced containment concrete walls below groundwater elevation.

For the cracking aging effect, the applicant identified the following plausible aging mechanisms: (1) freeze-thaw, (2) reaction with aggregates, (3) shrinkage, (4) settlement, (5) fatigue, and (6) elevated temperature. The applicant stated that none of these aging mechanisms are applicable for containment structure concrete components at Turkey Point that are located either above or below groundwater elevation, and therefore, listed no aging management program to manage cracking in Table 3.6-2 of the LRA.

For the change in material properties aging effect, the applicant identified the following plausible aging mechanisms: (1) leaching, (2) creep, (3) elevated temperature, (4) irradiation embrittlement, and (5) aggressive chemical attack. Of these aging mechanisms, the applicant stated that only aggressive chemical attack is applicable for containment structure concrete components exposed to groundwater. As such, the applicant committed to manage change in material properties only for reinforced containment concrete walls below groundwater elevation.

The staff considers each of the above aging effects (loss of material, cracking, and change in material properties) to be both plausible and applicable for containment structure concrete components located above groundwater elevation. As such, in RAI 3.9.1.4-1 the staff requested that the applicant identify the aging management program that will be used to manage the aging effects for containment structure concrete components that are located above groundwater elevation. In its response, the applicant argued that there are no aging effects that could cause a loss of intended function for containment concrete above groundwater. At the same time, the applicant recognized the existence of the concrete degradations depicted in Appendix A to NUREG-1522. The applicant proposed to modify its ASME Section XI, Subsection IWL aging management program to include aging management of containment reinforced concrete above groundwater elevation. FPL also committed to use ACI 201.1R, "Guide for Making a Condition Survey of Concrete in Service," to establish degradation type and IWL-3211 for the acceptance criteria. Once incorporated, as committed in this response, the staff considers this issue to be resolved.

3.6.1.1.2.2 Aging Management Program

The aging management program used by the applicant to manage loss of material and cracking for containment structure concrete components located below groundwater elevation is the systems and structures monitoring program. The structural monitoring program provides condition monitoring and appraisal of containment structure concrete components through periodic visual inspections. In its description of the systems and structures monitoring program, the applicant stated that external surfaces of concrete are monitored through visual examination for exposed rebar, extensive rust bleeding, cracks that exhibit rust bleeding, and cracking of block walls and building roof seals. The results of the visual inspection for systems, structures and components are documented and the frequency for the inspection may be adjusted, as necessary, based on the inspection results and industry experience. In RAI 3.6.1.1-1, the staff requested that the applicant specifically identify how the systems and structures monitoring program manages the two aging effects, loss of material and change in material properties caused by aggressive chemical attack for containment structure concrete components that are exposed to groundwater. In its response the applicant stated that, for the containment building concrete below groundwater, which is inaccessible, visual inspection of the tendon access gallery concrete below groundwater will be required to provide early indication of potential aging effects for the containment concrete and the visual inspection will look for signs of degradation (e.g., concrete cracking, spalling, scaling, leaching, discoloration, groundwater leakage, and rust stains). The staff finds the applicant's response acceptable because the walls of the tendon access gallery are much thinner than the containment base mat or the reactor pit and, thus, any aging effects on concrete will show up sooner on the tendon access gallery than containment base mat or the reactor pit. The systems and structures monitoring program is discussed in greater detail in Section 3.1.3 of this SER.

Since the applicant did not identify any plausible and applicable aging effects for containment structure concrete components located above groundwater elevation, there are no aging management programs listed in Table 3.6-2 of the LRA to manage loss of material, cracking, and change in material properties. However, in response to the staff's position that each of these aging effects are both plausible and applicable for containment structure concrete components located above groundwater elevation, the applicant committed in its response RAI 3.9.1.4-1 to modify its ASME Section XI, Subsection IWL aging management program to include aging management of containment reinforced concrete above groundwater elevation. The ASME Section XI, Subsection IWL inservice inspection program is discussed in greater detail in Section 3.9.1.4 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the aging effects for containment structure concrete components will be adequately managed by the systems and structures monitoring program for the period of extended operation.

3.6.1.1.3 Conclusion

The staff has reviewed the information in Sections 3.6.1.1, "Containment Structure Concrete Components," and Section 2.4, "Scoping and Screening Results – Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment structure concrete components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.1.2 Containment Structure Steel Components

3.6.1.2.1 Summary of Technical Information in the Application

Containment structure steel components are described in Section 3.6.1.2 of the LRA. The purpose of the containment structure steel components is to provide several safety-related functions including serving as a pressure boundary or a fission-product retention barrier, providing structural and/or functional support to safety and non-safety-related equipment, and serving as missile and flood protection barriers. The following containment structure steel components are identified by the applicant:

- liners (including the liner plate, anchors/embedments/attachments, leak chase channels, and moisture barriers)

- penetrations [including mechanical piping, mechanical ventilation, and steel portions (pressure boundary) of the electrical penetration assemblies]

- airlocks and hatches (personnel hatch, equipment hatch, escape hatch, including seals and gaskets)

- fuel transfer tube blind flanges

The containment structure steel components were designed and constructed in accordance with ASME Section III - 1965 for the pressure boundary, and the American Institute of Steel Construction (AISC) "Manual of Steel Construction" for structural steel. The gaskets, seals, and moisture barriers that protect the containment structure steel components are elastomers.

Containment structure steel components are exposed to containment air, both indoor (not air conditioned) and outdoor air, and embedded/encased environments. Borated water is also a potential environment for containment structure steel components.

The aging effects identified by the applicant that could cause loss of intended function(s) for containment structure steel components are loss of material, cracking, and change in material properties. The aging management programs used by the applicant to manage these aging effects are the ASME Section XI, Subsection IWE inservice inspection program and the boric acid wastage surveillance program.

3.6.1.2.2 Staff Evaluation

In addition to Section 3.6.1.2 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results — Structures," and the applicable aging management program descriptions provided in Appendix B to the LRA to determine whether the aging effects for the containment structure steel components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.1.2.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for containment structure steel components are loss of material, cracking, and change in material properties.

For the loss of material aging effect, applicant identified the following plausible aging mechanisms: (1) material compatibility, (2) mechanical wear, (3) corrosion, and (4) aggressive chemical attack. Of these aging mechanisms, the applicant stated that corrosion and aggressive chemical attack (due to boric acid) are applicable for containment structure steel components exposed to containment air or borated water leaks. As such, the applicant committed to manage loss of material using their ASME Section XI, Subsection IWE inservice inspection program for containment steel exposed to air and their boric acid wastage surveillance program for containment steel exposed to borated water leaks. The staff agrees with the applicant's findings.

For the cracking aging effect, the applicant identified the following plausible aging mechanisms: (1) stress corrosion and (2) fatigue. The applicant stated that neither of these aging mechanisms are applicable for containment structure steel components at Turkey Point that are exposed to any environment, and therefore, listed no aging management program to manage cracking in Table 3.6-2 of the LRA. In RAI 3.6.1.2-2 the staff requested that the applicant evaluate the potential for cracking of the radiant energy shields and reactor vessel supports due to stress corrosion cracking and thermal fatigue. These items along with the fuel transfer blind flanges, and non-safety-related pipe segments are made of stainless steel and listed in Table 3.6-2 of the LRA. Section 3.6.1.5 of the LRA provides only a brief explanation for concluding that these items do not require aging management. In its response, the applicant responded that, as stated in Section 5.2 of Appendix C to the LRA, cracking is a non-ductile failure of a component due to stress corrosion, fatigue, or embrittlement. Stress corrosion cracking (SCC) requires a combination of a susceptible material and tensile stress. Cracking due to thermal fatigue requires cyclic thermal stresses beyond the material endurance limit. The environment for the stainless steel components discussed in the RAI is containment air, which is dry. These components are not exposed to the corrosive environment necessary to cause SCC. Consequently, SCC is not an aging effect requiring management for these components. The applicant further stated that, by design, the components discussed in the RAI are not exposed to cyclic thermal stresses of the quantity or magnitude necessary to cause thermal fatigue. Consequently, thermal fatigue is not an aging mechanism that can lead to cracking for these components. The staff finds this response adequate to resolve RAI 3.6.1.2-2.

For the change in material properties aging effect, the applicant identified the following plausible aging mechanisms: (1) elevated temperature, (2) irradiation embrittlement, and (3) embrittlement and permanent set of elastomers. Of these aging mechanisms, the applicant stated that only embrittlement and permanent set of elastomers is applicable for elastomers associated with containment structure steel components. As such, the applicant committed to manage change in material properties using their ASME Section XI, Subsection IWE inservice inspection program for elastomers associated with containment steel exposed to air. In RAI 3.6.1.2-3, the staff asked the applicant to evaluate the potential for material changes for the steam generator support material (Lubrite), and to justify its exclusion for items requiring aging management. The applicant indicated in its response that Lubrite is the trade name for a low-friction lubricant material used in applications where relative motion (sliding) is desired. At Turkey Point, the intended function of the Lubrite plates is to facilitate relative motion (sliding) during RCS heatup and cooldown. As described in an engineering brief supplied by the applicant's Lubrite vendor, Lubrite resists deformation, has a low coefficient of friction, resists softening at elevated temperatures, absorbs grit and abrasive particles, is not susceptible to corrosion, withstands high intensities of radiation, and will not score or mar. In addition, the applicant stated that Lubrite products are solid, permanent, completely self lubricating, and require no maintenance. Also, the Lubrite lubricants used in nuclear applications are designed for the environments to which they are exposed. The applicant also performed an extensive search of industry and plant-specific operating experience and found no reported instances of Lubrite plate degradation or failure to perform their intended function. Based on the above information, the applicant determined that there are no known aging effects for the Lubrite material that would lead to a loss of intended function. The staff agrees with the applicant's determination, and considers RAI 3.6.1.2-3 resolved.

The staff finds that the applicant's approach for evaluating the applicable aging effects for containment structure steel components to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for containment structure steel components.

3.6.1.2.2.2 Aging Management Programs

The aging management programs used by the applicant to manage the above aging effects are the ASME Section XI, Subsection IWE inservice inspection program and the boric acid wastage surveillance program.

The ASME Section XI, Subsection IWE inservice inspection program is credited with managing the effects of loss of material for containment steel components and change in material properties for elastomers associated with containment steel components. The program provides inspection and examination of accessible surface areas, including surfaces of welds, pressure-retaining bolting, and moisture barriers intended to prevent intrusion of moisture against inaccessible containment metallic surfaces. In its description of the ASME Section XI, Subsection IWE inservice inspection program, the applicant stated that visual inspections of steel components are performed to detect loss of material due to general corrosion and visual inspections of elastomers are performed to detect change in material properties. The results of the visual inspections are documented and the examinations performed during any inspection interval that reveal flaws or areas of degradation exceeding the acceptance criteria are expanded to include additional examinations within the same category. In RAI 3.6.1.2-4, the staff asked the applicant to provide a discussion of any plant-specific program content for inspection of Class CC metallic liners and pressure retention components, which is part of the ASME Section XI, Subsection IWE inservice inspection program. Specifically, the staff requested that the applicant provide a discussion of how the visual inspection of the internal and external surfaces and fasteners is to be implemented, thereby providing assurance that the containment shell and internal structures have not degraded due to corrosion. In its response, the applicant stated that the Turkey Point ASME Section XI, Subsection IWE inservice inspection program includes a visual examination of all accessible interior and exterior surfaces of the metallic shell and penetrations, thereby providing assurance that the containment shell and internal structures have not degraded due to corrosion. The program also requires visual examination of moisture barriers intended to prevent intrusion of moisture against inaccessible areas of the pressure retaining liner at concrete-to-metal interfaces and at metal-to-metal interfaces that are not seal welded (Category E-D), thereby providing assurance that the moisture barriers are not degraded. The staff finds the applicant's response to be acceptable. The ASME Section XI, Subsection IWE inservice inspection program is discussed in greater detail in Section 3.9.1.2 of this SER.

The boric acid wastage surveillance program is credited for aging management of carbon steel and low alloy steel components for the containment structure. The boric acid wastage surveillance program manages the effects of loss of material due to aggressive chemical attack of steel components through the detection of leakage of coolant containing boric acid. Conditions leading to boric acid corrosion are detected by visual inspections on external surfaces in accordance with plant procedures. The boric acid wastage surveillance program is discussed in greater detail in Section 3.9.3 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the aging effects for containment structure steel components will be adequately managed by the Turkey Point ASME Section XI, Subsection IWE inservice inspection program and the boric acid wastage surveillance program for the period of extended operation.

3.6.1.2.3 Conclusion

The staff has reviewed the information in Sections 3.6.1.2, "Containment Structure Steel Components," and Section 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment structure steel components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.1.3 Containment Structure Post-Tensioning System

3.6.1.3.1 Summary of Technical Information in the Application

Containment structure post-tensioning system components are described in Section 3.6.1.3 of the LRA. The post-tensioning system provides pre-compression for the containment structure. The containment structure post-tensioning system components identified by the licensee are the tendon wires and tendon anchorage.

Each tendon of the containment structure post-tensioning system is housed in spirally wrapped, corrugated, thin wall sheathing and capped at each end with a sheathing filler cap. After fabrication, the tendon is shop dipped in grease. The tendon sheathing provides the channel in the concrete through which the tendon is pulled, and contains the tendon sheathing filler material, which is grease. The tendon anchorages and tendon wires are contained in the sheathing filler material.

The aging effects identified by the applicant that could cause loss of intended function(s) for the containment structure post-tensioning system are loss of material and loss of prestress. The aging management program used by the applicant to manage these aging effects is the ASME Section XI, Subsection IWL inservice inspection program.

3.6.1.3.2 Staff Evaluation

In addition to Section 3.6.1.3 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results — Structures," and the applicable aging management program descriptions provided in Appendix B to the LRA to determine whether the aging effects for the containment structure post-tensioning system have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.1.3.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for the containment structure post-tensioning system are loss of material and loss of prestress. The applicant identified corrosion as an aging mechanism that can lead to the loss of material aging effect for the containment structure post-tensioning system. Corrosion can affect both the tendon wires within the grease-filled conduits and the anchorages providing the tendon terminations.

The applicant identified elevated temperatures, irradiation, stress relaxation of the prestressing wire, shrinkage, creep or elastic deformation of the concrete, anchorage seating losses, and tendon friction as aging mechanisms that can lead to the loss of prestress aging effect for the containment structure post-tensioning system. The loss of prestress aging effect is monitored by the applicant through a time-limited aging analysis and is discussed in greater detail in Section 4.5 of this SER.

The staff finds that the applicant's approach for evaluating the applicable aging effects for the containment structure post-tensioning system to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for the containment structure post-tensioning system.

3.6.1.3.2.2 Aging Management Programs

The aging management program used by the applicant to manage the loss of material aging effect for the containment structure post-tensioning system is the ASME Section XI, Subsection IWL inservice inspection program. The program provides for inspection of tendon wires and tendon anchorage hardware surfaces for loss of material, as well as a confirmatory program for measurement of tendons for loss of prestress. In its description of the ASME Section XI, Subsection IWL inservice inspection program, the applicant stated that unbonded post-tensioning system components are examined. These components consist of tendons, wires or strand, anchorage hardware and surrounding concrete, corrosion protection medium, and free water. Surface conditions are monitored through visual examinations to determine the extent of corrosion or concrete degradation around anchorage locations. The results of the visual inspections are documented in accordance with the corrective action program and areas of degradation are evaluated to determine if repair or replacement is required. The ASME Section XI, Subsection IWL inservice inspection program is discussed in greater detail in Section 3.9.1.4 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the loss of material aging effect for the containment structure post-tensioning system will be adequately managed by the ASME Section XI, Subsection IWL inservice inspection program for the period of extended operation.

3.6.1.3.3 Conclusion

The staff has reviewed the information in Sections 3.6.1.3, "Containment Structure Post-Tensioning System," and 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment structure post-tensioning system will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.1.4 Containment Internal Structural Concrete Components

3.6.1.4.1 Summary of Technical Information in the Application

Containment internal structural concrete components are described in Section 3.6.1.4 of the LRA. The containment structure provides radiation shielding, protects the reactor vessel and other safety-related systems, equipment, and components against missiles and environmental conditions, and serves as the last engineered barrier to the release of radioactivity. The following containment internal structural concrete components are identified by the applicant:

- reinforced concrete primary shield walls

- reinforce concrete secondary shield walls
- reinforced concrete upper secondary compartment walls (steam generator and pressurizer cubicles)
- reinforced concrete refueling cavity walls
- reinforced concrete containment sumps
- reinforced concrete equipment pads
- reinforced concrete missile shields

- reinforced concrete beams, floors, mats, and walls
- reinforce concrete curbs

These components were designed and constructed in accordance with ACI and ASTM standards. Containment internal structural concrete components are exposed to the containment air environment.

The aging effects identified by the applicant that could cause loss of intended function(s) for containment internal structural concrete components are loss of material, cracking, and change in material properties. However, the applicant determined for the containment internal structural concrete components that none of these aging effects required aging management for the period of extended operation and as such there is no aging management program used by the applicant to manage these aging effects.

3.6.1.4.2 Staff Evaluation

In addition to Section 3.6.1.4 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results – Structures" and the applicable aging management program descriptions provided in Appendix B to the LRA to determine whether the aging effects for the containment internal structural concrete components have been completely identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.1.4.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for the containment internal structural concrete components are loss of material, cracking, and change in material properties. Loss of material is manifested in containment internal structural concrete components as scaling, spalling, pitting, and erosion. Cracking is manifested in containment internal structural concrete components as complete or incomplete separation of the concrete into two or more parts. Change in material properties is manifested in containment internal structural concrete components as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength.

For the loss of material aging effect, the applicant identified the following plausible aging mechanisms: (1) freeze-thaw, (2) abrasion and cavitation, (3) aggressive chemical attack, (4) corrosion of reinforcing and embedded steel, and (5) elevated temperature. The applicant stated that none of these aging mechanisms are applicable for containment internal structural concrete components at Turkey Point and therefore, listed no aging management program to manage loss of material in Table 3.6-2 of the LRA.

For the cracking aging effect, the applicant identified the following plausible aging mechanisms: (1) freeze-thaw, (2) reactions with aggregates, (3) shrinkage, (4) settlement, (5) fatigue, and (6) elevated temperature. The applicant stated that none of these aging mechanisms are applicable for containment internal structural concrete components at Turkey Point and therefore, listed no aging management program to manage cracking in Table 3.6-2 of the LRA.

For the change in material properties aging effect, the applicant identified the following plausible aging mechanisms: (1) leaching, (2) creep, (3) aggressive chemical attack, (4) irradiation embrittlement, and (5) elevated temperature. The applicant stated that none of these aging mechanisms are applicable for containment internal structural concrete components at Turkey Point and therefore, listed no aging management program to manage change in material properties in Table 3.6-2 of the LRA.

The staff considers that each of the above aging effects (loss of material, cracking, change in material properties) are both plausible and applicable for containment internal structural concrete components at Turkey Point. As such, in RAI 3.6.2.1-2 the staff requested that the applicant provide an aging management program to manage the aging of reinforced concrete structures. In its initial response, the applicant reasserted its position that aging management reviews performed on above groundwater reinforced concrete did not identify any aging effects requiring management. However, FPL proposed to use its inspections of containment structure concrete components through the ASME Section XI, IWL inservice inspection program as an indicator for the condition of other reinforced concrete structures. Subsequent communication between the staff and FPL culminated in a staff letter, dated October 30, 2001, in which the staff asserted its position that all concrete structures within the scope of license renewal require aging management via a dedicated aging management program. In its response to the staff's position (see supplemental response to RAI 3.6.2.1-2 dated November 1, 2001), the applicant committed to manage the aging of containment internal structural concrete components for loss of material, cracking, and change in material properties through inspections performed by its systems and structural monitoring aging management program. Once incorporate, as committed in this response, the staff considers this issue to be resolved.

3.6.1.4.2.2 Aging Management Programs

Since the applicant did not identify any plausible and applicable aging effects for containment internal structural concrete components, there are no aging management programs listed in Table 3.6-2 of the LRA to manage loss of material, cracking, and change in material properties. However, in response to the staff's position that each of these aging effects are both plausible and applicable for containment internal structural concrete components, the applicant committed in its supplemental response to RAI 3.6.2.1-2 to manage the aging of containment internal structural concrete components for loss of material, cracking, and change in material properties through inspections performed by its systems and structural monitoring aging management program. The systems and structures monitoring program is discussed in greater detail in Section 3.1.3 of this SER.

3.6.1.4.3 Conclusion

The staff has reviewed the information in Sections 3.6.1.2, "Containment Internal Structural Concrete Components," and 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment internal structural concrete components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.1.5 Containment Internal Structural Steel Components

3.6.1.5.1 Summary of Technical Information in the Application

Containment internal structure steel components are described in Section 3.6.1.5 of the LRA. The purpose of the containment internal structural steel components is to provide several safety-related functions including serving as a pressure boundary or a fission-product retention barrier, providing structural and/or functional support to safety-and non-safety-related equipment, and serving as missile and flood protection barriers. The following containment internal structural steel components are identified by the applicant:

- equipment component supports
- heating, ventilation, and air-conditioning (HVAC) ductwork supports
- piping supports
- pipe whip restraints
- cable trays, conduits, and supports
- electrical and instrument panels and enclosures
- anchorages/embedments exposed surfaces
- instrument line supports
- instrument racks and frames
- structural steel beams and columns
- stairs, platforms, and grating
- sump screens
- Lubrite plates
- radiant energy shields
- polar crane
- reactor coolant system supports (including reactor vessel supports, steam generator supports, reactor coolant pump supports, pressurizer supports, and the surge line support)
- non-safety-related piping between class break and anchor

The containment internal structural steel components were designed in accordance with the AISC "Manual of Steel Construction." The materials of construction for the reactor coolant system supports include structural steel, low alloy steel, and carbon steel pipe. The primary bolting material is carbon steel. Pipe segments beyond the safety-related/non-safety-related boundaries are constructed of carbon and stainless steel and consist of piping and inline components.

Containment internal structural steel components are exposed to containment air and treated water. Borated water is also a potential environment for containment internal structural steel components.

The aging effects identified by the applicant that could cause loss of intended function(s) for containment structure steel components are loss of material, cracking, and change in material properties. The aging management programs used by the applicant to manage these aging effects are the ASME Section XI, Subsection IWF inservice inspection program, the boric acid wastage surveillance program, and the systems and structures monitoring program.

3.6.1.5.2 Staff Evaluation

In addition to Section 3.6.1.5 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results – Structures," and the applicable aging management program descriptions provided in Appendix B of the LRA to determine whether the aging effects for the containment internal structural steel components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.1.5.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for containment internal structural steel components are loss of material, cracking, and change in material properties.

For the loss of material aging effect, applicant identified the following plausible aging mechanisms: (1) material compatibility, (2) mechanical wear, (3) corrosion, and (4) aggressive chemical attack. Of these aging mechanisms, the applicant stated that corrosion and aggressive chemical attack (due to boric acid) are applicable for containment internal structural steel components at Turkey Point that are exposed to containment air or borated water leaks. As such, the applicant committed to manage loss of material using their ASME Section XI, Subsection IWF inservice inspection program or the systems and structures monitoring program for containment internal structural steel exposed to air and their boric acid wastage surveillance program for containment steel exposed to borated water leaks.

