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

FPL

L-2001-60
10 CFR 54

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

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

By letter dated February 2, 2001, the NRC requested additional information regarding the Turkey Point Units 3 and 4 License Renewal Application (LRA). Attachment 1 to this letter contains the responses to the Requests for Additional Information (RAIs) associated with Section 3.3, Engineered Safety Features Systems of the LRA.

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

Very truly yours,

R. J. Hovey
Vice President - Turkey Point

RJH/EAT/hlo

Attachment

A084

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

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

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

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

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

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

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

RJH

R. J. Hovey

Subscribed and sworn to before me this

30 day of March, 2001.

Olga Hanek

Olga Hanek
Name of Notary Public (Type or Print)

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

R. J. Hovey is personally known to me.

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

SECTION 3.3 ENGINEERED SAFETY FEATURES SYSTEMS

RAI 3.3-1:
In Section 3.3.2 of the LRA, the applicant states that the aging effect requiring management for carbon steel mechanical closure bolting is the loss of mechanical closure integrity. This aging effect, as listed in Tables 3.3-1 through 3.3-7, is managed by Boric Acid Wastage Surveillance Program described in Appendix B, Section 3.2.3. For closure bolting, loss of material, loss of preload, and crack initiation and growth due to stress corrosion cracking are applicable aging effects that are not described in the boric acid wastage surveillance program. Provide a discussion of how the Boric Acid Wastage Surveillance Program would manage these aging effects including acceptance criteria and specific operating experience. If this program does not adequately manage these aging effects then discuss the specific aging management program that will be used to manage the aging effects of the closure bolting.

FPL RESPONSE:
The LRA, Appendix B, Subsection 3.2.3, page B-44 states that the Boric Acid Wastage Surveillance Program manages the effects of loss of material and loss of mechanical closure integrity due to aggressive chemical attack. 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). Loss of bolting material can result in failure of the mechanical joint and thus, loss of a component's pressure boundary integrity. Therefore, the LRA addresses loss of mechanical closure integrity resulting from borated water leaks and credits the Boric Acid Wastage Surveillance Program (LRA, Appendix B, page B-44) for management of this effect on carbon and low alloy steel bolting. The acceptance criteria for this program is provided in LRA Appendix B, Subsection 3.2.3, page B-46.

LRA Appendix C, Section 5.4, page C-21 states that loss of pre-load of mechanical closures can occur due to settling of mating surfaces, relaxation after cyclic loading, gasket creep, and loss of gasket compression due to differential thermal expansion. The loss of pre-load due to these mechanisms can result in leakage at the joint, e.g. gasket or seal leakage, not failure of the mechanical joint. It is noted that the ASME Code does not consider gaskets, seals and O-rings to perform a pressure retaining function. It follows that the loss of pre-load from the above mechanisms does not result in loss of mechanical closure or loss of pressure boundary integrity. Therefore, no aging effects associated with loss of pre-load resulting from settling, relaxation after cyclic loading, gasket creep, and temperature effects are considered to require management during the period of extended operation.

High stress in conjunction with an aggressive environment can cause cracking of certain bolting materials due to stress corrosion cracking (SCC). As identified in NRC IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," and Generic Letter 91-17, "Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants," cracking of bolting in the industry has occurred due to SCC. These instances of SCC have been primarily attributed to the use of high yield strength bolting materials, excessive torquing of fasteners, and contaminants, such as the use of lubricants containing molybdenum disulfide (MoS_2). In response to NRC IE Bulletin 82-02, Turkey Point verified that:

(a) Specific maintenance procedures were in place to address bolted closures.

(b) The procedures in use addressed detensioning and retensioning practices and gasket installation and controls.

(c) Threaded fastener lubricants used in pressure boundary applications have specified maximum allowable limits for chloride and sulfur content to minimize susceptibility to SCC environments.

(d) Maintenance crew training on threaded fasteners is performed.

At Turkey Point, the potential for SCC of fasteners is minimized by utilizing ASTM A193, Gr. B7 bolting material and by limiting contaminants such as chlorides and sulfur in lubricants and sealant compounds. Additionally, sound maintenance bolt torquing practices are used to control bolting material stresses. These actions have been effective in eliminating the potential for SCC of bolting materials. The results of a review of