For the cracking aging effect, the applicant identified the following plausible aging mechanisms: (1) stress corrosion and (2) fatigue. The applicant stated that neither of these aging mechanisms are applicable for containment internal structural steel components at Turkey Point that are exposed to any environment, and therefore, listed no aging management program to manage cracking in Table 3.6-2 of the LRA.

For the change in material properties aging effect, the applicant identified the following plausible aging mechanisms: (1) elevated temperature, (2) irradiation embrittlement, and (3) creep and stress relaxation. The applicant stated that neither of these aging mechanisms are applicable for containment internal structural steel components at Turkey Point that are exposed to any environment, and therefore, listed no aging management program to manage change in material properties in Table 3.6-2 of the LRA.

In addition to the three aging effects listed above, the staff requested in RAI 3.6.1.5-3 that the applicant provide the technical justification for not including loss of preload as an aging effect for expansion and undercut anchors due to the effects of vibration on the surrounding concrete. In its response, the applicant stated that the FPL design specification for expansion and undercut anchors specifically prohibits the use of these anchors in vibratory service conditions. In addition, the applicant stated that any degradation due to vibratory loading would occur relatively early in plant life and such an occurrence would be detected and corrective actions implemented to preclude recurrence. Therefore, degradation due to vibration is not an aging effect requiring management for Turkey Point. The staff finds this response acceptable.

The staff finds that the applicant's approach for evaluating the applicable aging effects for containment internal structural steel components to be reasonable and acceptable. The staff concludes that the applicant has properly identified the aging effects for containment internal structural steel components.

3.6.1.5.2.2 Aging Management Programs

The aging management programs used by the applicant to manage the above aging effects are the ASME Section XI, Subsection IWF inservice inspection program, the boric acid wastage surveillance program, and systems and structures monitoring program.

The ASME Section XI, Subsection IWF inservice inspection program is credited with managing the effects of loss of material for Class 1, 2, and 3 component supports in the containment. The program provides for a visual inspection of the surfaces of component supports for evidence of surface irregularities such as flaking, blistering, peeling or discoloration. The results of the visual examinations are documented and the component supports that are subjected to corrective measures are reexamined during the next inspection period. The ASME Section XI, Subsection IWF inservice inspection program is discussed in greater detail in Section 3.9.1.3 of this SER.

The boric acid wastage surveillance program is credited for aging management of carbon steel and low alloy steel components for the containment structure. The boric acid wastage surveillance program manages the effects of loss of material due to aggressive chemical attack of steel components through the detection of leakage of coolant containing boric acid. Conditions leading to boric acid corrosion are detected by visual inspections on external surfaces in accordance with plant procedures. In RAI 3.6.1.5-5, the staff requested that FPL provide the basis for omitting the boric acid waste surveillance program as the aging management program for containment anchorages/embedments that are located above the groundwater table or in an air conditioned environment. In its response, the applicant stated that Table 3.6-2 of the LRA lists two types of containment anchorages/embedments, which are located above groundwater. Specifically, these anchorages/embedments are those encased in concrete and those exposed to containment air. For anchorages and embedments that are encased in concrete, the applicant maintains that the surrounding concrete protects the anchorages/embedments; thus, aging management is not required. However, the applicant confirmed that the anchorages/embedments that are exposed to containment air and borated water leaks are subject to the loss of material aging effect, and are managed by the boric acid wastage surveillance program. The boric acid wastage surveillance program is discussed in greater detail in Section 3.9.3 of this SER.

The systems and structures monitoring program is credited with managing the loss of material aging effect due to corrosion for containment internal structural steel components other than component supports. The structural monitoring program provides condition monitoring and appraisal of containment internal structural steel components through periodic visual inspections. In its description of the systems and structures monitoring program, the applicant stated that external surfaces of steel components are examined for evidence of corrosion such as flaking, blistering, peeling or discoloration. The results of the visual inspections are documented and the frequency of the inspection may be adjusted based on the inspection results and industry experience. In RAI 3.6.1.5-1 the staff requested that the applicant explain the omission of the systems and structures monitoring program as an aging management program for the galvanized carbon steel components such as electrical, instrument panels and enclosures, miscellaneous steel (stairs, platforms, and grating). In its response, the applicant stated that galvanized carbon is steel that is not considered to be susceptible to general corrosion except where buried, submerged in fluid, or subject to wetting. Hence, the boric acid wastage surveillance program is the only aging management program for these galvanized carbon steel components. The systems and structures monitoring program is discussed in greater detail in Section 3.1.3 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the aging effects for containment internal structural steel components will be adequately managed by the Turkey Point ASME Section XI, Subsection IWF inservice inspection program, the boric acid wastage surveillance program, and the systems and structures monitoring program for the period of extended operation.

3.6.1.5.3 Conclusion

The staff has reviewed the information in Sections 3.6.1.5, "Containment Internal Structural Steel Components," and 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the containment internal structural steel components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.2 Other Structures

3.6.2.1 Steel-in-Air Structural Components

3.6.2.1.1 Summary of Technical Information in the Application

The following steel-in-air structural components are described in Section 3.6.2.1 of the LRA:

- framing, bracing, and connections
- decking, grating, and checkered plate
- stairs and ladders
- exposed anchors and embedments
- piping, duct, and component supports
- non-safety-related piping between class break and anchor
- crane rails and girders

- cable trays, conduits, and electrical enclosures
- instrumentation supports
- instrument racks and frames

The in-scope steel-in-air structural components listed above are found in the auxiliary building, control building, intake structure, turbine building, yard structures, and other smaller, miscellaneous enclosures listed in Tables 3.6-3 through 3.6-20.

The applicant stated that the steel-in-air structural components were designed and constructed in accordance with AISC standards. Turkey Point steel-in-air structural components are constructed of painted or galvanized carbon steel and stainless steel. The applicant stated that pipe segments beyond the safety-related/non-safety-related boundaries are constructed of carbon and stainless steel and consist of piping and inline components. The steel-in-air structural components are exposed to containment air, outdoor, indoor (both air and non-air conditioned), and potential borated water leak environments.

The aging effects identified by the applicant that could cause loss of intended function(s) for steel-in-air structural components are loss of material, cracking, and change in material properties. The aging management programs used by the applicant to manage these aging effects are the systems and structures monitoring program, ASME Section XI, Subsection IWF inservice inspection program, and boric acid wastage surveillance program.

3.6.2.1.2 Staff Evaluation

In addition to Section 3.6.2.1 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results – Structures" and the applicable aging management program descriptions provided in Appendix B to the LRA to determine whether the aging effects for the steel-in-air structural components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.2.1.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for the steel-in-air structural components are loss of material, cracking, and change in material properties.

For the loss of material aging effect, the applicant identified the following plausible aging mechanisms: (1) mechanical wear, (2) corrosion, (3) and aggressive chemical attack. Of these aging mechanisms the applicant identified corrosion of carbon steel in an air environment and aggressive chemical attack due to boric acid as applicable to steel structural components. As such, the applicant committed to manage loss of material for steel-in-air structural components.

For the cracking aging effect, the applicant identified the following plausible aging mechanisms: (1) stress corrosion and (2) fatigue. The applicant stated that neither of these aging mechanisms is applicable for steel-in-air structural components at Turkey Point, and therefore, listed no aging management program to manage cracking for these components in Tables 3.6-3 through 3.6-20 in the LRA.

For the change in material properties aging effect, the applicant identified thermal and irradiation embrittlement as a plausible aging mechanism. The applicant determined that since none of the steel-in-air structural components outside the containment are exposed to elevated temperatures or fluences that would cause reduction in fracture toughness, that change in material properties is not an aging effect requiring management for steel-in-air structural components.

The staff finds the applicant's approach for evaluating the applicable aging effects for steel-in-air structural components to be reasonable and acceptable. The staff concludes that the applicable aging effects for steel-in-air structural components have been properly identified.

3.6.2.1.2.2 Aging Management Programs

The aging management programs used by the applicant to manage the above aging effects are the ASME Section XI, Subsection IWF inservice inspection program, the boric acid wastage surveillance program, and systems and structures monitoring program.

The ASME Section XI, Subsection IWF inservice inspection program is credited with managing the effects of loss of material for Class 1, 2, and 3 component supports in the auxiliary building, containment, emergency diesel generator buildings, and yard structures. The program provides for a visual inspection of the surfaces of component supports for evidence of surface irregularities such as flaking, blistering, peeling or discoloration. The results of the visual examinations are documented and the component supports that are subjected to corrective measures are reexamined during the next inspection period. The ASME Section XI, Subsection IWF inservice inspection program is discussed in greater detail in Section 3.9.1.3 of this SER.

The boric acid wastage surveillance program is credited for aging management of carbon steel and low alloy steel components for several different systems and structures. The boric acid wastage surveillance program manages the effects of loss of material due to aggressive chemical attack of steel components through the detection of leakage of coolant containing boric acid. Conditions leading to boric acid corrosion are detected by visual inspections on external surfaces in accordance with plant procedures. The boric acid wastage surveillance program is discussed in greater detail in Section 3.9.3 of this SER.

The systems and structures monitoring program is credited with managing the loss of material aging effect due to corrosion for steel-in-air structural components other than component supports. The structural monitoring program provides condition monitoring and appraisal of structural steel components through periodic visual inspections. In its description of the systems and structures monitoring program, the applicant stated that external surfaces of steel components are examined for evidence of corrosion such as flaking, blistering, peeling or discoloration. The results of the visual inspections are documented and the frequency of the inspection may be adjusted based on the inspection results and industry experience. In Tables 3.6-3 and 3.6-13 of the LRA, the applicant credited the systems and structures monitoring program with managing the loss of material aging effect for anchorages/embedments located below groundwater elevation. In RAI 3.6.2.1-1 the staff questioned the effectiveness of the systems and structures monitoring program for managing the loss of material aging effect for the normally inaccessible steel components that are enclosed in concrete below groundwater elevation. In its response, the applicant stated that the systems and structures monitoring program will manage aging of concrete below the groundwater elevation by direct visual inspections of exposed surfaces of the concrete structures (i.e., intake structure and auxiliary building). Visual inspections of exposed surfaces of concrete below groundwater elevation will identify signs of degradation (e.g., concrete cracking, spalling, scaling, leaching, discoloration, groundwater in-leakage, or rust stains). Satisfactory inspection of the concrete surfaces will ensure adequate aging management for the steel anchorages/embedments below groundwater elevation. The applicant's response is acceptable to the staff. The systems and structures monitoring program is discussed in greater detail in Section 3.1.3 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the aging effects for steel-in-air structural components will be adequately managed by the Turkey Point ASME Section XI, Subsection IWF inservice inspection program, the boric acid wastage surveillance program, and the systems and structures monitoring program for the period of extended operation.

3.6.2.1.3 Conclusion

The staff has reviewed the information in Sections 3.6.2.1, "Steel-In-Air Structural Components," and 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the steel-in-air structural components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.2.2 Steel-in-Fluid Structural Components

3.6.2.2.1 Summary of Technical Information in the Application

The following steel-in-fluid structural components are described in Section 3.6.2.2 of the LRA:

- refueling pool cavity liner plates
- spent fuel pool liner plates
- spent fuel handling equipment and tools
- spent fuel pool keyway gates
- fuel transfer tubes, penetration sleeves, and gate valves
- reactor cavity seal rings
- spent fuel pool anchorages and embedments
- intake structure traveling screens

The in-scope steel-in-fluid structural components listed above are found primarily in the intake structure, spent fuel storage and handling, and yard structures listed in Tables 3.6-13, 3.6-16, and 3.6-20.

The applicant stated that the steel-in-fluid structural components were designed and constructed in accordance with AISC standards. Turkey Point steel-in-fluid structural components are constructed of painted or galvanized carbon steel and stainless steel. In addition, the spent fuel storage racks contain Boraflex panels. The steel-in-fluid structural components are exposed to fluid environments of raw water, borated water, and indoor, outdoor and containment air environments.

The aging effects identified by the applicant that could cause loss of intended function(s) for steel-in-fluid structural components are loss of material, cracking, and change in material properties. The aging management programs used by the applicant to manage these aging effects are the systems and structures monitoring program, Boraflex surveillance program, chemistry control program, and periodic surveillance and preventive maintenance program.

3.6.2.2.2 Staff Evaluation

In addition to Section 3.6.2.2 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results – Structures," and the applicable aging management program descriptions provided in Appendix B of the LRA to determine whether the aging effects for the steel-in-fluid structural components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.2.2.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for the steel-in-fluid structural components are loss of material, cracking, and change in material properties.

For the loss of material aging effect, the applicant identified the following plausible aging mechanisms: (1) leaching, (2) aggressive chemical attack, (3) mechanical wear, and (4) corrosion. In Appendix C of the LRA, the applicant stated that leaching, aggressive chemical attack, and mechanical wear are not aging mechanisms that can lead to the loss of material aging effect for steel-in-fluid structural components. Loss of material due to selective leaching is an aging effect requiring management for admiralty brass, brass, and gray cast iron in treated water environments and is not applicable to structural steel components. Aggressive chemical attack due to boric acid is an applicable aging mechanism for steel-in-air structural components, since steel-in-air may be exposed to highly concentrated boric acid solutions resulting from borated water leaks. However, steel-in-fluid structural components are not exposed to highly concentrated boric acid solutions and therefore the applicant determined that aggressive chemical attack is not an applicable aging mechanism that will lead to the loss of material aging effect. The applicant also determined that there is no mechanical wear for the in-scope steel-in-fluid structural components. Corrosion is identified as an aging mechanism that can lead to the loss of material aging effect for steel-in-fluid structural components.

For the cracking aging effect, the applicant identified the following plausible aging mechanisms: (1) fatigue, (2) hydrogen damage and (3) stress corrosion cracking. In Appendix C of the LRA, the applicant stated that none of these three aging mechanisms can lead to the cracking aging effect for steel-in-fluid structural components at Turkey Point, and therefore, listed no aging management program to manage cracking for these components in Tables 3.6-3 through 3.6-20 in the LRA. The applicant stated that vibration induced fatigue is fast acting and typically detected early in a component's life, at which time corrective actions are initiated to prevent recurrence. These corrective actions typically involve modifications to the plant, such as the addition of supplemental restraints to a piping system, replacement of tubing with flexible hose, etc. Based upon these considerations, the applicant concluded that cracking due to vibration induced fatigue is not an aging effect requiring management. The applicant also concluded, based on its own operating experience and a review of other PWR treated water systems, that cracking due to hydrogen damage is not an aging effect requiring management for stainless steel components. The applicant also stated that for carbon steels, stress corrosion cracking occurs most commonly in the presence of aqueous chlorides. Industry data do not indicate a significant problem of stress corrosion cracking of low strength carbon steels. Therefore, the applicant concluded that stress corrosion cracking of carbon steels is not an aging effect requiring management. Based on the above, the applicant concluded that cracking is not an aging effect requiring management for steel-in-fluid structural components.

For the change in material properties aging effect, the applicant identified the following plausible aging mechanisms: (1) creep and stress relaxation and (2) thermal and irradiation embrittlement. Regarding creep and stress relaxation, the applicant stated that this aging mechanism can lead to change in material properties for steel components if the component operating temperature approaches or exceeds the crystallization temperature for the metal. Austenitic stainless steel with temperatures less than 800 °F and carbon steel and low alloy steels with temperatures less than 700 °F are not susceptible to creep and stress relaxation. All Turkey Point plant systems operate at temperatures below 700 °F and, thus, are not susceptible to creep and stress relaxation. Therefore, the applicant concluded that creep and stress relaxation would not lead to change in material properties for steel-in-fluid structural components at Turkey Point. Regarding thermal and irradiation embrittlement, the applicant stated that steel-in-fluid structural components outside containment are not exposed to the elevated temperatures or fluences that would cause reduction in fracture toughness. However, the applicant stated that Boraflex panels, which are neutron absorbers inserted between the fuel storage cells in high-density fuel storage racks, are susceptible to change in material properties resulting from irradiation embrittlement.

The staff finds the applicant's approach for evaluating the applicable aging effects for steel-in-fluid structural components to be reasonable and acceptable. The staff concludes that the applicable aging effects for steel-in-fluid structural components have been properly identified.

3.6.2.2.2.2 Aging Management Programs

The aging management programs used by the applicant to manage the above aging effects are the Boraflex surveillance program, the chemistry control program, and the systems and structures monitoring program.

The Boraflex surveillance program is credited with managing the change in material properties aging effect for the Boraflex panels that are inserted between the fuel storage cells in the fuel storage racks. The Boraflex surveillance program seeks to determine the amount of degradation of the Boraflex material through blackness testing and tracking of the spent fuel pool silica levels. The results of the Boraflex surveillance testing are evaluated to determine the schedule for subsequent testing. The Boraflex surveillance program is discussed in greater detail in Section 3.9.2 of this SER.

The chemistry control program is credited with managing the loss of material aging effect for spent fuel storage and handling stainless steel components in a treated water-borated environment. The chemistry control program includes sampling activities and analysis for treated water-borated that determine the amount of corrosion inhibitors that is introduced to prevent loss of material. The acceptance criteria for the chemistry control program are in accordance with the Nuclear Chemistry Parameters Manual, Technical Specifications, and appropriate plant procedures. The chemistry control program is discussed in greater detail in Section 3.1.1 of this SER.

The systems and structures monitoring program is credited with managing the loss of material aging effect for most of the steel-in-fluid structural components. The structural monitoring program provides condition monitoring and appraisal of components through periodic visual inspections. In its description of the systems and structures monitoring program, the applicant stated that external surfaces of steel components are examined for evidence of corrosion such as flaking, blistering, peeling or discoloration. The results of the visual inspections are documented and the frequency of the inspection may be adjusted based on the inspection results and industry experience. The systems and structures monitoring program is discussed in greater detail in Section 3.1.3 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the aging effects for steel-in-fluid structural components will be adequately managed by the Boraflex surveillance program, the chemistry control program, and the systems and structures monitoring program for the period of extended operation.

3.6.2.2.3 Conclusion

The staff has reviewed the information in Sections 3.6.2.2, "Steel-In-Fluid Structural Components," and 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the steel-in-fluid structural components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.2.3 Concrete Structural Components

3.6.2.3.1 Summary of Technical Information in the Application

The concrete structural components are described in Section 3.6.2.3 of the LRA. Concrete structural components include foundations, columns, walls, floors, roofs, equipment pads, electric duct banks, manholes, trenches, masonry block walls, embedded steel, embedded anchors, and concrete piping. The in-scope concrete structural components listed above are found in the auxiliary building, cold chemistry lab, control building, diesel driven fire pump enclosure, discharge structure, electrical penetration rooms, emergency diesel generator buildings, intake structure, main steam and feedwater platforms, plant vent stack, spent fuel storage and handling room, turbine building, chimneys, and yard structures, listed in Tables 3.6-3 through 3.6-20.

The applicant stated that the concrete structural components were designed and constructed in accordance with ACI and ASTM standards. Turkey Point concrete structural components are exposed to environments of outdoor, indoor-not air-conditioned, indoor-air conditioned, buried, raw water cooling canals, and embedded/encased.

The aging effects identified by the applicant that could cause loss of intended function(s) for concrete structural components are loss of material, cracking, and change in material properties. The aging management program used by the applicant to manage these aging effects is the systems and structures monitoring program.

3.6.2.3.2 Staff Evaluation

In addition to Section 3.6.2.3 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results – Structures," and the applicable aging management program descriptions provided in Appendix B of the LRA to determine whether the aging effects for the containment structure concrete components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.2.3.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for concrete structural components are loss of material, cracking, and change in material properties. Loss of material is manifested in concrete structural components as scaling, spalling, pitting and erosion. Cracking is manifested in concrete structural components as complete or incomplete separation of the concrete into two or more parts. Change in material properties is manifested in concrete structural components as increased permeability, increased porosity, reduction in pH value, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength.

For the loss of material aging effect, the applicant identified the following plausible aging mechanisms: (1) freeze-thaw, (2) abrasion and cavitation, (3) elevated temperature, (4) aggressive chemical attack, and (5) corrosion of reinforcing and embedded/encased steel. Of these aging mechanisms, the applicant stated that only aggressive chemical attack and corrosion of reinforcing and embedded steel are applicable for concrete structural components exposed to groundwater or saltwater. As such, the applicant committed to manage loss of material only for reinforced concrete structures that are below groundwater elevation or exposed to saltwater.

For the cracking aging effect, the applicant identified the following plausible aging effects: (1) freeze-thaw, (2) reactions with aggregates, (3) fatigue, (4) shrinkage, (5) settlement, and (6) elevated temperature. Of these aging mechanisms, the applicant stated that only shrinkage and settlement are applicable for unreinforced masonry block walls at Turkey Point. For all other concrete structural components, the applicant stated that none of the above aging mechanisms are applicable and therefore, listed no aging management program to manage cracking for these components in Tables 3.6-3 through 3.6-20 of the LRA. For unreinforced masonry block walls, the applicant committed to using the systems and structures monitoring aging management program to manage cracking.

For the change in material properties aging effect, the applicant identified the following plausible aging effects: (1) leaching, (2) creep, (3) elevated temperature, (4) irradiation embrittlement, and (5) aggressive chemical attack. Of these aging mechanisms, the applicant stated that only aggressive chemical attack is applicable for concrete structural components exposed to groundwater or saltwater. As such, the applicant committed to manage change in material properties only for concrete structural components that are below groundwater elevation or exposed to saltwater.

The staff considers that each of the above aging effects (loss of material, cracking, change in material properties) are both plausible and applicable for containment internal structural concrete components. As such, in RAI 3.6.2.1-2 the staff requested that the applicant provide an aging management program to manage the aging of concrete structural components. In its initial response, the applicant reasserted its position that aging management reviews performed on above groundwater reinforced concrete did not identify any aging effects requiring management. However, FPL proposed to use its inspections of containment structure concrete components through the ASME Section XI, IWL inservice inspection program as an indicator for the condition of other reinforced concrete structures. Subsequent communication between the staff and FPL culminated in a staff letter, dated October 30, 2001, in which the staff asserted its position that all concrete structures within the scope of license renewal require aging management. In its response to the staff's position (see supplemental response to RAI 3.6.2.1-2 dated November 1, 2001), the applicant committed to manage the aging of concrete structural components for loss of material, cracking, and change in material properties through inspections performed by its systems and structural monitoring aging management program. Once incorporated, as committed in this response, the staff considers this issue to be resolved.

3.6.2.3.2.2 Aging Management Programs

The aging management program used by the applicant to manage the above aging effects is the systems and structures monitoring program. The structural monitoring program provides condition monitoring and appraisal of concrete structural components through periodic visual inspections. In its supplemental response to RAI 3.6.2.1-2 (dated November 1, 2001), the applicant stated that the "Parameters Monitored or Inspected" section of the systems and structural monitoring program has been revised to include the monitoring for change in material properties, cracking, and loss of material of all reinforced concrete components and not just those concrete components located below groundwater elevation or exposed to saltwater. The results of the visual inspection for systems, structures and components will be documented and the frequency for the inspection may be adjusted, as necessary, based on the inspection results and industry experience. In RAI 3.6.2.3-1, the staff requested that the applicant identify the specific inspection procedure used by the systems and structures monitoring program for monitoring the condition of masonry block walls. In its response, the applicant stated that the inspection procedures require visual inspection of masonry walls for signs of degradation, including cracks, missing or degraded mortar, missing or degraded masonry units, and degradation at bracing connections. When cracks are identified, they are evaluated under the Corrective Action Program to ensure the extent of cracking does not invalidate the evaluation basis established either in response to IEB 80-11 or established for implementation of USI A-46. The response is acceptable to the staff because all the components of the masonry block walls are inspected and the safety of the masonry block walls is evaluated against the acceptable criteria to the staff. The systems and structures monitoring program is discussed in more detail in Section 3.1.3 of this SER.

On the basis of the information discussed above, the staff concludes that the applicant has demonstrated that the aging effects for concrete structural components and masonry block walls will be adequately managed by the systems and structures monitoring program for the period of extended operation.

3.6.2.3.3 Conclusion

The staff has reviewed the information in Sections 3.6.2.3, "Concrete Structural Components," 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA and responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the concrete structural components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.6.2.4 Miscellaneous Structural Components

3.6.2.4.1 Summary of Technical Information in the Application

Miscellaneous structural components are described in Section 3.6.2.4 of the LRA. Miscellaneous structural components include fire rate assemblies, cooling water canals, weatherproofing, flood protection seals and stop logs, and control room ceiling and raised floor. The in-scope miscellaneous structural components listed above are found in the control building, cooling water canals, electrical penetration rooms, emergency diesel generator buildings, fire protection monitoring station, fire rated assemblies, spent fuel storage and handling, and turbine building listed in Tables 3.6-3 through 3.6-20.

The applicant stated that the miscellaneous structural components consist of a variety of materials including painted and galvanized carbon steel, aluminum, earth/rock, wood, gypsum board, acoustical panels, weatherproofing materials, and fire protection materials. Turkey Point miscellaneous structural components are exposed to environments of outdoor and indoor air (both air and non-air conditioned).

The aging effects identified by the applicant that could cause loss of intended function(s) for miscellaneous structural components are loss of material and loss of seal. The aging management programs used by the applicant to manage these aging effects are the systems and structures monitoring program, fire protection program, and periodic surveillance and preventive maintenance program.

3.6.2.4.2 Staff Evaluation

In addition to Section 3.6.2.3 of the LRA, the staff reviewed the pertinent information provided in Section 2.4, "Scoping and Screening Results – Structures," and the applicable aging management program descriptions provided in Appendix B of the LRA to determine whether the aging effects for the miscellaneous structural components have been properly identified and will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

3.6.2.4.2.1 Effects of Aging

The aging effects identified by the applicant that could cause loss of intended function(s) for the miscellaneous structural components are loss of material and loss of seal.

Aging mechanisms, identified by the applicant, that can lead to the loss of material aging effect for miscellaneous structural components are wear, weathering, corrosion, and organic decomposition. The applicant evaluated each of these aging mechanisms for loss of material of the fire rated assemblies, cooling water canals, weatherproofing (structures and sealants), control room ceiling and raised floor, and the fire protection monitoring station miscellaneous structural components. In particular, the applicant determined that loss of material due to (1) weathering is an aging effect requiring management for Thermo-Lag insulation materials, (2) corrosion is an aging effect requiring management for certain fire doors since these doors are constructed of carbon steel, and (3) organic decomposition is an aging effect requiring management for wooden stop logs, which provide flood protection. However, the applicant determined that loss of material is not an applicable aging effect for (1) the control room ceiling and raised floor and the fire protection monitoring station since these are indoor - air conditioned environments that are occupied 24 hours per day, (2) fire penetration seals, as concluded in SECY 96-146, (3) the cooling water canals and (4) aluminum stop logs. In response to RAI 3.6.2.4-1, the applicant stated that based on its plant operating experience and the findings of SECY 96-146, fire penetration seals are not subject to aging effects. The applicant further clarified that, as part of the plants' existing fire protection program mandated by Appendix R to 10 CFR Part 50 and Branch Technical Position (BTP) 9.5-1, visual inspections of the fire penetrations will continue to be performed. In response to RAI 3.6.2.4-3, the applicant stated that since aluminum is highly resistant to corrosion, there are no aging effects that would cause a loss of intended function for aluminum stop logs and pipe trench penetration seals. Based on the above, the applicant concluded that loss of material due to weathering, corrosion, and organic decomposition is an aging effect requiring management for miscellaneous structural components. The staff concurs with the applicant's conclusions.

The aging mechanism, identified by the applicant, that can lead to loss of seal for miscellaneous structural components is weathering. The applicant determined that loss of seal due to weathering is an aging effect requiring management for manholes and associated sealants, weatherproofing components, and pipe trench penetration seals. Regarding the flood protection provided by wooden and aluminum stop logs, the applicant stated, in response to RAI 3.6.2.4-4, that the purpose of the wooden and aluminum stop logs is to provide a flood protection barrier against wave run-up and that the stop logs are not intended to be leak-tight barriers. The staff agrees with the applicant's response.

The staff finds the applicant's approach for evaluating the applicable aging effects for miscellaneous structural components to be reasonable and acceptable. The staff concludes that the applicable aging effects for miscellaneous structural components have been identified.

3.6.2.4.2.2 Aging Management Programs

The aging management programs used by the applicant to manage the above aging effects are the systems and structures monitoring program, fire protection program, and periodic surveillance and preventive maintenance program.

The systems and structures monitoring program is credited with managing the loss of material and loss of seal aging effects for many of the miscellaneous structural components. The structural monitoring program provides condition monitoring and appraisal of components through periodic visual inspections. In its description of the systems and structures monitoring program, the applicant stated that external surfaces of steel components are examined for evidence of corrosion such as flaking, blistering, peeling or discoloration and inspection of weatherproofing material for deterioration is performed. The results of the visual inspections are documented and the frequency of the inspection may be adjusted based on the inspection results and industry experience. The systems and structures monitoring program is discussed in greater detail in Section 3.1.3 of this SER.

The fire protection program is credited for aging management of specific components associated with fire protection and fire rated assemblies. The fire protection program manages the loss of material and loss of seal aging effects for the miscellaneous structural components associated with fire protection. The fire protection program provides condition monitoring and appraisal of components through periodic visual inspections. In its description of the fire protection program, the applicant stated that component surface conditions are monitored visually to determine the extent of external material degradation such as loss of material due to general, crevice, and pitting corrosion; and loss of seal or cracking due to embrittlement. The results of the visual inspections are documented and the frequency of the inspection may be adjusted based on the inspection results and industry experience. The fire protection program is discussed in greater detail in Section 3.9.8 of this SER.

The periodic surveillance and preventive maintenance program is credited for aging management of weatherproofing, pipe trench penetrations, and wooden stop logs. The periodic surveillance and preventive maintenance program manages the loss of material and loss of seal aging effects for these miscellaneous structural components. The periodic surveillance and preventive maintenance program provides condition monitoring and appraisal of components through periodic visual inspections. In its description of the periodic surveillance and preventive maintenance program, the applicant stated that component surface conditions are monitored visually to determine the extent of material degradation, such as loss of material due to organic decomposition and loss of seal due to weathering. The results of the visual inspections are documented, and the frequency of the inspection may be adjusted based on the inspection results and industry experience. The periodic surveillance and preventive maintenance program is discussed in greater detail in Section 3.9.11 of this SER.

3.6.2.4.3 Conclusion

The staff has reviewed the information in Sections 3.6.2.4, "Miscellaneous Structural Components," and 2.4, "Scoping and Screening Results — Structures," as well as the applicable aging management program descriptions provided in Appendix B to the LRA and responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated that the aging effects associated with the miscellaneous structural components will be adequately managed so that there is reasonable assurance that these structural components will perform their intended functions in accordance with the CLB during the period of extended operation.

3.7 Electrical and Instrumentation and Controls

The applicant described its AMR results of electrical/I&C components requiring AMR at Turkey Point, Units 3 and 4, in Section 3.7 of the LRA. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the effect of aging on the electrical/I&C components will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

On the basis of this review, the staff requested additional information in a letter to the applicant, dated January 17, 2001. The applicant responded to this request for additional information (RAI) in a letter to the staff, dated March 30, 2001. The applicant provided supplemental responses to the staff's RAI 3.7.1-1 on May 11 and May 29, 2001.

3.7.1 Summary of Technical Information in the Application

3.7.1.1 Non-Environmentally Qualified Insulated Cables and Connections, and Electrical/I&C Penetrations

In Section 3.7.1.1 of the LRA, the applicant described the process used to identify the applicable aging effects of the electrical/I&C components. The process is based on the Department of Energy (DOE) aging management guide (AMG). This AMG provides a comprehensive compilation and evaluation of information on the insulated cables and connections, spliced connections, and terminal blocks. The electrical/I&C non-metallic materials are also evaluated with the cable and connector materials in this AMG. The DOE Cable AMG evaluated the stressors acting on cable and connection components, industry data on aging and failures of these components, and the maintenance activities performed on cable systems. Also evaluated were the main subsystem within cables, including the conductors, insulation, shielding, tape wraps, and jacketing, as well as all subcomponents associated with each type of connection.

The applicant also identified, evaluated, and correlated the principal aging mechanisms and anticipated effects resulting from environmental and operating stresses with plant experience to determine whether the predicted effects are consistent with field experience. As such, the information, evaluations, and conclusions contained in the DOE Cable AMG are used for the evaluation of aging effects.

The most significant and observed aging mechanisms for insulated cable and connections are listed in the DOE Cable AMG, Table 4-18. The applicant used the aging mechanisms from that table as the starting point for identifying aging effects for insulated cables and connections. The applicant presents the potential aging effects along with the applicable stressors that are evaluated for insulated cables and connections in Table 3.7-1 of the LRA.

In its response to the NRC letter dated March 30, 2001, the applicant also states that in order to provide reasonable assurance that the intended functions of non-environmentally qualified (EQ) cables, connections, and penetration exposed to postulated adverse localized equipment environments caused by heat or radiation will be maintained consistent with the CLB throughout the period of extended operation, the applicant proposes an AMP for non-EQ cables, connections, and penetration in the containment at Turkey Point. The non-EQ cables, connections, and penetrations managed by this program include those used for power and instrumentation and control that are within the scope of license renewal.

The applicant stated that this program is an acceptable aging management program for non-EQ cables, connections and penetration within the scope of license renewal exposed to adverse localized equipment environments due to heat and radiation in the Turkey Point Containment. This program will be added to the Turkey Point LRA.

3.7.1.1.1 Low-Voltage Metal Connector Contact Surfaces — Moisture and Oxygen

The applicant stated that the DOE AMG states that 3% of all low-voltage metal connector failures were identified as being caused by moisture intrusion. In each case, the source of moisture was precipitation. Based on the total number of reported connector failures in the DOE Cable AMG, moisture intrusion accounted for only 10 failures in all of the operating plants in the United States.

The applicant indicates structures where electrical/I&C components may be exposed to moisture in Table 3.7-2 of the LRA. The potential moisture sources from Table 3.7-2 that are applicable to connectors at Turkey Point are precipitation and potential boric acid leaks. The applicant also indicated that all metal connectors are located in enclosures or protected from the environment with Raychem splices. Thus, aging related to moisture and oxygen do not require an AMP for low-voltage connectors at Turkey Point. The applicant also noted that electrical enclosures are treated as structural components and are discussed with each structure, as applicable, in Section 3.6 of the LRA.

3.7.1.1.2 Low-Voltage Metal Compression Fittings – Vibration and Tensile Stress

The applicant states that the aging mechanism of mechanical stress will not result in aging effects requiring AMP for the following reasons:

- Damage to cables during installation at Turkey Point is unlikely due to standard installation practices, which include limitations on cable pulling tension and bend radius. Even though installation damage is unlikely, most (including all safety-related) cables are tested after installation and before operation. Failures induced by installation damage generally occur within a short time after the damaged cable is energized.

- NRC resolution of License Renewal Issue No. 98-0013, which states, "Based on the above evaluation, the staff concludes that the issue of degradation induced by human activities need not be considered as a separate aging effect and should be excluded from an AMR."

- Mechanical stress due to forces associated with electrical faults is mitigated by the fast action of circuit protective devices at high currents. However, mechanical stress due to electrical faults is not considered an aging mechanism since such faults are infrequent and random in nature.

- Vibration is generally induced in cables and connections by the operation of external equipment, such as compressors, fans, and pumps. Vibration can affect cable connections at a running motor by producing fatigue damage of the metallic cable or termination components in the immediate vicinity of the connection point. Normally, there has to be some physical damage as well to have an effect (e.g., a nicked connector). Terminations at equipment are part of the equipment and are inspected and maintained along with the equipment. These terminations are not within the evaluation boundary for insulated cable and connections and are not included in the insulated cable and connection review.

- Manipulation of cables is not considered an aging mechanism since such manipulation occurs during maintenance activities. Such activities require post-maintenance testing to detect any deficiency in the cables. Any evidence of cable abnormalities would result in condition being addressed under the corrective program.

3.7.1.1.3 Medium-Voltage Cable and Connections and Electrical/I&C Penetration Insulation – Moisture and Voltage Stress

The applicant stated that electrical/I&C penetrations are not located in structures exposed to outside ambient conditions and therefore, not subject to moisture.

In Table 3.7-2 of the LRA, the applicant indicates structures where electrical/I&C cable and connectors may be exposed to moisture. The effects of moisture-produced water trees on medium-voltage cable were examined in Section 4.1.2.5 of the DOE Cable AMG. Water trees occur when the insulating materials are exposed to long-term, continuous electrical stress and moisture. These trees eventually result in breakdown of the dielectric materials and ultimate failure. The growth and propagation of water trees is somewhat unpredictable and few occurrences have been noted for cables operated below 15 kV. Water treeing is a long-term degradation and failure phenomenon that is documented only for medium-voltage electrical cable with cross-linked polyethylene (XLPE) or high molecular weight polyethylene (HMWPE) insulation. However, some cables are located in structures exposed to outside ambient conditions and are evaluated for the potential of moisture-produced water trees.

The applicant also indicates that Turkey Point Unit 3 and 4 medium-voltage applications, defined as 2 kV to 15 kV, use lead sheath cable to prevent effects of moisture on the cables. In addition, the applicant indicated Turkey Point does not use XLPE or HMWPE insulated cable in medium-voltage applications. Therefore, aging effects related to cable exposed to moisture and voltage stress do not require AMP at Turkey Point.

3.7.1.1.4 Medium-and Low-Voltage Cable and Connections and Electrical/I&C Penetration Insulation — Radiation and Oxygen

The applicant states that DOE Cable AMG, Section 4.1.4, Table 4-7, provides a threshold value and a moderate dose for various insulating materials. The threshold value is the amount of radiation that causes incipient to mild insulation damage. Once this threshold is exceeded, damage to the insulation increases from mild to moderate to severe as the total dose increases. The moderate damage value indicates the value at which the insulating material has been damaged but is still functional. Turkey Point evaluations use the moderate damage dose from the DOE Cable AMG as the limiting radiation value. The maximum operating dose and moderate damage dose of insulation material is shown in Table 3.7-3 of the LRA. The maximum operating dose shown in Table 3.7-3 includes the maximum 60-year normal exposure for inside containment.

The applicant compares the maximum operating dose and the moderate damage doses in Table 3.7-3 and indicates that all of the insulation materials included in this AMR will not exceed the moderate damage doses and concludes that aging effects caused by radiation exposure will not adversely affect the intended function of insulated cables and connections and electrical/I&C penetration during the extended period of operation. Therefore, the applicant concludes that aging effects related to radiation do not require an AMP for cables and connections and electrical/I&C penetrations.

3.7.1.1.5 Medium- and Low-Voltage Cable and Connections and Electrical/I&C Penetration Insulation – Heat and Oxygen

The applicant states that a maximum operating temperature was developed for each insulation type based on cable application at Turkey Point, Units 3 and 4. The maximum operating temperature indicated in Table 3.7-4 in the LRA incorporates a value for self-heating for power applications combined with the maximum design ambient temperature.

The applicant used Arrhenius method, as described in EPRI NP-1558, "A Review of Equipment Aging Theory and Technology," to determine the maximum continuous temperature to which the insulation material can be exposed so that the material has an indicated "endpoint of 60 years." These limiting temperatures for 60 years of service are provided in Table 3.7-4.

The applicant then compared the maximum operating temperature to the maximum 60-year continuous use temperature for the various insulation materials and indicated that except for polyethylene (PE) and Butyl used in power application, all of the insulation materials used in low- and medium-voltage power cables and connections can withstand the maximum operating temperature for at least 60 years.

For PE and Butyl cable insulation, the applicant states that the maximum operating temperatures, including self-heating, for PE and Butyl are 138.7 °F and 132.6 °F, respectively. The maximum temperatures for a 60-year life are 131 °F for PE and 125.1 °F for Butyl, which are 7.7 °F and 7.5 °F, respectively, less than the maximum operating temperature. The applicant states that the difference is small and is considered to be within the conservatism incorporated in the maximum operating temperatures and the maximum 60-year continuous use temperature.

The applicant also states that Butyl and PE insulated cables and connections are not used in containment and are not subject to an accident environment. Therefore, the endpoints chosen for this aging management review are extremely conservative and the 60-year endpoint values can be reduced without a loss of function, thus resulting in higher maximum 60-year continuous use temperatures.

The applicant states that maximum operating temperature in the application Table 3.7-4 includes a calculated self-heating temperature rise that assumes normal operation 100% of the time since receipt of the original operating licenses. In addition, the actual daily and seasonal temperature vary from 30 °F to 95 °F, which is less than the 104 °F limit assumed in the calculation of 60-year lifetime for Butyl and PE. The Turkey Point units have historically operated less than the 90% of the time since receipt of the original operating licenses. This amount of shutdown time lessens the amount of aging actually occurring and thus extends the lives of the materials.

Given these conservatisms, the applicant states that there is reasonable assurance that PE and Butyl insulated cables will not thermally age to the point at which they will not be able to perform their intended function during the period of extended operation. The applicant states that aging effects related to heat and oxygen do not require management for cables and electrical/I&C penetration included in the aging management review.

3.7.1.2 Uninsulated Ground Conductors

The applicant states that the ground cable material used at Turkey Point, Units 3 and 4, is copper. Copper is a good choice for this application because of its high electrical conductivity, high fusing temperature, and high corrosion resistance. Copper is also relatively strong, and it is easy to join by welding, compression, or clamping. Ground connections are commonly made with welds or mechanical type connectors, which include compression-, bolted-, and wedge-type devices.

The applicant also states that a review of available technical information regarding material aging revealed that there are no aging effects requiring management for copper grounding materials. In addition, a review of industry and plant operating experiences did not identify any failures of copper grounding systems due to aging effects. Also, several underground portions of the Turkey Point grounding system were inspected during plant modification to add two additional emergency diesel generators in 1990 and 1991 and no aging-related effects were identified. The system was approximately 20 years old at the time of that inspection. The portion of the grounding system inspected is buried in the same type of soil as other underground portions of the grounding system. Therefore, based on industry and plant-specific experiences, no aging effects requiring management were identified for the plant grounding system.

The applicant also reviewed industry and plant operating experience to ensure that no unique aging effects exist beyond those discussed in Section 3.7 for cables and connections.

3.7.2 Staff Evaluation

The staff evaluated the information on aging management presented in the LRA, Section 3.7 and in the applicant's response to the staff RAIs, dated March 30, May 11, and May 29, 2001, to determine if there is a reasonable assurance that the applicant has demonstrated that the effects of aging will be adequately managed, consistent with its CLB throughout the period of extended operation, in accordance with 10 CFR 54.21(a)(3).

3.7.2.1 Aging Management Program

The staff evaluation of the applicant's AMPs focused on the program element rather than details of specific plant procedures. The staff's approach to evaluating each program and activity used to manage the applicable aging effects is described in Section 3.1 of this SER.

[Program Scope] The scope of inspection includes accessible non-EQ cables, connections and penetrations within the scope of license renewal in the containment structures at Turkey Point that are installed in adverse localized environments caused by heat or radiation in the presence of oxygen. The staff found the scope of the program acceptable because it includes cables, connections and penetrations that are subject to potentially adverse localized environments that can result in applicable aging effects on these insulated cables, connections and penetrations.

[Preventive/Mitigative Actions] There are no preventive or mitigative actions taken as part of this program, and the staff did not identify the need for such actions.

[Parameter Inspected/Monitored] Accessible non-EQ cables, connections and penetrations within the scope of license renewal in the containment structures installed in adverse localized environments are visually inspected for cable and connection jacket surface anomalies such as embrittlement, discoloration, of cracking of surfaces. The staff found this approach acceptable because it provides means for monitoring the applicable aging effects for accessible in-scope cables, connections, and penetrations.

[Detection of Aging Effects] Cable and connection jacket surface anomalies are precursor indication of conductor insulation aging degradation from heat or radiation in the presence of oxygen and may indicate existence of an adverse localized equipment environment. An adverse localized environment is a condition in a limited plant area that is significant more severe than the specified service condition for the electrical cable, connection, or penetration. Accessible non-EQ cables, connections, or penetrations that are within the scope of license renewal in the containment structures installed in adverse localized environment are visually inspected at least every 10 years, which is an adequate period to preclude failures of the conductor insulation. The first inspection will be performed before the end of the initial 40-year licence term. EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments," will be used as guidance in performing inspections. The staff found the inspection scope and inspection technique for accessible non-EQ cables, connections, and penetrations acceptable on the basis that the AMP is focused on detecting change in material properties of the conductor insulation, which is the applicable aging effect when cables and connections are exposed to an adverse, localized environment.

[Monitoring and Trending] Trending actions are not included as part of this program because the ability to trend inspection results is limited. The staff found the absence of a trending program acceptable.

[Acceptance Criteria] No unacceptable visual indications of cables and connection jacket surface anomalies, which suggest that conductor insulation degradation exists, as determined by engineering evaluation. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function. The staff found these acceptance criteria acceptable because it should ensure that the cables and connections intended functions are maintained under all CLB design condition during the period of extended operation.

[Corrective Actions] Further investigation is performed through the corrective action program on non-EQ cables, connections and penetrations when the acceptance criteria are not met in order to ensure that the intended function will be maintained consistent with the current licensing basis. Corrective action may include, but are not limited to, testing, shielding or

otherwise changing the environment, relocation or replacement of the affected cable, connection, or penetration. Corrective actions implemented as part of the corrective action program are performed in accordance with FPL's 10 CFR Part 50, Appendix B, Quality Assurance Program. As indicated above, if an unacceptable condition or situation is identified for a cable, connections, penetration in the inspection, the applicant will perform further investigation through the corrective action program. However, the applicant did not specifically include a determination of whether the same condition or situation is applicable to other accessible or inaccessible cables, connections and penetration. In this regard, the staff requested the applicant address the aging management associated with inaccessible cables, connections, or penetrations. In response to the staff's request, the applicant specifically requires that when an adverse localized environment is identified for a cable, connection, or penetration, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables, connections, or penetrations. The staff found the applicant response acceptable because selected cables, connections, and penetrations from accessible areas (the inspection sample) are inspected and represent, with reasonable assurance, all cables, connections, and penetrations in the adverse localized environments. It also found that as discussed in Section 3.1.2 of this report, the requirement of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.

[Confirmation Process] The confirmation process implemented as part of the corrective program is performed in accordance with FPL's 10 CFR Part 50, Appendix B, Quality Assurance Program. As discussed in Section 3.1.2 of this report, the staff found the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.

[Administrative Controls] Administrative controls associated with this program will be performed in accordance with FPL's 10 CFR Part 50, Appendix B, Quality Assurance Program. The staff found the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.

[Operating Experience] The licensee did not identify the presence of adverse localized heat and radiation environments in the containment at Turkey Point. However, operating experience identified by the staff in the draft GALL Report has shown that adverse localized environments caused by heat or radiation for electrical cables and connections may exist next to or above (within three feet of) steam generators, pressurizers or hot process pipes such as feedwater lines. The staff found that the proposed inspection program will detect the adverse localized environment caused by heat or radiation of electrical cables and connections.

3.7.2.2 Non-Environmentally Qualified Insulated Cables and connections, and Electrical/I&C Penetrations

The NRC staff has evaluated the information presented in Sections 3.7.1.1.1, 3.7.1.1.2, 3.7.1.1.3, 3.7.1.1.4, 3.7.1.1.5, and 3.7.1.2 of the LRA to determine if there is a reasonable assurance that the applicant has identified the applicable aging effects and the bounding conditions for electrical/I&C components. The process to determine the applicable effect on these components is based on industry literature defining the operating environment and operating stresses for each of the components that are subject to an AMR. The NRC staff reviewed each of the environments and resulting mechanisms and effects as they apply to the electrical/I&C component commodities.

3.7.2.2.1 Low-Voltage Metal Connector Contact Surfaces — Moisture and Oxygen

3.7.2.2.1.1 Effects of Aging

The potential aging mechanisms considered for low-voltage metal connector surfaces is corrosion due to moisture intrusion. Structures where electrical/I&C components may be exposed to moisture are indicated in the LRA Table 3.7-2. The potential moisture sources from this table that are applicable to connectors at Turkey Point are precipitation and potential boric acid leaks. All metal connectors at Turkey Point are located in enclosures or protected from the environment with Raychem splices. Thus, aging effects related to moisture and oxygen do not require management for low-voltage connectors at Turkey Point.

3.7.2.2.1.2 Aging Management Program

The NRC staff has evaluated the information on low-voltage metal connectors as presented in Section 3.7.1.1.1 of the LRA to determine if there is a reasonable assurance that the applicant has demonstrated that the aging affects for low-voltage connectors will be adequately managed, consistent with the applicant's CLB throughout the period of extended operation.

The staff agrees with the applicant's assessment and the conclusion that low-voltage connectors are located in enclosure or protected from the environment with Raychem splices and aging effects related to moisture and oxygen do not require an AMP for low-voltage connectors at Turkey Point.

3.7.2.2.2 Low-Voltage Metal Compression Fitting - Vibration and Tensile Stress

3.7.2.2.2.1 Effects of Aging

The aging mechanism of mechanical stress will not result in effects requiring management for the following reasons (1) damage to cables during installation at Turkey Point is unlikely due to standard installation practice, which include limitation on cable pulling tension and bend radius. (2) NRC resolution of License Renewal Issue No. 98-0013 states that the issue of degradation induced by human activities need not be considered as a separate aging affect and should be excluded from an AMR. (3) Mechanical stress due to forces associated with electrical faults is mitigated by the fast action of circuit protective devices at high currents. However, the mechanical stress due to electrical faults is not considered an aging mechanism since such faults are infrequent and random in nature. (4) Vibration is generally induced in cables and connections by the operation of external equipment, such as compressor, fans, and pumps. Vibration can affect cable connections at a running motor by producing fatigue damage of the metallic cable or termination components in the immediate vicinity of the connection point. Normally, there has to be some physical damage as well to have an effect (e.g., a nicked connector). Terminations at equipment are part of the equipment and are inspected and maintained along with the equipment. These terminations are not within the evaluation boundary for insulated cable and connections and are not included in the insulated cable and connection review. (5) Manipulation of cables is not considered an aging mechanism since such manipulation occurs during maintenance activities. Such activities require post-maintenance testing to detect any deficiencies in the cables. Any evidence of cable abnormalities would result in the condition being addressed under the corrective action program.

3.7.2.2.2.2 Aging Management Programs

The staff has evaluated the information on low-voltage metal compression fittings as presented in Section 3.7.1.1.2 of the LRA to determine if there is a reasonable assurance that the applicant has demonstrated that the aging affects for low-voltage metal compression fittings will be adequately managed, consistent with the applicant's CLB throughout the period of extended operation.

The staff agrees with the applicant's assessment and conclusion that aging mechanism of mechanical stress will not result in aging effects requiring an AMP for low-voltage metal compression fittings.

3.7.2.2.3 Medium-Voltage Cable and Connections and Electrical/I&C Penetration Insulation — Moisture and Voltage Stress

3.7.2.2.3.1 Effects of Aging

Electrical/I&C penetrations are not located in structures exposed to outside ambient conditions and, therefore, not subject to moisture.

Structures where electrical/I&C cable and connectors may be exposed to moisture are indicated in Table 3.7-2. Water trees occur when the insulating materials are exposed to long-term, continuous electrical stress and moisture. These trees eventually result in breakdown of the dielectric materials and ultimately failure. The growth and propagation of water trees is somewhat unpredictable, and occurrences have been noted for cable operated below 15 kV. Water treeing is a long-term degradation and failure phenomenon that is documented only for medium-voltage electrical cable with conductor insulation made of various organic polymers (e.g., EPR, SR, EPDM, XLPE).

3.7.2.2.3.2 Aging Management Program

The staff has evaluated the information on medium-voltage cable and connections and electrical/I&C penetration insulation as presented in Section 3.7.1.1.3 of the LRA to determine if there is a reasonable assurance that the applicant has demonstrated that the aging affects for medium-voltage cable, connections, and electrical/I&C penetration insulation will be adequately managed, consistent with the applicant's CLB throughout the period of extended operation.

The applicant states that Turkey Point Unit 3 and 4 medium-voltage applications use lead sheath to prevent effects of moisture on the cables. In addition, the applicant states that Turkey Point does not use XLPE or HMWPE insulated cables in medium-voltage applications and, therefore, the aging effects related to cable exposed to moisture and voltage stress do not require an AMP at Turkey Point.

Most electrical cables in nuclear power plant are located in dry environments. However, some cables may be exposed to condensation and wetting in some locations (such as conduits, cable trenches, cable troughs, duct banks, underground vaults, or direct buried installations). When an energized cable that is not specifically designed for submergence is exposed to these conditions, water treeing or a decrease in the dielectric strength of the conductor insulation can occur. This can potentially lead to electrical failure. The purpose of the aging management program is to provide a reasonable assurance that the intended functions of medium-voltage cables exposed to adverse localized environments caused by moisture while energized will be maintained consistent with the current licensing basis through the period of operation. It was not clear to the staff that the lead sheath would prevent moisture ingress if the cable was subjected to significant moisture simultaneously with significant voltage. Water treeing is a long-term degradation and failure phenomenon with conductor insulation made of various organic polymers. In the letter to the applicant dated January 17, 2001 (RAI Number 3.7.1-1), the staff requested the applicant to provide an aging management program for accessible and inaccessible electrical cable operated below 15 kV exposed to an adverse localized environment caused by moisture-produced water trees.

In response to the staff's request, the applicant stated that Turkey Point medium voltage application use lead sheath cable to prevent effects of moisture on cables. This cable is designed with a thick layer of lead over the cable insulation with an overall jacket over the lead and insulation. This differs from the typical medium voltage cable design of insulation with an overall jacket. The applicant uses lead sheath cable as a standard for medium voltage applications because of its good characteristics in moisture environments. The applicant's cable specification states that lead sheath cables are designed to be installed in wet environments for extended periods. In addition, the cable manufacturer's specification for lead sheath cable states that "...EPR/lead sheath cable is designed for application in which liquid contamination is present and reliability is paramount. The sheath combined with the overall jacket provided a virtually impenetrable barrier against hostile environments — liquids, fire hydrocarbons, acids, caustic, sewage, etc." As an added level of protection, Turkey Point underground medium-voltage cables are only routed in concrete-encased duct banks. Industry experience shows no failures of the medium-voltage lead sheath cable under various environments, including moisture. In addition, the applicant performed an extensive review of Turkey Point's plant-specific operating experience and found no cases of medium-voltage cable failures due to adverse localized environments.

Based on the review of the LRA and the applicant's response to the staff's RAIs, the staff concludes that since the applicant uses lead sheath in medium voltage cables, an aging management program for accessible and inaccessible medium-voltage cable to address adverse localized environments caused by moisture-produced water trees and voltage stress is not required.

3.7.2.2.4 Medium- and Low-Voltage Cable and Connections and Electrical/I&C Penetration Insulation — Radiation and Oxygen

3.7.2.2.4.1 Effects of Aging

Radiation-induced degradation in cable jacket and insulated material produces change in organic material properties, including reduced elongation and changes in tensile strength. Visible of indication of radiative aging may include embrittlement, cracking, discoloration, and swelling of the jacket and insulation. Table 3.7-3 of the LRA lists both the maximum operating doses and the moderate damage doses.

3.7.2.2.4.2 Aging Management Program

The staff has evaluated the information on medium-voltage cable and connections/I&C penetration insulation as presented in Section 3.7.1.1.4 of the LRA to determine if there is a reasonable assurance that the applicant has demonstrated that the aging affects for medium- and low-voltage cable and connections and electrical/I&C penetration insulation will be adequately managed, consistent with the applicant's CLB throughout the period of extended operation.

The applicant compares the maximum operating dose and the moderate damage doses in Table 3.7-3 and shows that all of the insulation materials included in this AMR will not exceed the moderate damage doses and concludes that aging effects caused by radiation exposure will not adversely affect the intended function of insulated cables and connections and electrical/I&C penetrations during the extended period of operation. The applicant concludes that aging effects related to radiation do not require an AMP for cables and connections and electrical/I&C penetrations.

Conductor insulation material used in cables and connections and electrical/I&C penetrations may degrade more rapidly than expected in the adverse localized environments due to radiation. The radiation levels most equipment experience during normal service have little degrading effect on most materials. The evaluations or calculations that determine or bound the expected radiation doses usually account for doses seen in all plant areas. However, some localized areas may experience higher-than-expected radiation conditions. Typical areas prone to elevated radiation levels include areas near primary reactor-coolant-system piping or the reactor-pressure-vessel; areas near waste processing system and equipment (e.g., gaseous-waste system, reactor-purification system, reactor-water-cleanup system, and spent-fuel-pool cooling and cleanup system); and areas subject to radiation streaming. The applicant's conclusion is not consistent with the aging management program and activities for electrical cables and connections exposed to localized environments caused by radiation as described in the previous LRAs that have been approved by the staff. In a letter to the applicant dated January 17, 2001 (RAI Number 3.7.1-1), the staff requested the applicant to provide an aging management program for accessible and inaccessible electrical cable and connections and electrical/I&C penetrations exposed to an adverse localized environment caused by radiation.

In response to the staff's request, in a letter dated March 30, 2001, the applicant stated that the intake structure, main steam and feedwater platforms, and yard structures are outdoor areas where cable, connections, and penetrations are not subject to adverse localized radiation effects. The turbine building is an outdoor area with no external walls or roof. The only buildings with any appreciable radiation levels are the containment and the auxiliary buildings. In order to provide reasonable assurance that the intended functions of non-EQ cables, connection, and penetration exposed to adverse localized equipment environments cause by radiation will be maintained consistent with the current licensing basis through the period of extended operation, the applicant proposes an aging management program for non-EQ cables, connections, and penetrations in the containment at Turkey Point. The non-EQ cables, connections and penetrations managed by this program include those used for power and instrumentation and control that are within the scope of license renewal. The acceptability of the AMP is evaluated in section 3.7.2.1 of the staff's safety evaluation report.

As indicated above, the applicant states that the only buildings with any appreciable radiation levels are the containment and auxiliary buildings. However, the aging management program that the applicant proposed is only include the non-EQ cables, connections, and penetrations in the containment. It does not include those cables, connections and penetrations in the auxiliary buildings. In a telephone conference, the staff requested the applicant explain why auxiliary building was not included in the scope of electrical inspection program. In response to the staff request, the applicant modified its response to the staff's RAI to state that with regard to radiation, the only buildings with any appreciable radiation levels are the containment building and the auxiliary building. However, non-EQ cables, connections, and penetration in the auxiliary building are not located in the areas which would be subject to adverse localized radiation environments during plant operation. The staff finds the applicant's response to be acceptable because it is consistent with the scope of the proposed aging management program.

Based on the review of the LRA and the applicant's response to the staff's RAIs, the staff concludes that aging effects of radiation on medium- and low-voltage cables, connections, and electrical/I&C penetrations will be managed through an AMP. This program will provide reasonable assurance that the intended functions of electrical cables, connections and electrical/I&C penetration exposed to adverse localized environments caused by radiation will be maintained consistent with the CLB through the period of extended operation.

3.7.2.2.5 Medium- and Low-Voltage Cable and Connections and Electrical/I&C Penetration Insulation — Heat and Oxygen

3.7.2.2.5.1 Effects of Aging

Thermal-induced degradation in cable jacket and insulation materials can result in reduced elongation and changes in tensile strength. Visible indications of thermal aging may include embrittlement, cracking, discoloration, and swelling of the jacket and insulation. Arrhenius methodology with the time period fixed at 60 years was used by the applicant to determine the maximum continuous temperature to which the material can be exposed so the material will not have reached the endpoint at the end of 60 years.

3.7.2.2.5.2 Aging Management Program

The staff has evaluated the information as presented in Section 3.7.1.1.5 of the LRA to determine if there is a reasonable assurance that the applicant has demonstrated that the aging effects for medium- and low-voltage cables, connections, and electrical/I&C penetration insulation will be adequately managed, consistent with the applicant's CLB, for the period of extended operation.

The applicant uses the Arrhenius method to determine the maximum continuous temperature to which the insulation material can be exposed so that the material has an indicated "endpoint of 60 years." It then compares the maximum 60-year continuous use temperature to the maximum operating temperature for the various insulation materials. The applicant concludes that except for polyethylene (PE) and Butyl used in power application, all of the insulation materials used in low- and medium-voltage power cables and connections and electrical/I&C penetration insulation can withstand the maximum operating temperatures for at least 60 years. For Butyl and PE insulated cables and connections, the applicant states that Butyl and PE are not used in containment and are not subject to an accident environment. Therefore, the endpoint chosen for this AMR are extremely conservative, and the 60-year endpoints values can be reduced without a loss of function, thus resulting in higher maximum 60-year continuous use temperature. The applicant concludes that aging effects related to heat do not require management for cables, connections, and electrical/I&C penetrations included in the AMR.

The most common adverse localized equipment conditions are those created by elevated temperature. Elevated temperature can cause equipment to age prematurely, particularly equipment containing organic materials and lubricants. The effect of elevated temperature can be quite dramatic. The types of areas that are prone to high temperature include areas with high-temperature process fluid piping and vessels, areas with equipment that operates at high temperature, and areas with limited ventilation. The staff did not agree with the applicant that the Arrhenius method can be used to extend the qualified life of insulation material that is exposed to elevated temperature to 60 years. In a letter dated January 17, 2001, the staff requested that the applicant provide an aging management program for accessible and inaccessible electrical cables and connections and electrical/I&C penetrations exposed to adverse localized environments caused by heat. In response to the staff's request, in a letter dated March 30, 2001, the applicant stated that the intake structure, main steam and feedwater platforms, and yard structures are outdoor areas where cable, connections, and penetrations are not subject to adverse localized temperature effects. The turbine building is an outdoor area with no external wall or roof. The auxiliary building does not contain any high temperature reactor coolant, main steam, or feedwater and blowdown system piping and components. In order to provide reasonable assurance that the intended functions of non-EQ cables, connections, and penetrations exposed to adverse localized equipment environments caused by heat will be maintained consistent with the CLB basis through the period of extended operation, the applicant proposed an aging management program for non-EQ cables, connections, and penetrations in the containment at Turkey Point. The non-EQ cables, connections, and penetrations managed by this program include those used for power and instrumentation and control that are within the scope of license renewal. The acceptability of this AMP is evaluated in section 3.7.2.1 of the staff's safety evaluation report.

Based on the review of the LRA and the applicant's response to the staff's RAIs, the staff concludes that aging effects of heat on medium-and low-voltage cable and connections and electrical/I&C penetrations should be managed through the AMP. This program will provide reasonable assurance that the intended functions of electrical cables and connections and electrical/I&C penetration exposed to adverse localized environments caused by heat will be maintained consistent with the current licensing basis through the period of extended operation.

3.7.2.3 Uninsulated Ground Conductors

The ground cable material used at Turkey Point, Units 3 and 4, is copper. Copper is a good choice for this application because of its high electrical conductivity, high fusing temperature, and high corrosion resistance. Copper is also relatively strong, and it is easy to join by welding, compression, or clamping. Ground connections are commonly made with welds or mechanical type connectors, which include compression-, bolted-, and wedge-type devices.

The applicant has reviewed the available industry technical information regarding material aging and has determined that there are no aging effects requiring management for copper grounding materials. In addition, the applicant has reviewed of industry and plant operating experience and did not identify any failures of copper ground system due to aging affects. The applicant also inspected several underground portions of the Turkey Point grounding system during plant modification to add two additional emergency diesel generators in 1990 and 1991, and did not identify any aging-related effects. The system was approximately 20 years old at the time of that inspection. The applicant states that portion of the grounding system inspected is buried in the same type of soil as other underground portions of the grounding system. Therefore, based on industry and plant-specific experience, no aging affects requiring management were identified for the plant grounding system. The staff agrees with the applicant's assessment and conclusion that no AMP is required for the plant ground system.

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 UFSAR Supplement. By letter dated November 1, 2001, the applicant provided summary description of the programs in Appendix A, Chapter 16, section 16.1.8, "Containment Cable Inspection Program," of the UFSAR Supplement. The summary description is sufficient, and therefore, confirmatory item 3.0-1 FSAR item 3.7-1 is closed.

3.7.4 Conclusion

On the basis of the staff's evaluation described above, the staff finds that there is reasonable assurance that the effects of aging of cables, connections, and electrical/I&C penetrations at Turkey Point will be adequately managed so that the intended function will be maintained consistent with the applicant's CLB throughout the period of extended operation in accordance with the requirements of 10 CFR 54.21(a)(3).

3.8 New Aging Management Programs

3.8.1 Auxiliary Feedwater Pump Oil Coolers Inspection Program

3.8.1.1 Summary of Technical Information in the Application

The auxiliary feedwater system supplies feedwater to the steam generators when normal feedwater sources are not available. It provides for auxiliary feedwater steam and feedwater isolation during a postulated steam generator tube rupture event and, for auxiliary feedwater isolation to the faulted steam generator. The auxiliary feedwater system also limits feedwater flow to the steam generators to limit positive reactivity insertion during a postulated steam line break. The auxiliary feedwater system contains three steam turbine-driven pumps. Table 3.5.3 of the LRA indicates that the auxiliary feedwater pumps oil coolers inspection and chemistry control programs are credited for the aging management of the auxiliary feedwater pump oil coolers for the pumps in the auxiliary feedwater and condensate storage systems. The condensate storage tank stores water for use by the auxiliary feedwater system to support safe shutdown of the plant. The intended functions for the auxiliary feedwater and condensate storage components subject to an aging management review are pressure boundary integrity, heat transfer, and throttling.

3.8.1.2 Staff Evaluation

In Table 3.5-3 of the LRA, the applicant identified loss of material to be the aging effect requiring management for the cast steel auxiliary feedwater pump lube oil coolers channel and covers exposed to treated water. The lube oil coolers are tube and shell type heat-exchangers, with the lube oil flowing in the tubes and the feedwater on the shell side. The purpose of the coolers is to transfer heat from the lube oil to the feedwater and maintain the lube oil temperature to within acceptable limits. The applicant credited the auxiliary feedwater pump oil coolers inspection program for the aging management of the identified aging effect. This program is described in Appendix B, Section 3.1.1, of the LRA.

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

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

[Program Scope] The applicant stated that this inspection is intended to be a one-time inspection of an oil cooler of one of the three shared auxiliary feedwater pumps. The auxiliary feedwater pump oil coolers inspection will manage the effects of loss of material due to graphitic corrosion (i.e., selective leaching) and other types of corrosion of the internal surfaces of cast iron parts of the coolers wetted internally by treated secondary water. A visual inspection will be performed to detect loss of material. The inspection will include the cast iron bonnet of one of the auxiliary feedwater pump lube oil coolers and, if necessary, the cast iron parts of an auxiliary feedwater turbine governor controller oil cooler. Commitment dates associated with the implementation of this new program are contained in Appendix A to the LRA.

In RAI 3.8.1-1, dated February 2, 2001, the staff requested that the applicant provide justification for only inspecting the oil cooler of one of the three pumps, and for doing a one-time-only inspection instead of multiple inspections with intervals of 3 or 5 years, as is generally prescribed in ASME Section XI programs for similar components.

In its response (FPL Letter L-2001-75, dated April 19, 2001), the applicant stated that the three auxiliary feedwater pump oil coolers are identical units and are subjected to the same internal environments and operating conditions. Therefore, the condition of one cooler is representative of all three coolers.

The applicant stated that the one-time inspection will provide confirmatory information on the condition of the coolers. Although Turkey Point's operating experience has not identified graphitic corrosion degradation of these coolers, the materials of construction and environment make them potentially susceptible to such degradation. This corrosion mechanism is not anticipated due to the quality of the water in the auxiliary feedwater system, and thus, a one-time inspection was selected. The results of the inspection will be evaluated to determine if further inspections are warranted. If significant loss of material is detected, the appropriate corrective action, including program revision, if needed, will be taken in accordance with the applicant's 10 CFR Part 50 Appendix B, Corrective Action Program. The staff finds the applicant's response reasonable and acceptable, and on this basis the issue in RAI item 3.8.1-1 is resolved. With the resolution of the staff's concerns, the staff finds the overall scope of the Auxiliary Feedwater Pump Oil Coolers Inspection Program is acceptable.

[Preventive Actions] The applicant stated that no preventive actions are applicable to this program. The staff finds this acceptable because the staff does not find a need for any.

[Parameters Monitored] The applicant stated that the auxiliary feedwater pump coolers inspection will identify the presence of graphitic corrosion activity and will quantify the loss of structurally sound wall thickness of cast iron parts. The inspection will consist of two parts, an "as found" inspection of parts and an inspection of parts after light sandblasting to bare metal. The staff finds the program is acceptable.

[Detection of Aging Effects] The applicant stated that the visual inspection will be used to verify whether graphitic corrosion has taken place. The aging effect of concern, loss of material because of graphitic corrosion and other types of corrosion, will be further evident by the reduced wall thickness of the material in the cast iron parts being examined (following the sandblasting). The staff concurs with the applicant and finds the detection methods acceptable.

[Monitoring and Trending] As stated above, the applicant intends to do a one-time inspection of one cooler. If significant loss of material due to graphitic corrosion or other corrosion is detected, the applicant will assess the extent of the corrosion, and determine if inspection of other coolers and additional future monitoring are required. The staff finds the applicant's monitoring and the trending method acceptable.

[Acceptance Criteria] The applicant states that if the inspection results in white non-porous metallic surface without major indications, it may declare the part as "not affected by graphitic corrosion" and to not require further evaluation. If there is evidence of significant effects of graphitic corrosion, an evaluation will be prepared to establish the minimum required wall

thickness including a corrosion allowance adequate for a pre-determined inspection interval. Wall thickness measurements greater than minimum wall thickness values will be acceptable. In RAI 3.8.1-2, dated February 2, 2001, the staff requested the applicant to provide the basis for the quantitative acceptance criteria which will be used to make the determination that inspection of other coolers and future monitoring are required.

In its response (FPL Letter L-2001-75, dated April 19, 2001), the applicant stated that the auxiliary feedwater pump oil cooler inspection program, as described in Appendix B, Section 3.1.1, page B-10, consists of a confirmatory one-time inspection of one auxiliary feedwater oil cooler to verify that loss of material due to graphitic corrosion is not occurring. In the event that significant loss of material is detected during this inspection, appropriate corrective actions will be established per FPL's 10 CFR Part 50, Appendix B, Corrective Action Program. Evaluation of inspection results will consider the minimum required wall thickness for the component and a corrosion allowance. Followup inspections, if required, will be scheduled based on actual corrosion rates or inspection findings. The staff finds the applicant's response reasonable and acceptable and, on this basis, the issue of concern in RAI 3.8.1-2 is considered resolved. The staff finds that the acceptance criteria are adequate.

[Operating Experience and Demonstration] Visual inspections and wall thickness measurements of equipment have been performed at Turkey Point for many years. The techniques have proven successful in determining actual material condition of components.

The auxiliary feedwater pump oil coolers inspection is a new program that will use techniques with demonstrated capability and a proven industry record to detect loss of material due to graphitic corrosion. Visual examination has been used in the past to identify graphitic corrosion. This inspection will be performed utilizing approved procedures and qualified personnel. The staff finds the applicant's inspection methods applicable and acceptable.

3.8.1.3 FSAR Supplements

The staff has reviewed the information in the UFSAR Supplement Section 16.1.1 of Appendix A to the LRA and has confirmed that it contains the appropriate elements of the program.

3.8.1.4 Conclusion

In conclusion, based on the information provided by the applicant, the staff finds the implementation of the auxiliary feedwater pump oil coolers inspection program will provide reasonable assurance that loss of material will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.8.2 Auxiliary Feedwater Steam Piping Inspection Program

3.8.2.1 Summary of Technical Information in the Application

The auxiliary feedwater system supplies feedwater to the steam generators when normal feedwater sources are not available. It provides for auxiliary feedwater steam and feedwater isolation during a postulated steam generator tube rupture event, for auxiliary feedwater isolation to the faulted steam generator and limits feedwater flow to the steam generators to limit positive reactivity insertion during a postulated steam line break. The auxiliary feedwater system contains three steam turbine-driven pumps. The pumps can be supplied steam from the steam generators in either unit. The pumps take suction from either condensate storage tank and discharge to one of two redundant headers. Each header can supply each of the steam generators. The auxiliary feedwater system is normally maintained in standby with one pump aligned to one discharge header and two pumps aligned to the other header. Upon initiation, all three pumps start to supply the affected steam generator with feedwater. The condensate storage tank stores water for use by the auxiliary feedwater system to support safe shutdown of the plant. The intended functions for the auxiliary feedwater and condensate storage components subject to an aging management review are pressure boundary integrity, heat transfer, and throttling.

3.8.2.2 Staff Evaluation

In Table 3.5-3 of the LRA, the applicant identified loss of material to be the aging effect requiring management for the carbon steel auxiliary feedwater pump turbine casings, valves, steam traps, and piping/fittings that are exposed to either treated water-secondary and air/gas environments or outdoor environments. The applicant credited the Auxiliary Feedwater Steam Piping Inspection Program (Appendix B Section 3.1.2 of the Application) for the aging management of the identified aging effect.

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

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

[Program Scope] The applicant stated in Appendix B Section 3.1.2 of the Application that the Auxiliary Feedwater Steam Piping Inspection Program will manage the effects of loss of material due to general and pitting corrosion on the internal and external surfaces of the auxiliary feedwater steam supply carbon steel piping and fittings. The program will provide for representative volumetric examinations to detect loss of material in the auxiliary feedwater steam piping between the steam supply check valves and each of the three auxiliary feedwater pump turbines. In its RAI dated February 2, 2001, the staff requested the applicant to provide a detailed description of how samples will be selected for the representative volumetric examinations and the basis of the selection. The staff also requested the applicant to explain why, in Table 3.5-3 of the LRA, components other than piping and fittings, such as auxiliary feedwater pump turbine casings, are listed as the in-scope components to be managed by the program. The applicant provided its response to the RAI in a submittal dated April 19, 2001 (cf. L-2001-75). In its response, the applicant stated that sample selections will be based upon the potential for exposure to a wetted environment. This includes sections of lines where water

can accumulate, such as at the bottom of horizontal pipe runs and areas of contact with the lower section of wetted insulation. The staff considers the basis of the applicant's sample selection reasonable and, therefore, acceptable. The applicant also stated that having the least wall thickness, piping and fittings are considered the limiting components and the primary inspection points. However, where significant loss of material due to corrosion is detected, valves and steam traps would be inspected, as required. This is acceptable to the staff In regard to the staff's question on inclusion of the above-mentioned turbine casings. The applicant stated that Table 3.5-3 (pages 3.5-17 and 3.5-20) of the LRA inadvertently identified internal and external loss of material as an aging effect requiring management for the auxiliary feedwater (AFW) turbine casings, and credited the auxiliary feedwater steam piping integrity program for aging management. The applicant stated that, based on an inspection of an AFW turbine casing, after 17 years of operation, the aging management review of the AFW turbine casing has demonstrated that loss of material is indeed not an aging effect requiring management. The applicant stated that Table 3.5-3 (pages 3-17 and 3-20) will be revised accordingly. The staff finds the applicant's response to be in general accord with the industry experience and is, therefore, acceptable.

[Preventive Actions] The applicant stated that no preventive actions are applicable to this program, and the staff did not identify the need for any.

[Parameters Monitored or Inspected] The applicant stated that the program will monitor the wall thickness of representative piping/fittings in the auxiliary feedwater steam supply headers and the drain lines upstream of the steam traps. The volumetric examination will identify potential effects of inside diameter corrosion due to accumulation of water at the bottom of horizontal run pipes and outside diameter corrosion at areas of contact with the lower section of wet insulation. Based on the scope of the inspection, the staff finds parameters monitored are acceptable.

[Detection of Aging Effects] The applicant stated that the aging effect of concern, loss of material due to general and pitting corrosion, will be evident by the reduced wall thickness in the piping/fittings. Based on the scope of the inspection, the staff finds the detection method is acceptable.

[Monitoring and Trending] The applicant stated that the examination will initially be performed every five years. Piping/fittings thickness measurements will permit calculation of an integrated inside diameter and outside diameter corrosion rate. Inspection frequency may be adjusted based on corrosion rate to ensure that the minimum wall thickness requirements will be maintained. Based on the scope of the inspection the staff finds, the identified inspection frequency is acceptable.

[Acceptance Criteria] The applicant stated that wall thickness measurements greater than minimum values for the component design of record will be acceptable. Wall thickness measurements less than required minimum values will be entered into the corrective action program. This will ensure that the component section identified to have potentially inadequate wall thickness will be subject to subsequent evaluations and remedy actions. It is, therefore, acceptable to the staff.

[Operating Experience] Ultrasonic and computer-aided radiography wall thickness measurement techniques have been performed at Turkey Point for years. The applicant stated that these techniques have proven successful in determining wall thickness of piping and other components. Computer-aided radiography has been used in the auxiliary feedwater steam supply headers and drain lines. The results of these examinations have detected some areas of localized corrosion in the headers. The applicant stated that this new program will use the techniques with demonstrated capability and a proven industry record to measure pipe wall thickness. The examinations will be performed utilizing approved plant procedures and qualified personnel. Based on the applicant's description of the examination techniques and the evidence of their successful performance in the past, the staff considers the examination methods to be acceptable for the program.

3.8.2.3 FSAR Supplement

Section 16.1.2, "Auxiliary Feedwater Steam Piping Inspection Program," of Appendix A to the LRA provides an updated FSAR supplement for the auxiliary feedwater steam piping inspection program. The staff concludes that the updated FSAR Supplement is sufficient.

3.8.2.4 Conclusion

Based on the information provided by the applicant, the staff finds that the implementation of the auxiliary feedwater steam piping inspection program will provide reasonable assurance that loss of material will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.8.3 Emergency Containment Coolers Inspection

3.8.3.1 Summary of Technical Information in the Application

The emergency containment coolers inspection program is designed to determine the extent of loss of material due to erosion in the emergency containment cooler tubes. This is a one-time inspection of a representative sample of tubes in the containment coolers. The results of the inspection will be evaluated to determine an actual erosion rate and projected minimum wall thickness at the end of the extended period of operation. Programmatic changes will be made on the basis of the inspection results.

Emergency containment cooling components subject to an aging management review include the emergency fan cooler units (pressure boundary only) and associated heat exchanger coils. The intended functions for emergency containment cooling components subject to an aging management review include pressure boundary integrity and heat transfer. A complete list of emergency containment cooling components requiring an aging management review and the component intended functions are provided in Table 3.3-1 of the LRA. The aging management review for emergency containment cooling is discussed in Section 3.3 of the LRA. The applicant has credited the emergency containment cooler inspection program for managing the identified aging effects. The program is described in Appendix B, Section 3.1.3 of the LRA

The analyses for the current licensing basis for emergency containment cooling tubes have used conservative erosion rates. The applicant contends that the actual wall loss is expected to be less and confirmation of the actual wall thickness degradation will be obtained through inspection.

3.8.3.2 Staff Evaluation

In Table 3.3-1 of the LRA, the applicant identified loss of material to be the aging effect requiring management for the admiralty brass emergency containment cooling tubes exposed to a treated water environment. Emergency containment cooling tube wear was identified as a TLAA as discussed in Section 4.7.2 of the LRA. Option (iii) of 10 CFR 54.21 (c) (1) was selected to address this aging effect.

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

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

[Program Scope] The applicant stated that the emergency containment cooling inspection is a one-time inspection that will determine the extent of loss of material due to erosion in the Unit 3 and 4 emergency containment cooler tubes. A sample of tubes for examination will be selected based on piping geometry and flow conditions that represent those with the greatest susceptibility to erosion. This inspection and evaluation will be implemented prior to the end of the initial operating terms.

In RAI 3.8.3-1, dated February 2, 2001, the staff requested the applicant to provide a justification for their determination that a one time inspection of the emergency containment coolers is adequate. Operating experience with these coolers at other nuclear power plants indicates that loss of material caused by erosion and flow-induced vibration can vary during plant operation due to unanticipated transients and flow conditions.

In its response dated April 19, 2001, the applicant stated that the aging effect requiring management for the emergency containment coolers is loss of material due to erosion on the inside surface of the cooler tubes. Cracking due to flow-induced vibration is not an aging effect requiring management. Except for surveillance testing, the emergency containment coolers are normally not in operation and have minimal cooling water flow through the tubes (see UFSAR Section 6.3.2, page 6.3-6). Therefore, the tubes are not susceptible to unanticipated transients. The results of the inspection will be evaluated to determine an actual erosion rate to verify that the minimum required wall thickness for the emergency containment cooling tubes will be maintained during the period of extended operation. As stated in Section 3.1.3 of Appendix B to the LRA, the evaluation of the inspection results may determine the need for additional testing, monitoring, and trending. The staff concurs with the applicant's response. RAI Item 3.8.3-1 is therefore closed.

In RAI 3.8.3-2, dated February 2, 2001, the staff requested the applicant to provide the specific percentage of tubes that will be examined during the inspection. In its response (FPL Letter L-2001-75, dated April 19, 2001), the applicant stated that a sample of tubes for examination will be selected based on geometry and flow conditions that represent those with the greatest susceptibility to erosion. All six emergency containment coolers (three in each unit) are identical and are subjected to the same cooling water conditions. Additionally, the emergency containment coolers are in service (during testing) approximately the same amount of time. On this basis, one emergency containment cooler will be selected for inspection as representative of all six. The number of tubes to be inspected in this cooler will be in accordance with the sampling plan recommended by the American Society for Quality Control publications. The staff finds the applicant's response reasonable and acceptable. On this basis, the issue in RAI item 3.8.3-2 is considered closed. With the resolution of the staff's concerns as discussed above, the staff finds the scope of the program acceptable.

[Preventive Actions] The applicant stated that no preventive actions are applicable to this inspection. The staff does not find a need for any preventive actions and therefore this is acceptable.

[Parameters Monitored or Inspected] The applicant stated that the inspection will document wall thickness of the emergency containment cooler heat exchanger tubes. The staff finds that the parameters monitored will permit timely detection of aging effects and are therefore acceptable.

[Detection of Aging Effects] The aging effect of concern, loss of material due to erosion, will be detected and sized in accordance with the volumetric technique chosen by the applicant. The staff finds the detection method will provide a satisfactory means for identifying the aging effect and is therefore acceptable.

[Monitoring and Trending] The results of the inspection will be evaluated by the applicant to verify that the minimum required wall thickness for the emergency containment cooler heat exchanger tubes will be maintained during the period of extended operation. In RAI 3.8.3-3, dated February 2, 2001, the staff requested that the applicant discuss the acceptance criteria which it will use for tube examination in the emergency containment coolers inspection program, and also clarify the source and basis for the acceptance criteria to be applied for this examination.

In its response (FPL Letter L-2001-75, dated April 19, 2001), the applicant stated that the acceptance criteria for the emergency containment cooler tubes is minimum wall thickness plus margin based upon actual erosion rate. The minimum wall thickness for the Emergency containment cooler tubes is based on the coolers' design pressure as calculated per ASME Section III, Class 3. Appendix B Section 3.1.3 of the LRA states that the results of the inspection will be evaluated to verify that the minimum required wall thickness for the Emergency containment cooler tubes will be maintained during the period of extended operation. The staff finds the applicant's response reasonable and acceptable. Therefore, the issue in RAI item 3.8.3-3 is closed.

In RAI 3.8.3-4, dated February 2, 2001, the staff requested that the applicant discuss how the acceptance criteria for the emergency containment cooler heat exchanger tubes consider fatigue failure due to flow-induced vibration. In its response (FPL Letter L-2001-75, dated April 19, 2001), the applicant stated that vibration induced fatigue is fast acting and typically detected early in the component's life, and, as a result, corrective actions are initiated to prevent recurrence. A review of Turkey Point's operating experience for the Emergency containment coolers did not indicate the presence of flow-induced vibration degradation conditions. Therefore, cracking due to mechanical fatigue is not an aging effect requiring management for the Emergency containment coolers. The staff finds the applicant's response reasonable and acceptable and therefore the issue in RAI item 3.8.3-4 is considered resolved. With the resolution of the staff's concerns as discussed above, the staff finds the monitoring and trending methods acceptable.

[Operating Experience] The applicant proposed a one-time inspection which is a new activity that will use techniques with demonstrated capability and a proven industry record to detect wall thickness (loss of material due to erosion). Effective and proven volumetric examination techniques will be selected for use in performing this inspection. This inspection will be performed utilizing approved procedures and qualified personnel.

The staff finds that based on operating experience, the implementation of the emergency containment coolers inspection will provide reasonable assurance that loss of material due to erosion will be managed in the containment coolers and is therefore acceptable.

3.8.3.3 FSAR Supplements

The staff has reviewed the UFSAR Section 16.1.3 provided in Appendix A to the LRA and confirmed that it contains the applicable elements of the program.

3.8.3.4 Conclusion

On the basis of the information provided by the applicant, the staff finds the implementation of the emergency containment cooler inspection program will provide reasonable assurance that loss of material will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

3.8.4 Field-Erected Tanks Internal Inspection

Florida Power and Light described its field-erected tanks internal inspection program in Section 3.1.4 of Appendix B to the LRA. The applicant credits this inspection program with managing, in part, the aging effect of the loss of material due to corrosion of the tanks within the scope of the program. The staff reviewed the Application to determine whether the applicant has demonstrated that the field-erected tanks internal inspection program will adequately manage the loss of material aging effect for the tanks within the scope of the program during the period of extended operation as required by 10 CFR 54.21(a) (3).

3.8.4.1 Summary of Technical Information in the Application

In Appendix B, Section 3.1.4, of the LRA, the applicant described a new aging management program, the field-erected tanks internal inspection, that manages, together with the chemistry control program, the loss of material aging effect for the two condensate storage tanks, two refueling water storage tanks, and the shared demineralized water storage tank. The applicant lists these tanks in Table 3.3-4 (refueling water storage tanks), Table 3.5-2 (demineralized water storage tank), and Table 3.5-3 (condensate storage tanks) of the LRA. These tanks are fabricated from carbon steel and the internal tank surfaces are coated to reduce corrosion. Each of the tanks contains treated water beneath an environment of air/gas.

The applicant plans on implementing the field-erected tanks internal inspection program as a one-time inspection of the two condensate storage tanks, two refueling water storage tanks, and the shared demineralized water storage tank rather than as a periodic inspection program. This one-time inspection will utilize either direct (e.g., divers) or remote (e.g., television cameras, fiber optic scopes, periscopes) observations. In addition to the field-erected tanks internal inspection program, the applicant also plans to use the chemistry control program to monitor the condition of the treated water in each of the tanks.

3.8.4.2 Staff Evaluation

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

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

[Program Scope] The applicant stated in Appendix B, Section 3.1.4, of the LRA, that the field-erected tanks internal inspection program will be a one-time inspection of the two condensate storage tanks, two refueling water storage tanks, and the shared demineralized water storage tank. This one-time inspection will cover selected internal areas, including surface welds, to determine the extent of internal corrosion in the tanks listed above. In order to ensure that the most susceptible internal areas of each tank are inspected, in RAI 3.8.4-2, dated February 2, 2001, the staff requested that the applicant describe the locations within each of the tanks that are the most susceptible to corrosion and to discuss why these locations are the most susceptible. In response (FPL Letter L-2001-75, dated April 19, 2001), the applicant stated that all locations within each of the tanks are considered to be susceptible to corrosion and therefore, all accessible internal surfaces of the tanks will be visually inspected rather than focusing on limited select locations suspected of being more susceptible to corrosion. The applicant's response to the staff is acceptable to close the issue in RAI Item 3.8.4-2.

[Preventive or Mitigative Actions] There are no preventive or mitigative actions taken as part of this program, and the staff did not identify the need for such actions.

[Parameters Inspected or Monitored] The applicant will perform visual inspections to determine the extent of any internal corrosion for each of the tanks. The internal tank surfaces will be examined for evidence of flaking, blistering, peeling, discoloration, pitting, or excessive corrosion. To determine the adequacy of the visual inspection, the staff requested in RAI 3.8.4-4 that the applicant describe the visual examination procedures in more detail, including any lighting and resolution requirements. In addition, 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. Since the program requirements have not yet been established, RAI 3.8.4-4 became Open Item 3.8.4-1(b). In response to Open Item 3.8.4-1, dated November 1, 2001, the applicant stated that 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, which is described in IWA-2210 of ASME Section XI. In addition, the applicant stated that if visual examination of the tanks reveals significant loss of material, the condition would be resolved through the FPL 10 CFR Part 50, Appendix B, corrective action program, which may involve volumetric or surface examinations. The applicant's response to the staff is acceptable to close Open Item 3.8.4-1(b). The staff finds that the monitoring of evidence of flaking, blistering, peeling, discoloration, pitting, or excessive corrosion is acceptable since they are directly related to the degradation of the internal tank surfaces, and visual inspections are effective and adequate to detect this condition.

[Detection of Aging Effects] An appropriate inspection frequency interval is important to ensure that the loss of material aging effect is identified before there is a loss of intended function; however, the applicant has determined that the field-erected tanks internal inspection program is to be a one-time inspection. In RAI 3.8.4-1, the staff has 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 the CSTs. Since the results of previous inspections of the RWSTs and DWST were not provided by the applicant in response to RAI 3.8.4-1, the staff requested further information and RAI 3.8.4-1 became Open Item 3.8.4-1(c). In response to Open Item 3.8.4-1, dated November 1, 2001, the applicant stated that 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. The applicant's expectation is that the RWST will similarly show little or no degradation and, therefore, the one-time field-erected tank internal inspection will provide confirmation that there are no aging effects requiring management for the field-erected tanks. However, if the inspection reveals internal surface degradation of the tanks,

then the degradation will be evaluated and repaired, as necessary, and additional inspections will be scheduled, as needed. The applicant's response to the staff is acceptable to close Open Item 3.8.4-1(c).

[Monitoring and Trending] Since the field-erected tanks internal inspection program is to be a one-time inspection, no monitoring and trending is anticipated; however, the applicant stated in Section 3.1.4 of Appendix B to the LRA, that the results of the one-time inspection will be evaluated to determine if additional actions are required.

[Acceptance Criteria] Specific acceptance criteria have not yet been developed for the field-erected tanks internal inspection program. In Section 3.1.4 of Appendix B to the LRA, the applicant stated that acceptance criteria will be provided in the implementing procedure. Since the review of acceptance criteria are an essential part of the staff evaluation of the effectiveness of an aging management program, the staff requested as Part A of Open Item 3.8.4-1 that the applicant provide acceptance criteria for the field-erected tanks internal inspection program. In response to Open Item 3.8.4-1(a), dated November 1, 2001, the applicant stated that 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 that the tank's intended functions are maintained under all CLB design conditions. The applicant further stated that 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 inspections will be scheduled based on the corrective actions implemented. The applicant's response to the staff is acceptable to close Open Item 3.8.4-1(a).

[Operating Experience] The field-erected tanks internal inspection program is a new program; thus, the applicant did not submit Turkey Point-specific operating experience. However, in response to the staff's RAI 3.8.4-1, the applicant stated that previous inspections of the Unit 4 CST in 1983 and the Unit 3 CST in 1991 revealed corrosion at some of the welds at the roof to wall connection and coating degradation at several areas in the floor and wall of the tank. The applicant attributed these conditions to operational practices and the inadequacy of the original coatings, however, the applicant stated that both of these causes have been corrected. In addition, as documented above under *Detection of Aging Effects*, the DWST was also recently inspected and there were no signs of degradation. The RWSTs have not been previously internally inspected; however, the applicant expects to find little or no degradation to the internal surfaces of the RWSTs. In the event that the field-erected tanks internal inspection reveals degradation of the internal tank surfaces, appropriate corrective actions will be implemented and additional inspections will be scheduled based on the corrective actions implemented. The staff finds that the applicant's operating experience has demonstrated that significant aging of the internal tank surfaces is unlikely and, therefore, a one-time inspection, with the need for further inspections and corrective actions to be determined based on the one-time inspection results, is reasonable and sufficient.

3.8.4.3 FSAR Supplement

The staff has reviewed the UFSAR Section 16.1.4 as amended by the resolution of Open Item 3.8.4-1, and confirmed that it contains an acceptable program description.

3.8.4.4 Conclusions

The staff has reviewed the information in Appendix B, Section 3.1.4, of the LRA and responses to the staff's RAIs and Open Item. The staff also reviewed the program description provided in Section 16.1.4 of the UFSAR. The staff concludes that the applicant has demonstrated that the field-erected tanks internal inspection program will be adequate to detect the presence of the loss of material aging effect for each of the tanks covered by this inspection and that the one-time inspection results will be used to determine the need for additional inspections and/or corrective actions.

3.8.5 Galvanic Corrosion Susceptibility Inspection Program

3.8.5.1 Summary of Technical Information in the Application

Section 3.1.5, "Galvanic Corrosion Susceptibility Inspection Program," of Appendix B to the LRA, describes the program aimed at verifying the integrity of components subject to galvanic corrosion. The staff reviewed this section of the application to determine whether the applicant has demonstrated that the effects of aging caused by galvanic corrosion will be adequately managed during the period of extended operation as required by 10 CFR 54.21(a)(3).

As identified in Chapter 3, the galvanic corrosion susceptibility inspection program is credited for aging management of specific component/commodity groups in the following systems: auxiliary feedwater and condensate storage; chemical and volume control; CCW; containment spray; control building ventilation; emergency containment cooling; emergency diesel generators and support systems; feedwater and blowdown; fire protection; instrument air; normal containment and control rod drive mechanism cooling; reactor coolant; residual heat removal; safety injection; spent fuel pool (SFP) cooling; turbine building ventilation; and waste disposal.

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 of 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. On the basis of 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.

3.8.5.2 Staff Evaluation

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

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

[*Program Scope*] The galvanic corrosion susceptibility inspection program will manage the potential effects of loss of material due to galvanic corrosion on the internal surfaces of susceptible piping and components. Carbon steel components directly coupled to stainless steel components in raw water systems at Turkey Point are the most susceptible to galvanic corrosion. However, baseline examinations will be performed and evaluated to establish if the corrosion mechanism is active in other systems. The program will involve selected one-time inspections, the results of which will be utilized to determine the need for additional actions. The staff finds that the scope of the galvanic corrosion susceptibility inspection program is adequate because locations likely to experience galvanic corrosion will be examined.

[*Preventive or Mitigative Actions*] Components and systems utilize insulating flanges or cathodic protection to minimize galvanic corrosion. The use of insulated flanges and cathodic protection is not credited with the elimination of galvanic corrosion. Since the applicant does not take credit for systems and components that minimize galvanic corrosion, there are no preventive or mitigative actions and the staff does not find a need for any.

[*Parameters Inspected or Monitored*] The program will assess the loss of material due to galvanic corrosion between dissimilar metals in locations determined to represent the most limiting conditions. Selection of the most limiting conditions will be based on high galvanic potential, high cathode/anode area ratio, and high conductivity of the fluid in contact with the materials. The staff finds the general program is acceptable because visual examination of selected locations will establish if galvanic corrosion is occurring.

[*Detection of Aging Effects*] Loss of material due to galvanic corrosion will be evident by material loss at the location of the junction between the dissimilar metals. Volumetric examinations or visual inspections will be utilized to address the extent of material loss.

Initial inspection results will be utilized to assess the need for expanded sample locations. Inspection frequency will be determined based on the corrosion rate identified during the initial inspections. The staff agrees that these are acceptable methods for identifying loss of material.

[*Monitoring and Trending*] These are planned as one-time inspections; therefore, there is no monitoring or trending, and the staff does not find any need for monitoring and trending.

[*Acceptance Criteria*] Wall thickness measurements greater than required minimum wall thickness values for the components will be acceptable. Wall thickness measurements less than required minimum values will be entered into the corrective action program. The staff finds that the acceptance criteria are adequate because this program will establish if the minimum wall thickness requirement is being satisfied.

[*Operating Experience*] Visual and volumetric inspection techniques have been used at Turkey Point for years. These techniques have proven successful in determining the material condition of components.

This is a new program that will use techniques with demonstrated capability and a proven industry record to monitor material loss due to galvanic corrosion. This examination will be performed utilizing approved procedures and qualified personnel. The inspection techniques used in this program have been previously used to monitor material condition for plant systems.

The applicant did not provide any operating experience on galvanic corrosion, either for Turkey Point, Units 3 and 4, or for the nuclear industry in general in the LRA. The applicant provided a summary of their operating experience in RAI response L-2001-65, Attachment 1. They reviewed their plant operating and maintenance history and discovered only a few incidences of loss of material in treated water systems. The applicant identified loss of material due to galvanic corrosion in the plant ventilation chilled water systems. The applicant installed electrical isolation kits and no further galvanic corrosion has been observed. There were also instances of loss of material in air handling units where aluminum fins are in contact with copper tubing in areas where condensation pooling has occurred.

The applicant stated in RAI response L-2001-65 that galvanic corrosion is more likely in raw water than in treated water. The applicant states that the effects of galvanic corrosion are precluded by design using such things as isolation and coating of dissimilar metals. The applicant states that galvanic corrosion is most likely in the intake cooling water (ICW). However, the applicant has the ICW system inspection program instead of the galvanic corrosion inspection program to manage this aging.

3.8.5.3 FSAR Supplement

The staff has reviewed the UFSAR Section 16.1.5 and confirmed that it contains an acceptable program description.

3.8.5.4 Conclusions

The staff has reviewed the information provided in Appendix B, Section 3.1.5 of the LRA and responses to the staff's RAIs. On the basis of this review, as set forth above, the staff concludes that the applicant has demonstrated that there is reasonable assurance that the galvanic corrosion susceptibility inspection program will adequately manage aging effects for dissimilar metals in contact with fluid for the period of extended operation.

3.8.6 Reactor Vessel Internals Inspection Program

Section 3.1.6, "Reactor Vessel Internals Inspection Program," of Appendix B to the LRA describes the program credited for aging management of the reactor vessel internals. The reactor vessel internals inspection program consists of two types of examinations, visual and ultrasonic testing (UT), to manage the aging effects of cracking, reduction in fracture toughness, and loss of mechanical closure integrity.

As described in the LRA, the reactor vessel internals inspection program will involve the combination of several activities culminating in the inspection of Turkey Point Unit 3 and 4 reactor vessel internals once for each unit during the 20-year period of extended operation. The applicant states that this program is intended to supplement the reactor vessel internals inspections required by the ASME Section XI, Subsections IWB, IWC, and IWD inservice inspection program. In addition, ongoing industry efforts are aimed at characterizing the aging effects associated with the reactor vessel internals. As described in response to RAI 3.8.6-1, the applicant is a participant in industry research activities addressing aging effects on reactor vessel internals being conducted by the materials reliability project (MRP) of EPRI. Further understanding of these aging effects will be developed by industry over time and will provide additional bases for the inspections under this program. Pending results of industry progress with regard to validation of the significance of dimensional changes due to void swelling, the applicant states that the visual examinations may be supplemented to incorporate requirements for measurement of critical parts to evaluate potential dimensional changes. Accordingly, an evaluation will be performed to establish the requirements for dimensional verification of critical reactor vessel internals parts as part of the visual examination scope.

Commitment dates associated with the implementation of this new program are provided in Appendix A to the LRA. Specifically, this program will be in place prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4. As described in response to RAI 3.8.6-4, FPL will submit to the NRC a report that will summarize the understanding of aging effects to apply to the reactor vessel internals, and will provide the Turkey Point inspection plan, including required methods for detection and sizing of cracks and acceptance criteria. This report will be submitted prior to the end of the initial 40-year operating license term for Unit 3. As described in response to RAI 3.8.6-3, the first of the reactor vessel internals inspections will occur early in the license renewal period on the unit leading in fluence at that time, and the second inspection will be conducted on the other unit at the next 10-year inspection interval, or 10 to 12 years into the license renewal term.

Since the application focuses discussion of this program around the visual examinations and the UT examinations, the review and evaluation of this program will be structured along those same lines.

3.8.6.1 Visual Examination

3.8.6.1.1 Summary of Technical Information in the Application

The application provides a description of this examination in Section 3.1.6.1 of Appendix B the LRA. The examination description is covered under eight items: scope, preventive actions, parameters monitored, or inspected, detection of aging affects, monitoring and trending, acceptance criteria, confirmation process, and operating experience and demonstration. A description of the contents of the application is provided below in the staff evaluation.

3.8.6.1.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Section 3.1.6.1 of Appendix B of the LRA regarding the applicant's demonstration of the visual examination activity of the reactor vessel internals inspection program to ensure that the effects of aging, as discussed above, will be adequately managed so that the intended functions will be maintained consistent with the CLB throughout the period of extended operation.

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

[*Program Scope*] As described in the application, this activity will manage the aging effects of cracking due to irradiation-assisted stress corrosion (IASCC) and reduction in fracture toughness due to irradiation and thermal embrittlement on accessible parts of the Turkey Point Unit 3 and 4 reactor vessel internals. The reactor vessel internals susceptible to these aging effects and included in the visual examination scope are accessible areas of the lower core plates and fuel pins, lower support columns, core barrel, baffle/former assemblies, thermal shields, and lower support forgings. The program will consist of VT-1 examinations utilizing remote equipment such as television cameras, fiber-optic scopes, periscopes, etc. The staff finds the scope of this program adequate for managing the aging effects for which it is intended because the program addresses the reactor vessel internal components of interest.

[*Preventive or Mitigative Actions*] The application states that there are no practical preventive actions available that will prevent IASCC and reduction in fracture toughness. However, to minimize the potential for IASCC, the concentrations of chlorides, fluorides, and sulfates in the reactor coolant are controlled by implementation of the chemistry control program. The staff agrees with the applicant's conclusions that there are no practical preventive actions.

[*Parameters Monitored or Inspected*] This examination monitors the effects of cracking and reduction in fracture toughness on the reactor vessel internals selected parts by the detection and sizing of cracks. The staff finds that the cited examination will be effective in managing IASCC and reduction in fracture toughness in reactor vessel internals components because this is a proven method for detecting and sizing of cracks in the components.

[*Detection of Aging Effects*] Cracking of reactor vessel selected parts will be detected by performance of VT-1 examinations. Cracking is expected to initiate at the surface and, therefore, will be detectable by visual examination. If ultrasonic examination of bolting (see Section 3.8.6.2.2 of this SER) determines that IASCC is occurring, then enhanced VT-1 inspections capable of detecting 0.5 mile 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 will be performed. The staff finds that the visual examinations described by the applicant will be effective in detecting the aging effects cited in the application because this is a proven method for inspecting these components.

[*Monitoring and Trending*] The VT-1 examination of selected parts of the reactor vessel internals will be performed one time for each unit during the period of extended operation. On the basis of the results of each examination, additional examinations and/or repairs will be scheduled. The staff finds this approach to be acceptable because it provides a reasonable approach for addressing any degradation identified.

[*Acceptance Criteria*] The LRA states that acceptance criteria will be developed prior to the visual examinations, and cracks that are detected during the inspections will be evaluated for determination of the need and method of repair. The staff finds this approach to be acceptable because the acceptance criteria will be developed using acceptable procedures.

[*Operating Experience*] The LRA states that the remote visual examination proposed by this program utilizing equipment such as television cameras, fiber-optic scopes, periscopes, etc., has previously been demonstrated as an effective method to detect cracking of reactor vessel internals. The applicant states in the LRA that similar visual examinations were successfully performed at St. Lucie Unit 1 during the core barrel repair/modification. The staff concludes that the visual examination will be effective in managing the aging effects cited by the applicant because it uses proven techniques for the components of interest.

3.8.6.2 Ultrasonic Examination

3.8.6.2.1 Summary of Technical Information in the Application

The application provides a description of this examination in Section 3.1.6.2 of Appendix B of the LRA. The examination description is covered under eight items: scope, preventive actions, parameters monitored or inspected, detection of aging affects, monitoring and trending, acceptance criteria, confirmation process, and operating experience and demonstration. A description of the contents of the application is provided below in the staff evaluation.

3.8.6.2.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Section 3.1.6.2 of Appendix B of the LRA regarding the applicant's demonstration of the UT examination activity of the reactor vessel internals inspection program to ensure that the effects of aging, as discussed above, will be adequately managed so that the intended functions will be maintained consistent with the CLB throughout the period of extended operation.

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

[*Program Scope*] This activity manages the aging effect of loss of mechanical closure integrity on reactor vessel internals baffle/former bolts, barrel/former bolts, and lower support column bolts. The volumetric examination will involve UT examination on the baffle/former bolts in each unit to supplement the current examination techniques. The results of this examination will be utilized to determine the need for similar examinations of the barrel/former bolts, lower support column bolts, and other reactor vessel internals bolting. Additionally, the baffle/former bolting is the leading location for determining the extent of IASCC that may be occurring 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 0.5mil 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. The staff finds the scope of this program adequate for managing the aging effects for which it is intended.

[*Preventive or Mitigative Actions*] There are no practical preventive actions available that will prevent loss of mechanical closure integrity of reactor vessel internals bolting. However, to minimize the potential for loss of mechanical closure integrity due to IASCC, the concentrations of chlorides, fluorides, and sulfates in the reactor coolant are controlled by implementation of the chemistry control program. There are no preventive or mitigative actions associated with the ultrasonic examination, nor did the staff identify a need for such actions.

[*Parameters Monitored or Inspected*] This examination monitors loss of mechanical closure integrity of the reactor vessel internals bolts by the detection and sizing of cracks. The staff finds that the ultrasonic examination will be effective in managing the cited aging effects in the reactor vessel internals components.

[*Detection of Aging Effects*] The aging effect of loss of mechanical closure integrity of reactor vessel internals bolting will be detected by performance of ultrasonic examinations. The staff finds that the ultrasonic examinations described by the applicant will be effective in detecting the aging effects cited in the application because approved methods will be used to develop these criteria.

[*Monitoring and Trending*] The ultrasonic examination of the reactor vessel internals baffle/former bolts will be performed one time on each unit during the period of extended operation. On the basis of the results of the examination, additional examinations and/or repairs will be scheduled. The staff finds this approach to be acceptable based on industry experience of limited cracking of baffle/former bolts.

[*Acceptance Criteria*] The LRA states that the quantity of cracked baffle/former bolts shall be less than the number of bolts that can be damaged without affecting the intended function of the reactor vessel internals. This quantity will be established by evaluation. The staff finds this approach to be acceptable.

[*Operating Experience*] The LRA states that the UT examination methods are proven techniques that have been used in other programs to successfully detect cracking, and that UT examinations have been demonstrated as an effective method of detecting cracking in baffle/former bolting at other Westinghouse plants. The ultrasonic examinations utilize techniques with a demonstrated capability and a proven industry record to detect cracking. These examinations are performed utilizing approved procedures and qualified personnel.

The staff agrees that UT examination methods are effective for the components of interest.

3.8.6.3 FSAR Supplement

Section 16.1.6, "Reactor Vessel Internals Inspection Program," of Appendix A to the LRA provides an updated FSAR supplement for the reactor vessel internals inspection program, as amended by the applicant's response to RAI 3.8.6-4. The staff concludes that the updated FSAR Supplement is sufficient.

3.8.6.4 Conclusion

The staff has reviewed the information in Appendix B, Section 3.1.6 of the LRA and responses to the staff's RAIs. On the basis of the above evaluations of the visual and ultrasonic examination activities of the reactor vessel internals inspection program, the staff finds that this program provides reasonable assurance that the applicable aging effects will be managed so that reactor vessel internal components will continue to perform their intended functions consistent with the CLB throughout the period of extended operation.

3.8.7 Small Bore Class 1 Piping Inspection

The small bore Class 1 piping inspection program is credited for aging management of small bore Class 1 piping in the reactor coolant systems (RCS).

3.8.7.1 Summary of Technical Information in the Application

The applicant describes the piping inspection in Section 3.1.7, "Small Bore Class 1 Piping Inspection," of Appendix B to the LRA. This inspection will be a one-time inspection of a sample of Class 1 piping less than 4 inches in diameter. As described in Appendix A to the LRA, this inspection will be performed prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4, using a volumetric technique chosen to permit detection and sizing of significant cracking of small bore Class 1 piping. Since this is a one-time inspection, no monitoring or trending is anticipated by the applicant. The evaluation of the inspection results may result in additional examinations consistent with ASME Section XI, subsection IWB. A small sample of the affected welds will be selected for examination based on piping geometry, piping size, and flow conditions. As described in response to RAI 3.8.7-1, the sample of welds to be examined will be selected using a risk-informed approach approved previously by the NRC.

This one-time inspection is described in the LRA as a new activity, which will use techniques with demonstrated capability and a proven industry record to detect piping weld and base material flaws. The applicant states that effective and proven volumetric examination techniques will be selected for use in performing this inspection. This inspection will be performed utilizing approved procedures and qualified personnel. Results and recommendations from industry initiatives will be incorporated into the inspection. The staff reviewed the applicant's description of the program in Section 3.1.7 of Appendix B to the LRA to determine if the small bore Class 1 piping inspection will adequately manage cracking in small bore Class 1 piping welds such that these components will to perform their intended functions for the period of extended operation as required by 10 CFR 54.21(a)(3).

3.8.7.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information included in Section 3.1.7 of Appendix B to the LRA regarding the applicant's demonstration of the small bore Class 1 piping inspection to ensure that the effects of aging, as discussed above, will be adequately managed so that the intended functions will be maintained consistent with the CLB throughout the period of extended operation.

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

[*Program Scope*] The small bore Class 1 piping inspection is a one-time inspection of a sample of Class 1 piping less than 4 inches in diameter. As described in response to RAI 3.8.7-1, the sample of welds to be examined will be selected using a risk-informed approach approved previously by the NRC. Commitment dates associated with the implementation of this new program are provided in Section 16.1.7, of Appendix A to the LRA. The staff agrees with the adequacy of the applicant's description of the scope of this program.

[*Preventive or Mitigative Actions*] The applicant states that no preventive actions are applicable to this inspection. The staff concurs with this finding.

[*Parameters Monitored or Inspected*] The LRA states that the volumetric technique chosen will permit detection and sizing of significant cracking of small bore Class 1 piping. The staff agrees with the adequacy of the examination technique described by the applicant because this is a proven method for this type of inspection.

[*Detection of Aging Effects*] The applicant states that the aging effect requiring management, cracking, will be detected and sized in accordance with the volumetric technique chosen. The staff agrees with the adequacy of the examination technique described by the applicant because this is a standard industry technique.

[*Monitoring and Trending*] The LRA states that this is a one-time inspection and, as such, no monitoring or trending is anticipated. Further, the LRA states that the evaluation of the inspection results may result in additional examinations consistent with ASME Section XI, Subsection IWB. The staff finds this approach acceptable because cracking of small bore piping has not been prevalent in the industry and a one-time inspection program is adequate.

[*Acceptance Criteria*] The LRA states that any cracks identified will be evaluated and, if appropriate, entered into the corrective action program. The staff finds this approach acceptable because industry standards will be used in the acceptance criteria.

[*Operating Experience*] The LRA describes this one-time inspection as a new activity, which will use techniques with demonstrated capability and a proven industry record to detect piping weld and base material flaws. Effective and proven volumetric examination techniques will be selected for use in performing this inspection. This inspection will be performed utilizing approved procedures and qualified personnel. Results and recommendations from industry initiatives will be incorporated into the inspection. The staff finds this approach acceptable.

3.8.7.3 FSAR Supplement

Section 16.1.7 of Appendix A to the LRA states that the small bore Class 1 piping inspection will be performed prior to the end of the initial operating license terms for Turkey Point, Units 3 and 4. The staff agrees with the timing for this inspection. Because the LRA does not specify the number of items to be inspected nor the specific lines to be inspected, the applicant has committed to provide to the NRC a report describing the inspection plan prior to implementation of this inspection (Ref. FPL letter L-2001-136, dated June 25, 2001).

3.8.7.4 Conclusion

The staff has reviewed the information in Section 3.1.7 of Appendix B of the LRA and responses to the staff's RAIs. On the basis of the evaluation of the small-bore Class 1 piping inspection program, the staff finds that this program provides reasonable assurance that the applicable aging effects will be managed so that the small bore Class 1 piping and nozzles will continue to perform their intended functions consistent with the CLB throughout the period of extended operation.

3.9 Existing Aging Management Programs

3.9.1 ASME Section XI Inservice Inspection Programs

The applicant described the inservice inspection (ISI) programs, Section 3.2.1, "ASME Section XI Inservice Inspection Program," of Appendix B to the LRA. The applicant credits the examinations performed under the ASME Code, Section XI, ISI program with managing the effects of aging for Class 1, 2, 3, and MC pressure-retaining components and their supports during the period of extended operation. The staff has reviewed the section of the application to determine whether the applicant has demonstrated that the effects of aging will be adequately managed by the ISI program during the extended period of operation as required by 10 CFR 54.21(a)(3). The ASME Section XI ISI programs are broken down into the following four programs:

- ASME Section XI, subsections IWB, IWC, and IWD inservice inspection program
- ASME Section XI, subsection IWE inservice inspection program
- ASME Section XI, subsection IWF inservice inspection program
- ASME Section XI, subsection IWL inservice inspection program

3.9.1.1 ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program

3.9.1.1.1 Summary of Technical Information in the Application

The ASME Section XI, subsections IWB, IWC, and IWD ISI program is described in Section 3.2.1.1, "ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program," of Appendix B to the LRA. The applicant credits this program for managing the effects of cracking, loss of mechanical closure integrity, and loss of material for piping and components in the reactor coolant system during the period of extended operation. The staff has reviewed the section of the application to determine whether the applicant has demonstrated that the effects of aging will be adequately managed by the ISI plan during the period of extended operation as required by 10 CFR 54.21(a)(3).

As identified in Chapter 3, Table 3.2-1 of the LRA, the ASME Section XI, Subsections IWB, IWC, and IWD ISI program is credited for aging management of specific component/commodity groups in the RCS.

The staff notes that the licensee submitted a request to revise the Turkey Point Unit 3 ISI scope for Class 1 piping to risk informed inservice inspection (RI-ISI). The revision affects the nondestructive examination (NDE) scope of Class 1 piping currently required by ASME Section XI. Examinations performed are based upon the postulated failure mechanism associated with the piping being inspected. The licensee plans to submit a similar request for Turkey Point Unit 4 at a later date. The staff's evaluation of the Unit 3 request is dated November 30, 2000.

In Section 3.2.1.1 of Appendix B of the LRA, the applicant stated its intent to meet the requirements of the latest edition and addenda to the ASME Code, Section XI, that are incorporated by reference in 10 CFR 50.55a(b) for ISI.

3.9.1.1.2 Staff Evaluation

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

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

[*Program Scope*] The ASME Section XI, subsections; IWB, IWC, and IWD ISI program, as defined by the third interval ISI program for Turkey Point, Units 3 and 4, is credited with managing the aging effects of cracking, loss of mechanical closure integrity, and loss of material for piping and components. This program provides for the inspection and examination of components, including welds, pump casing, valve bodies, steam generator tubing, and pressure-retaining bolting. (The staff notes that steam generator tubing is also covered under LRA Section 3.2.14, "Steam Generator Integrity Program," and in this safety evaluation in Section 3.9.14.)

ISI requirements may be modified by applicable relief requests and code cases that are approved specifically for each unit. A particular code edition is applicable for a 120-month interval. Prior to the end of each interval, the program is revised to reflect the updated requirements of 10 CFR 50.55a.

Although ASME Section XI, subsection IWD is included in the scope of this program, this application does not credit subsection IWD for managing the effects of aging of in-scope Class 3 pressure retaining components and their integral attachments. The aging effects of these items are credited by other aging management programs.

The staff finds that the scope of the ASME Section XI, subsections; IWB, IWC, and IWD ISI program is adequate.

[*Preventive or Mitigative Actions*] There are no specific actions under this program to prevent or mitigate the effects of aging. Specific actions that serve to limit the effects of aging for Class 1, 2, and 3 piping and components are conservative design, fabrication, construction, ISIs, and strict control of chemistry. The operating experience with the ISI program indicates that it has been successful in identifying and leading to correction of degradation effects as expected of this program. The staff did not identify a need for preventive actions.

[*Parameters Inspected or Monitored*] ISI includes visual inspections, surface examinations, and volumetric examinations in accordance with the requirements of ASME Section XI. The parameters monitored are specified in the ASME Code for each type of examination required. The staff accepts the parameters being monitored during ISI of Class 1, 2, and 3 components in managing age-related degradation.

[*Detection of Aging Effects*] The degradation of piping and components is determined by visual, surface, or volumetric examination in accordance with the requirements of ASME Section XI as modified by the third interval ISI program for Turkey Point, Units 3 and 4 [Reference B-4 of the LRA]. Piping and components are examined for evidence of operation-induced flaws using volumetric and surface techniques. The VT-1 visual examination is used to detect cracks, symptoms of wear, corrosion, erosion, or physical damage. VT-2 examinations are conducted to detect evidence of leakage from pressure-retaining components. VT-3 examinations are conducted to determine the general mechanical and structural condition of components and to detect discontinuities and imperfections such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. The extent and frequency of inspections are specified in ASME Section XI, as modified in accordance with the third interval ISI program for Turkey Point, Units 3 and 4. The frequency and scope of examinations are sufficient to ensure that the aging effects are detected before they impact the components' intended functions. The inspection intervals are not restricted by the Code to the current term of operation, and are valid for the period of extended operation. The staff accepts the NDE methods prescribed by the Code for each class of components to be reliable and effective in detecting age-related degradation of components that are within the scope of license renewal.

[*Monitoring and Trending*] The frequency and scope of examinations are sufficient to ensure that the aging effects are detected before impacting the component's intended functions. Inspections are performed in accordance with the inspection intervals specified by ASME Code Section XI as modified by the third interval ISI program for Turkey Point, Units 3 and 4.

Examinations performed during any inspection interval that reveal flaws or areas of degradation exceeding the acceptance criteria are to be extended to include additional examinations within the same category. When examination results require evaluation of flaws or areas of degradation, the areas are reexamined during subsequent inspection intervals in accordance with the requirements of ASME Section XI.

Records of the inspection program, examination and test procedures, results of activities, examination/test data, and corrective actions taken or recommended are maintained in accordance with the requirements of ASME Section XI, subsection IWA.

The staff accepts this methodology to undertake further programmatic actions, including additional examinations, corrective actions, and repair and replacement in accordance with ASME Section XI, to manage these aging effects.

[*Acceptance Criteria*] Acceptance standards for the ISIs are identified in ASME Section XI. Relevant indications that are revealed by the ISI may require additional inspections of similar components in accordance with ASME Section XI. Examinations that reveal indications exceeding the acceptance standards are made acceptable by repair, replacement, or evaluation. The staff accepts the flaw evaluation methodology of the Code as the industry standard and, therefore, the management of aging effects based on the Code criteria is acceptable.

[*Operating Experience*] ASME Section XI provides rules and requirements for ISI, testing, repair, and replacement of Class 1, 2, and 3 components. Components are chosen for inspection in accordance with the requirements of subsections IWB, IWC, and IWD and are inspected using the volumetric, surface, or visual examination methods.

The ASME Section XI inspections are conducted as part of the ISIs typically performed during plant refueling outages. The ISI of Class 1, 2, and 3 components and piping has been conducted since initial plant start-up as required by the plant technical specifications and 10 CFR 50.55a. These inspections have documented, evaluated, and corrected degraded conditions associated with piping and components inspected under the program.

Implementation of the ASME Section XI program at Turkey Point currently includes more than 480 Class 1, 2, and 3 examinations per unit per 10-year interval. For Class 1 piping, the examinations have yielded only indications of surface anomalies and surface geometry with no indication of fatigue cracking. For Class 2 piping, the only indications have been surface anomalies, acceptable slag inclusion, surface geometry, and fatigue cracking of steam generator feedwater nozzle reducers. The feedwater reducers were replaced and subsequent inspections are being performed in accordance with the requirements of ASME Section XI.

The staff finds that operating experience shows the ASME Section XI, subsections IWB, IWC, and IWD ISI program has been successful in identifying and leading to correction of aging effects. Therefore, the staff finds the program effective in the management of age related degradation.

3.9.1.1.3 FSAR Supplement

Section 16.2.1.1, "ASME Section XI, IWB, IWC, and IWD ISI Program," of Appendix A to the LRA provides an updated FSAR supplement for the ASME Section XI, Subsections IWB, IWC, and IWD ISI program. The staff concludes that the updated FSAR supplement is sufficient.

3.9.1.1.4 Conclusion

The staff has reviewed the information in Section 3.2.1 of Appendix B of the LRA. On the basis of this review, the staff finds that the ASME Section XI, subsections IWB, IWC, and IWD ISI program provides reasonable assurance that the aging effects of cracking, loss of mechanical closure integrity, and loss of material will be managed such that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis during the period of extended operation.

3.9.1.2 ASME Section XI, Subsection IWE Inservice Inspection Program

3.9.1.2.1 Summary of Technical Information in the Application

The applicant credits this program with managing the effects of loss of material for containment steel components and changes in material properties for elastomers (seals, gaskets, and moisture barriers) associated with containment steel components. The program addresses the following program elements: scope, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, confirmation process, and operating experience and demonstration. These elements are discussed in 3.9.1.2.2.

3.9.1.2.2 Staff Evaluation

Addressing the 10 program elements (scope, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience) provides an efficient method of describing the processes involved in an aging management program.

It is noted that corrective actions, confirmation process, and administrative controls for license renewal are in accordance with the site-controlled quality assurance program pursuant to 10 CFR Part 50, Appendix B, and covers all SSCs subject to an aging management review.

The staff evaluation of the applicant's quality assurance program is provided separately in Section 3.1.2 of this safety evaluation report. This program satisfies the elements of corrective actions, confirmation process and administrative controls. The remaining seven elements are discussed below.

[Program Scope] The program includes the examination, testing, and repair/replacement activities for the metallic components, moisture barriers, seals, and gaskets of the containment pressure boundary.

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

In its response, the applicant stated, "moisture barriers located at the interface of the containment liner and concrete floor are credited in the determination of aging effects for the containment liner plate, and the aging degradation of the moisture barrier is provided by the implementation of ASME Section XI, Subsection IWE." On the subject of the protective benefits of coatings and cathodic protection, the applicant stated that the existing plant procedures ensured that these protective measures were effective. However, the applicant argued that these protective measures did not perform a license renewal intended function as defined in 10 CFR 54.4(a)(1), (2), and (3) and they were not credited in the determination of aging effects requiring management for protected structures and components. Therefore, coatings and cathodic protection did not require aging management review and aging management. In response to RAI 3.6.1.5-2, however, the applicant has provided the procedures used for ensuring the effectiveness of protective coatings. In a discussion on April 11, 2001, the applicant emphasized that the procedures for ensuring the effectiveness of the cathodic protection system are available at the plant site for a staff review. In the AMR inspection during August 20 - September 14, 2001, the inspectors verified that the operation procedures for the CPS are available at the plant site and are adequate.

[Parameters Monitored or Inspected] The parameters monitored and inspected are in accordance with the requirements of Subsection IWE of the ASME Section XI Code (the Code). They include examination categories E-A for containment surfaces; E-C for augmented examination; E-D for seals, gaskets, and moisture barriers; E-G for pressure-retaining bolting; and E-P for pressure-retaining components. For seals and gaskets, the applicant takes credit for implementing Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," as required by Examination Category E-P. The staff finds the parameters monitored and inspected acceptable.

[Detection of Aging Effects] Implementation of Subsection IWE examination requirements is credited for detecting aging effects of metal surfaces, such as pitting, excessive corrosion, and arc strikes. To detect the aging effects on seals and gaskets, the applicant relies on the requirements of Appendix J to 10 CFR Part 50.

With reference to the detection of aging effects element, the bottom liner plate of the containment structure at Turkey Point is covered with fill concrete, and hence its direct examination is not feasible. At the same time, borated water leaks and thermal and shrinkage related cracking of the fill concrete could give rise to corrosion of the bottom liner plate. In RAI 3.9.1.2-2 the staff asked if FPL has any program, whether as part of the IWE ISI or as part of the maintenance rule programs to detect the degradations and aging effects of the bottom liner plate. In the absence of a specific program, the applicant was asked to confirm that the bottom liner plate is not subjected to such degradation.

In its response to RAI 3.9.1.2-2, the applicant explained, that the Turkey Point containment structures have bottom liner plates that are embedded in the concrete with no exposed surfaces. The 18-inch thick concrete over the bottom liner protects the steel from corrosion. Containment concrete components are constructed of dense, well-cured concrete consistent with the guidance provided in ACI 201.2R-77. The concrete was designed in accordance with ACI 318-63. The aggregates were tested in accordance with ASTM C295. The concrete over the bottom liner is not normally exposed to an aggressive environment. These features ensure concrete cracking is minimized. Consequently, the concrete over the containment liner plate provides adequate protection of the inaccessible portions of the liner plate. In addition, a moisture barrier is provided that prevents intrusion of moisture between the concrete and the inaccessible liner surfaces. Additionally, when events occur such as borated water leaks, potential degradation of inaccessible structures is evaluated as part of the Corrective Action Program. Finally, the containment liner plate is periodically pressure tested in accordance with the ASME Section XI, Subsection IWE, Inservice Inspection Program (Category E-P), described in Application Appendix B, Subsection 3.2.1.2 (page B-30). Based on the design features and programs discussed above, there is reasonable assurance that the containment liner plate will continue to perform its intended function throughout the period of extended operation. Based on this response, the staff concludes that the applicant's program for managing the degradation of inaccessible liner plate is reasonable. The issue in RAI 3.9.1.2-2 is therefore closed. Based on the program element and the additional information provided by the applicant, the staff finds this element of the program acceptable.

[Monitoring and Trending] For frequency of examinations and augmented examinations which are required for monitoring and trending the aging effects the applicant relies on the examination and accepted criteria prescribed in subsection IWE. Furthermore, the applicant states that examinations performed during any inspection interval that reveal flaws or areas of degradation exceeding the acceptance criteria are expanded to include additional examinations within the same category. When examination results require evaluation of flaws or areas of degradation, the area(s) are reexamined during the next inspection interval. Flaws or areas of degradation are documented and evaluated in accordance with the corrective action program and the requirements of the ASME Section XI, Subsection IWE Inservice Inspection Program. The staff finds it acceptable

[Acceptance Criteria] Acceptance criteria are based on the acceptance standards established in IWE-3000 of Subsection IWE of the ASME Section XI Code. Moreover, the applicant stated that the inspection results that reveal evidence of degradation exceeding the acceptance standards may be subjected to additional inspections to determine the nature and extent of the conditions. The staff considers this acceptable.

[Operating Experience and Demonstration] The applicant stated that prior to the implementation of Subsection IWE as required by 10 CFR 50.55a, the examination of the containment's steel components were performed in accordance with the requirements of Appendix J to 10 CFR Part 50. The Appendix J tests performed at the Turkey Point units during the years of operation have not shown any loss of intended function of the containment steel components. Moreover, the applicant stated that material properties for nonmetallic components (such as gaskets and seals) change over time, and these components are replaced in accordance with approved plant procedures. Based on the inspections performed prior to the implementation of Subsection IWE, as part of the operating experience, in RAI 3.9.1.2-4 the applicant was asked to provide a summary of significant events related to the following failure mechanisms:

- liner corrosion
- major penetrations leakage (equipment hatches, airlocks, main steam line, feedwater line) that does not meet the Type B leakage rate requirements
- leakage and corrosion of bellows (if applicable)
- isolation valve leakages (system or Type B test)
- Type A tests that do not meet the containment leak rate criteria

The applicant was also asked to include the corrective actions taken and procedures modified to alleviate such events in the future. In its response, the applicant provided a summary of the operating experience related to the five items in the RAI. These responses indicated that the applicant is fully cognizant with the plant-specific experience, and the aging management program factors in the lessons learned from the operating experience. The issue in RAI item 3.9.1.2-4 is therefore closed

The staff believes that the applicant has provided pertinent operating experience and the program element is acceptable.

3.9.1.2.3 FSAR Supplement

UFSAR Supplement Section 16.2.1.2 included with the application contains a sufficient program description.

3.9.1.2.4 Conclusion

Based on the staff's review described above, the staff concludes that this aging management program provides reasonable assurance that the aging of the pressure retaining components of the primary containment structures at Turkey Point, Units 3 and 4, will be adequately managed during the period of extended operation.

3.9.1.3 ASME Section XI Subsection IWF Inservice Inspection Program

3.9.1.3.1 Summary of Technical Information in the Application

The applicant described its ASME Section XI, Subsection IWF Inservice Inspection Program in Section 3.2.1.3 of Appendix B to the Application. The applicant stated that the program is credited for aging management of Class 1, 2, and 3 component supports in the containments, auxiliary building, emergency diesel generator building, and yard structures.

3.9.1.3.2 Staff Evaluation

As indicated in Table 3.6-2 of the Application, the containments contain safety-related piping and component supports, reactor vessel supports, steam generator supports, pressurizer supports, reactor coolant supports, and surge line supports, all manufactured from carbon steel, which are exposed to the containment air environment. The applicant credited the Subsection IWF Inservice Inspection Program for managing the aging effect (loss of material) for these piping and component supports. Tables 3.6-3, 3.6-10, and 3.6-20 of the Application indicated the auxiliary building, emergency diesel generator building, and yard structures contain safety-related piping and component supports, manufactured from carbon steel, which are exposed to an indoor environment that is not air-conditioned. Based on Table 3.6-20, the yard structures also contain safety-related piping and component supports, manufactured from carbon steel, which are exposed to the outdoor environment. The applicant credited the Subsection IWF Inservice Inspection Program for managing the aging effect (loss of material) for these piping and component supports.

The staff evaluation of the Subsection IWF Inservice Inspection Program focused on how the program manages the aging effect of loss of material through effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

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

[Program Scope] The applicant stated that the ASME Section XI, Subsection IWF Inservice Inspection Program is credited with managing the aging effect of loss of material for Class 1, 2, and 3 component supports (including pipe supports) located in the containments, auxiliary building, emergency diesel generator building, and yard structures. The scope of the Turkey Point program provides inspection and examination of accessible surface areas of these component supports. This is acceptable to the staff.

[Preventive Actions] The applicant stated that carbon steel surfaces are typically coated, in accordance with plant procedures, to reduce the effects of loss of material due to corrosion. Coatings minimize corrosion by limiting exposure to the environment. However, coatings are not credited in the determination of the aging effects requiring management. Therefore, no preventive actions are applicable to this program.

[Parameters Monitored or Inspected] The applicant stated that Class 1, 2, and 3 component supports are examined in accordance with ASME Section XI, Subsection IWF. The Subsection IWF Inservice Inspection Program provides for visual examination for general corrosion that could reduce the structural capacity of the component supports. This is acceptable to the staff, because the is in accordance with accepted industry code.

[Detection of Aging Effects] The applicant stated that the presence of corrosion that could lead to loss of material is determined by visual inspection of component supports. Surfaces are examined for evidence of flaking, blistering, peeling, discoloration, wear, pitting, corrosion, arc strikes, gouges, surface discontinuities, dents, or other signs of surface irregularities. The extent and frequency of the inspections are in accordance with ASME Section XI, Subsection IWF. This is acceptable to the staff.

[Monitoring and Trending] Selected supports are monitored during each inspection period. The program inspects 25% of non-exempt Class 1 piping supports, 15% of Class 2 piping supports, and 10% of Class 3 piping supports, plus several unique supports other than piping supports. The applicant stated that, for those component supports within a system that have similar design, function, and service, only one support is examined. Unacceptable supports are subject to corrective measures or evaluation, and are reexamined during the next inspection period. This is acceptable to the staff, because this is in accordance with accepted industry code.

[Acceptance Criteria] The applicant stated that acceptance standards for the examination and evaluation of supports are provided in ASME Section XI, Subsection IWF. A condition observed during a visual examination that requires supplemental examination, corrective measures, repair, replacement, or analytical evaluation is categorized as a relevant condition and is not considered acceptable. This is acceptable to the staff, because this is in accordance with accepted industry code.

[Operating Experience] The ASME Section XI, Subsection IWF, inspections are conducted as part of the inservice inspections typically during plant refueling outages. The applicant stated that the inspection of Class 1, 2, and 3 component supports has been conducted since initial plant startup, as required by the Technical Specifications.

ASME Section XI provides the rules and requirements for inservice inspection testing, repair, and replacement of Class 1, 2, and 3 component supports. The ASME Section XI, Subsection IWF Inservice Inspection Program applies to Class 1, 2, and 3 component supports. These supports are chosen for inspection in accordance with the requirements of ASME Section XI, Subsection IWF, and shall be inspected using visual examination methods.

The visual examinations of Class 1, 2, and 3 component supports look for deformations or structural degradations, corrosion, and other conditions that could affect the intended function of the support. All conditions noted during the inspection of component supports, whether or not they are considered to require further review, are documented on inspection reports. The applicant stated that the FPL Nuclear Division Quality Assurance Department performed an audit of the inservice inspection program, and concluded that the program was complete and in compliance with the requirements of the ASME Code, Section XI, and applicable commitments.

The staff finds that the past plant operation serves to ensure successful future performance of the ASME Section XI, Subsection IWF ISI program, and is acceptable.

3.9.1.3.3 FSAR Supplement

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

3.9.1.3.4 Conclusion

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

3.9.1.4 ASME Section XI, Subsection IWL Inservice Inspection Program

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

3.9.1.4.1 Summary of Technical Information in the Application

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

3.9.1.4.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.1.4.3 FSAR Supplement

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

3.9.1.4.4 Conclusion

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

3.9.2 Boraflex Surveillance Program

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

3.9.2.1 Summary of Technical Information in the Application

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

3.9.2.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.2.3 FSAR Supplement

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

3.9.2.4 Conclusion

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

3.9.3 Boric Acid Wastage Surveillance Program

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

3.9.3.1 Summary of Technical Information in the Application

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

3.9.3.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.3.3 FSAR Supplement

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

3.9.3.4 Conclusions

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

3.9.4 Chemistry Control Program

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

3.9.5 Containment Spray System Piping Inspection Program

3.9.5.1 Summary of Technical Information in the Application

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

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

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

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

3.9.5.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.5.3 FSAR Supplements

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

3.9.5.4 Conclusion

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

3.9.6 Environmental Qualification Program

3.9.6.1 Summary of Technical Information in the Application

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

3.9.6.2 Staff Evaluation

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

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

[Program Scope] The scope of the program includes the environmentally qualified devices that are within the scope of 10 CFR 50.49 including the following categories of electrical equipment located in a harsh environment:

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

The staff considers the scope of the program acceptable.

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

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

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

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

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

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

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

3.9.6.3 FSAR Supplement

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

3.9.6.4 Conclusion

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

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

3.9.7 Fatigue Monitoring Program

3.9.7.1 Summary of Technical Information in the Application

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

3.9.7.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

3.9.7.3 FSAR Supplement

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

3.9.7.4 Conclusion

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

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

3.9.8 Fire Protection Program

3.9.8.1 Summary of the Technical Information in the Application

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

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

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

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

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

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

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

3.9.8.2 Staff Evaluation

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

The following specific component/commodity groups are identified:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.8.3 FSAR Supplements

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

3.9.8.4 Conclusion

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

3.9.9 Flow-Accelerated Corrosion Program

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

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

3.9.9.1 Summary of Technical Information in the Application

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

- main steam and turbine generator
- feedwater and blowdown

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

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

3.9.9.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information in the LRA regarding the applicant's demonstration that the FAC program will ensure that the effects of aging due to FAC will be adequately managed so that intended functions will be maintained consistent with CLB throughout the period of extended operation for all affected components in the systems included in the LRA. After completing the initial review, by letter dated February 1, 2001, the staff issued several requests for additional information (RAIs). By letter dated April 19, 2001, the applicant responded to the staff's RAIs.

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

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

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

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

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

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

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

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

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

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

3.9.9.3 FSAR Supplement

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

3.9.9.4 Conclusions

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

3.9.10 Intake Cooling Water System Inspection Program

3.9.10.1 Summary of the Technical Information in the Application

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

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

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

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

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

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

3.9.10.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.10.3 FSAR Supplements

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

3.9.10.4 Conclusion

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

3.9.11 Periodic Surveillance and Preventive Maintenance Program

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

3.9.11.1 Summary of Technical Information in Application

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

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

3.9.11.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.9.11.3 FSAR Supplement

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

3.9.11.4 Conclusion

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

3.9.12 Reactor Vessel Head Alloy 600 Penetration Inspection Program

3.9.12.1 Summary of Technical Information in the Application

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

3.9.12.2 Staff Evaluation

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

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

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

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

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

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

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

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

3.9.12.3 FSAR Supplement

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

3.9.12.4 Conclusions

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

3.9.13 Reactor Vessel Integrity Program

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

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

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

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

The four subprograms are reviewed separately in the following paragraphs.

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

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

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

3.9.13.1 Reactor Vessel Surveillance Capsule Removal and Evaluation

3.9.13.1.1 Summary of Technical Information in the Application

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

3.9.13.1.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

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

3.9.13.1.3 FSAR Supplement

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

3.9.13.1.4 Conclusion

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

3.9.13.2 Fluence and Uncertainty Calculations

3.9.13.2.1 Summary of Technical Information in the Application

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

3.9.13.2.2 Staff Evaluation

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

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

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

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

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

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

The fluence calculation methods include the following:

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

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

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

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

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

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

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

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

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

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

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

3.9.13.2.3 FSAR Supplement

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

3.9.13.2.4 Conclusion

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

3.9.13.3 Monitoring Effective Full Power Years

3.9.13.3.1 Summary of Technical Information in the Application

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

3.9.13.3.2 Staff Evaluation

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

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

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

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

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

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

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

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

3.9.13.3.3 FSAR Supplement

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

3.9.13.3.4 Conclusion

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

3.9.13.4 Pressure-Temperature Limit Curves

3.9.13.4.1 Summary of Technical Information in the Application

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

3.9.13.4.2 Staff Evaluation

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

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

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

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

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

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

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

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

3.9.13.4.3 FSAR Supplement

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

3.9.13.4.4 Conclusion

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

3.9.14 Steam Generator Integrity Program

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

3.9.14.1 Summary of Technical Information in the Application

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

3.9.14.2 Staff Evaluation

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

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

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

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

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

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

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

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

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

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

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

3.9.14.3 FSAR Supplement

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

3.9.14.4 Conclusion

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

3.9.15 Systems and Structures Monitoring Program

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

3.9.16 Thimble Tube Inspection Program

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

3.9.16.1 Summary of Technical Information in the Application

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

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

3.9.16.2 Staff Evaluation

In accordance with 10 CFR 54.21(a)(3), the staff reviewed the information in the LRA regarding the applicant's demonstration that the effects of aging will be adequately managed so that intended function will be maintained consistent with the CLB throughout the period of extended operation for the incore instrumentation thimble tubes. The staff evaluation of the thimble tube inspection program focused on effective incorporation of the following 10 elements: program scope, preventive or mitigative actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience.

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

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

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

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

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

3.9.16.3 FSAR Supplement

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

3.9.16.4 Conclusion

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

4. TIME-LIMITED AGING ANALYSIS

4.1 Identification of Time-Limited Aging Analyses

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

4.1.1 Summary of Technical Information in the Application

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

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

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

4.1.2 Staff Evaluation

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

4.1.3 Conclusions

The NRC staff concludes that the applicant has provided a list of acceptable TLAAs as defined in 10 CFR 54.3, and that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA as defined in 10 CFR 54.3.

4.2 Reactor Vessel Irradiation Embrittlement

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

4.2.1 Summary of Technical Information in the Application

The applicant described its reactor vessel irradiation embrittlement TLAAs in Section 4.2 of the LRA. The TLAAs evaluated in Section 4.2 of the LRA include analyses and calculations performed to show compliance with 10 CFR Part 50.60, Appendix G to 10 CFR Part 50 regarding P-T limits and acceptable Charpy USE values and 10 CFR 50.61 regarding protection against PTS events. The TLAAs are reviewed by the staff in the following paragraphs.

4.2.2 Staff Evaluation

In Section 4.2 of the LRA, the applicant stated that the group of TLAAs in this section relate to the effect of irradiation embrittlement on the beltline regions of the Turkey Point Units 3 and 4 reactor vessels. The calculations discussed in this section use predictions of the cumulative effects of irradiation embrittlement on the reactor vessels. The staff has reviewed the reactor vessel integrity program in Section 3.9.13 of this SER and finds it acceptable for the period of extended operation. The three aspects of reactor vessel embrittlement are reactor vessel resistance to failure during PTS events, the maintenance of acceptable Charpy USE levels, and analysis of P-T limits. The maximum anticipated effects of PTS, USE, and P-T limits would be in the reactor vessel beltline region at the end of the period of extended operation. A discussion of the three TLAAs is provided below.

Pressurized Thermal Shock

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

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

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

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

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

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

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

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

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

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

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

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

and σ_Δ is the standard deviation for ΔRT_{NDT}

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

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

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

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

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

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

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

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

Charpy Upper-Shelf Energy

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

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

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

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

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

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

The applicant's evaluation concluded that the limiting weld for the Turkey Point Units 3 and 4 reactor vessels satisfies the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Appendix K, for ductile flaw extension and tensile instability. The analysis associated with USE has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii). The staff concludes that the Turkey Point RPVs will have continued acceptable equivalent margins of safety against fracture through 48 EFPYs.

Pressure-Temperature Limits

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

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

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

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

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

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

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

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

4.2.3 FSAR Supplement

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

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

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

4.2.4 Conclusion

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

4.3 Metal Fatigue

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

4.3.1 Summary of Technical Information in the Application

The applicant discusses the criteria used for the design of reactor coolant loop components in Section 4.3.1 of the LRA. The applicant indicates that the reactor vessels, reactor vessel internals, pressurizers, steam generators, reactor coolant pumps, and pressurizer surge lines have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. Fatigue analyses were performed for the critical locations in these components using conservative assumptions regarding the anticipated plant operational cycles. The applicant indicated that a review of the Turkey Point Units 3 and 4 operating history indicates that the number of operational cycles assumed in the design of these components bounds the number of cycles anticipated for the period of extended operation and, therefore, the analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

The applicant referenced the Turkey Point Fatigue Monitoring Program (FMP) as a confirmatory program that assures the number of design cycle limits are not exceeded during the period of extended operation. The FMP is described in Appendix B of the LRA.

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

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

The applicant described the criteria used for the reactor coolant loop piping and balance-of-plant piping in Section 4.3.4 of the LRA. This piping, except for the pressurizer surge lines and the Unit 4 emergency diesel generator safety-related piping, was designed to the requirements the ANSI B31.1, "Power Piping." The pressurizer surge lines were designed to the Class 1 requirements of the ASME Code. These lines are covered in the applicant's fatigue assessment discussed in Section 4.3.1 of the LRA. The Unit 4 emergency diesel generator safety-related piping was designed to the Class 3 requirements of the ASME Code, which are equivalent to the ANSI B31.1 requirements.

ANSI B31.1 requires a reduction factor be applied to the allowable bending stress range if the number of full-range thermal cycles exceeds 7,000. The applicant stated that its review of plant operating practices indicates that the number of thermal cycles assumed in the analysis will not be exceeded during the period of extended operation. Therefore, the applicant concluded that the analyses of these piping components remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

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

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

4.3.2 Staff Evaluation

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

The specific design criterion for Class 1 components involves calculating the cumulative usage factor (CUF). The fatigue damage caused by each thermal or pressure transient depends on the magnitude of the stresses caused in the component by the transient. The CUF sums the fatigue resulting from each transient. The design criterion requires that the CUF not exceed 1.0. The applicant stated that a review of the plant operating history indicates that the postulated number of cycles and severity of the transients assumed in the design of these components envelops the expected transients during the period of extended operation. Table 4.1-8 of the Turkey Point UFSAR contains a list of transient design conditions and associated design cycles used to evaluate RCS components. In RAI 4.3.1-1, dated February 2, 2001, the staff requested that the applicant provide the following information:

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

- The number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years.

The applicant provided the information in its April, 19, 2001, response to RAI 4.3.1-1. The applicant obtained the current number of operating cycles from its FMP, which has been ongoing since initial plant startup. The applicant based its estimate of the number of cycles of design transients for 60 years of plant operation on the mean frequency of occurrence through June 1988 for most of the design transients. The applicant indicated that the mean frequency method is too conservative for plant heatup and cooldown transients and for reactor trip transients because of the large number of these transients in the early years of operation. The applicant gave more weight to recent operating history of the plant to estimate the number of cycles of these transients for 60 years of plant operation. The staff considers the method described by the applicant to estimate the number of transient cycles for 60-years of plant operation reasonable. The applicant's FMP will continue to track the number of these cycles during the period of extended operation. The staff review of the FMP is contained in Section 3.9.7 of this SER.

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

NRC Bulletin (BL) 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," identified a concern regarding the potential for temperature stratification or temperature oscillations in unisolable sections of piping attached to the RCS. NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification," identified a concern regarding the potential temperature stratification and thermal striping in the pressurizer surge line. In RAI 4.3.1-3, dated February 2, 2001, the staff requested that the applicant describe the actions taken to address these bulletins during the period of extended operation. In an April 19, 2001, response to RAI 4.3.1-3, the applicant indicated that no calculations that meet the definition of a TLAA were performed in response to NRC BL 88-08. The applicant further indicated that fatigue analyses of the Turkey Point Units 3 and 4 surge lines were performed in response to NRC Bulletin 88-11 and these were evaluated as TLAAs for the period of extended operation. The applicant's TLAA of the surge lines is discussed later in the staff evaluation. This RAI item is therefore closed.

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

(1) Show the ASME Section III Class 1 CLB CUFs for the applicable subcomponents of Turkey Point Units 3 and 4 pressurizers specified in Table 2-10 of WCAP-14574, including consideration of environmental effects on the fatigue curves and the corresponding CUFs for the extended period of operation.

(2) WCAP-14574, Section 3.8.3, lists other off-normal and additional transients. WCAP-14574, Section 3.8.4, described recently discovered surge line inflow/outflow thermal transients. These thermal cyclic transients were not considered in the CLB fatigue analyses of Westinghouse pressurizers, including Turkey Point Units 3 and 4. Provide the highest CUFs, considering these transients for the following pressurizer subcomponents for the extended period of operation:

    (a) Surge nozzle
    (b) Lower head region
    (c) Heater wells
    (d) Support skirt and flange

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

In an April 29, 2001, response to RAI 4.3.1-4, the applicant provided a table of the CLB CUFs for the pressurizer subcomponents. All CUFs were shown to meet the ASME Section III Class 1 CUF criterion, without environmental considerations. The applicant stated that the CUFs for critical locations on the pressurizer were originally determined using CLB design transients and frequencies that were intended to be conservative and bounding for all foreseeable plant operational conditions. As discussed previously, the applicant compared the frequencies of the actual plant transients, obtained from its FMP, with the frequencies of the design transients. Based on this comparison, the applicant concluded that the CLB design transients and their frequencies were conservative and bounding for the period of extended operation, and therefore, the CLB CUFs for the Turkey Point Units 3 and 4 pressurizers were also conservative and bounding for the period of extended operation. The applicant further indicated that it will monitor the CLB design transients using the Turkey Point FMP to assure that the number of design transients used in the evaluation of the pressurizer is not exceeded in the period of extended operations. The applicant stated that the CLB CUFs for the surge nozzle, the lower head region, the heater wells, and the support skirt and flange given in the response to this RAI include consideration of the off-normal and additional transients discussed in Section 3.8.3 of WCAP-14754, as applicable, including specific consideration of insurge/outsurge transients described in Section 3.8.4 of the report.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Renewal Application Item (1):

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

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

Renewal Application Item (2):

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

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

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

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

4.3.3 FSAR Supplement

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

4.3.4 Conclusions

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

4.4 Environmental Qualification

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

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

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

4.4.1 Summary of Technical Information in the Application

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

Scope of EQ Equipment

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

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

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

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

EQ Process

The EQ program has three main elements:

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

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

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

Replacement of Equipment

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

Analysis of the Qualified Life

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

- Thermal Considerations

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

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

- Radiation Considerations

  The Turkey Point EQ program has established bounding radiation dose qualification values for all environmentally qualified components. These bounding radiation dose values were determined by component vendors through testing. To verify that the bounding radiation values are acceptable for the period of extended operation, 60-year total integrated dose (TID) values were determined and then compared to the bounding values. The TID for the 60-year period is determined by adding the established accident dose to the 60-year normal operating dose for the component. The 60-year normal operating dose is obtained by multiplying the current 40-year normal operating dose by 1.5. The established post-accident dose is large when compared to the change in normal operating dose from 40 to 60 years and the original 40-year inside containment TID was rounded up. The current 40-year inside containment TID bounds the 60-year TID.

- Wear Cycle Considerations

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

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

Refurbishment of EQ Electrical Equipment

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

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

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

Ongoing Qualification or Retesting

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

Procurement of EQ Equipment

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

Plant Environmental Changes

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

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

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

EQ Generic Safety Issue (GSI)

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

- GSI-168 issues have not been identified to a point that a license renewal applicant can be reasonably expected to address these issues, specifically at this time; and

- An acceptable approach is to provide a technical rationale demonstrating that the CLB for EQ will be maintained in the period of extended operation.

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

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

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

- An applicant can develop a plant-specific aging management program that incorporates a resolution to the aging issue.

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

4.4.2 Staff Evaluation

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

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

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

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

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

- 4.4.1.7 Conax Penetrations
- 4.4.1.8 Conax Unitized Resistance Temperature Detectors
- 4.4.1.9 Champlain Cables
- 4.4.1.10 Crouse Hinds Penetrations
- 4.4.1.11 General Atomic Radiation Monitors
- 4.4.1.12 General Cables
- 4.4.1.13 General Electric Cables
- 4.4.1.14 General Electric Terminal Blocks
- 4.4.1.15 Joy Emergency Containment Cooler and Emergency Containment Filtration Fan Motors
- 4.4.1.16 Limitorque Valve Operators With Reliance Motors for Use Inside Containment
- 4.4.1.17 Limitorque Valve Operators With Reliance Motors With Class H(RH) Insulation for Use Inside Containment
- 4.4.1.18 Limitorque Valve Operators With Reliance Motors for Use Outside Containment
- 4.4.1.19 Limitorque Valve Operators With Peerless Motors for Use Outside Containment
- 4.4.1.20 Okonite Cables
- 4.4.1.21 Raychem Heat Shrink Sleeving
- 4.4.1.22 Raychem Cables
- 4.4.1.23 Macworth Rees Pushbutton Stations
- 4.4.1.24 Rockbestos Cables
- 4.4.1.26 3M Insulating Tape and Scotchfil
- 4.4.1.27 Westinghouse Residual Heat Removal Pump Motors
- 4.4.1.28 Westinghouse Containment Spray Pump Motors
- 4.4.1.29 Westinghouse Safety Injection Pump Motors
- 4.4.1.30 Combustion Engineering Mineral-Insulated Cables and Connectors
- 4.4.1.31 Kerite HTK/FR Cables
- 4.4.1.32 Kerite FR2/FR Cables
- 4.4.1.33 Kerite FR/FR Cables
- 4.4.1.34 Kerite HTK/FR Power Cables
- 4.4.1.35 Teledyne Thermatics Cables
- 4.4.1.36 Weed Resistance Temperature Detectors
- 4.4.1.37 Amertace NQB Terminal Blocks
- 4.4.1.38 Patel/EGS Conformal Splices
- 4.4.1.39 Patel/EGS Grayboot Connectors

- In response to the staff's concern regarding the aging effect of energizing the ASCO solenoid valves during testing, the applicant stated that the FPL calculation assumed that these are normally deenergized and the energization time during testing of the valves (which could be 1000 times over their lifetimes) was considered to be insignificant because it takes 2 hours for an ASCO solenoid valve to reach thermal equilibrium once it is energized. In addition, 60 years is 29% of the calculated deenergized life of 207 years (EQ Documentation Package 3.0, Rev. 6). This leaves 71% of the calculated life as margin. Multiplying the calculated inside containment energized life of 4.6 years by 71% leaves 3.25 years that the normally deenergized solenoid valves could be energized over the 60-year qualified life. This would allow each of the 1000 cycles to remain energized for over 28 hours (3.25 years x 365.25 days x 24 hours/1000 cycles) plus the additional time to reach thermal equilibrium.

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

- In response to the staff's concern regarding major plant modifications or events of sufficient duration to change the temperature and radiation values that were assumed in the EQ calculations, the licensee stated that there have been no major plant modifications or events at Turkey Point Units 3 and 4 that have changed the temperature and radiation values used in the EQ analyses. The postulated normal operating dose rates are based on the assumption of 1% failed fuel, which is 10 times the amount of fuel leakage that has been recorded at Turkey Point. The postulated accident doses are based on the conservative assumptions and methodologies in NUREGs-0578, -0737, and -0588. Any plant modifications that could affect the qualification of a component in the EQ program are addressed and resolved in the modification package. The effect of events on the qualification is addressed and resolved by the corrective action process. The staff concludes that the applicant addressed the staff's concern adequately.

- In response to the staff's request for the basis for 10°C rise above the maximum ambient temperature for power cables, the licensee stated that the 10 °C rise is conservative based on the maximum cable temperature rise of 3.2 °C for the 4160 VAC EQ motors of the safety injection and residual heat removal pumps. The applicant performed additional screening of the cable temperature rise for the 480 VAC EQ motors inside and outside containment including the emergency containment filter, emergency containment cooler, and containment spray pump motors. For the emergency containment cooler and filter motor cable inside containment, the temperature rises are 13.31 °C and 9.72 °C, respectively, above the 50 °C ambient.

  For the emergency containment cooler, filter, and containment spray pump motor cable outside containment, the temperature rises are 22.89 °C, 9.39 °C, and 18.63 °C respectively, above a 40 °C ambient. Although the actual temperature rises are greater than the 10 °C continuous temperature rise assumption, when actual operating times of the emergency containment cooler and containment spray pump motors are considered (0.25 and 0.3 years, respectively, over a 60-year period), the 10 °C continuous temperature rise assumption is over three times as harsh for both inside and outside containment. Therefore, the 10 °C rise applied continuously for 60 years is a conservative value for ohmic heating. The staff concludes that the applicant addressed the staff's concern adequately.

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

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

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

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

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

4.4.3 FSAR Supplement

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

4.4.4 Conclusions

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

4.5 Containment Tendon Loss of Prestress

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

4.5.1 Summary of Technical Information in the Application

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

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

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

4.5.2 Staff Evaluation

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

- Analysis of the time-dependent assumptions of prestressing losses from 40 years through the extended period of operation. If the plant-specific operating experience indicates that the losses were underestimated, the new assumptions should be used in the analysis reflecting this experience. This analysis establishes the predicted lower limit (PLL) and the minimum required value (MRV) of the prestressing force at the end of the extended period of operation.

- Analysis of the effects of aging on the measured prestressing forces determined by trending the available data of the actual measured prestressing to the end of the extended period of operation if Option (ii) of 10 CFR 54.21(c)(1) is used for demonstrating that the trend line will stay above the PLL and MRV.

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

- Curves showing the projected measured prestressing forces (i.e., trend lines) vs. the minimum required prestressing forces in each major direction, with a short description of the method used to project the measured forces (for both units, if different).

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

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

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

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

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

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

4.5.3 FSAR Supplement

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

4.5.4 Conclusion

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

4.6 Containment Liner Plate Fatigue

4.6.1 Summary of the Technical Information in the Application

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

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

(1) 40 thermal cycles corresponding to 40 years of annual outdoor temperature variations, corresponding to the plant life of 40 years

(2) 500 thermal cycles corresponding to containment interior temperature variations during RCS heatup and cooldown

(3) One thermal cycle corresponding to the maximum hypothetical accident

(4) The containment liner plate piping penetrations are isolated from the piping system thermal loads by concentric sleeves. These sleeves were designed in accordance with the 1965 Edition of the ASME Section III fatigue considerations as subject to the thermal load cycles of the piping system.

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

The applicant evaluated the above fatigue conditions for the period of extended operation. For item (a), the applicant stated that the increase in the number of cycles from 40 to 60 is considered to be insignificant, since the containment is designed for 500 heatup/cooldown cycles. For item (b), the applicant stated that the assumed 500 thermal cycles was evaluated based on the more limiting number of 200 heatup/cooldown design transients for the RCS. An evaluation described in Section 4.3.1 of the application determined that the originally projected number of maximum RCS design cycles is conservative enough to envelop the projected cycles for the extended period of operation, and therefore the original containment liner plate fatigue analysis based on 500 heatup/cooldown cycles is considered valid for the period of extended operation. For item (c), the assumed value is considered to remain valid for 60 years of operation. For item (d), the applicant stated that the design of the containment penetrations meets the general requirements of the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III. The applicant identified the main steam piping, feedwater piping, blowdown piping, and letdown piping as the only piping penetrating the containment wall and the liner plate that contributes significant thermal loading on the liner plate. The applicant also stated that the projected number of actual operating cycles for these piping systems was determined to be less than the original design limits.

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

4.6.2 Staff Evaluation

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

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

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

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

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

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

4.6.3 FSAR Supplement

The applicant updated UFSAR Chapter 16, Section 16.3.5, "Containment Liner Plate Fatigue," to reflect the change in thermal cycling due to outdoor annual temperature variation from 40 cycles to 60 cycles of plant life operation. FPL also provided a discussion showing that the fatigue analysis of the containment liner plate and piping penetrations remains valid for the period of extended operation. The staff finds this acceptable.

4.6.4 Conclusion

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

4.7 Other Plant-Specific Time-Limited Aging Analyses

4.7.1 Bottom Mounted Instrumentation Thimble Tube Wear

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

4.7.1.1 Summary of Technical Information in the Application

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

4.7.1.2 Staff Evaluation

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

As described above, the TLAA on thimble tube thinning was based on results of two eddy current test inspections of the thimble tubes. These two inspections provided a gross status of tube thinning conditions at that time. It appears that the testing results were utilized to estimate wear rates. The wear rates were then used in the TLAA for justifying the adequacy of performing surveillance on a single thimble tube (N-05 in Unit 3) during the one-time inspection described in the thimble tube AMP.

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

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

4.7.1.3 FSAR Supplement

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

4.7.1.4 Conclusion

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

4.7.2 Emergency Containment Cooler Tube Wear

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

4.7.2.1 Summary of Technical Information in the Application

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

4.7.2.2 Staff Evaluation

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

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

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

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

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

4.7.2.3 Conclusion

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

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

The applicant addresses the TLAA evaluations performed to address thermal and mechanical fatigue analyses of plant mechanical components in Section 4.3 of the LRA. Other plant-specific TLAAs are addressed in Section 4.7 of the LRA.

4.7.3.1 Summary of Technical Information in the Application

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

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

The RCS primary loop piping LBB analysis has been projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

4.7.3.2 Staff Evaluation

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

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

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

4.7.3.3 FSAR Supplement

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

4.7.3.4 Conclusion

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

4.7.4 Crane Load Cycle Limits

4.7.4.1 Summary of Technical Information in Application

In Section 4.7.4 of the LRA, the applicant identified the crane load cycle limit as a TLAA for the cranes within the scope of license renewal. They include the polar cranes, reactor cavity manipulator cranes, spent fuel pool bridge cranes, spent fuel cask crane, turbine gantry crane, and intake structure bridge crane. The applicant stated that the spent fuel pool bridge cranes were analyzed for up to 200,000 cycles of maximum load. The other cranes in the scope of license renewal were analyzed for up to 2,000,000 cycles of maximum load based on the design codes utilized for these cranes. In addition, the applicant stated that for each crane, the actual usage over the projected life through the period of extended operation will be far less than the analyzed number of cycles. The applicant further stated that the analyses associated with crane design, including fatigue, remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.7.4.2 Staff Evaluation

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

4.7.4.3 FSAR Supplement

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

4.7.4.4 Conclusion

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

5. REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

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

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

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ACRS R-1992

April 19, 2002

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

Dear Chairman Meserve:

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

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

Recommendation and Conclusion

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

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

Background and Discussion

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

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

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

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

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

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

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

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

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

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

Sincerely,

/RA/

George E. Apostolakis
Chairman

References:

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

6. CONCLUSIONS

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

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

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

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APPENDIX A
CHRONOLOGY

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

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

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

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

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

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

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

APPENDIX B
REFERENCES

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

American Concrete Institute (ACI)

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

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

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

American Society of Mechanical Engineers (ASME)

ASME Boiler and Pressure Vessel Code Section III, 1965.

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

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

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APPENDIX C
ABBREVIATIONS

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

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

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

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11. ABSTRACT *(200 words or less)*

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

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

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

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license renewal, 10 CFR Part 54, license renewal rule, license renewal application (LRA), current licensing basis (CLB), aging effects, aging management review (AMR), aging management program (AMP), scoping, time limited aging analysis (TLAA), operating license, updated final safety analysis report or updated safety analysis report (UFSAR or USAR).

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300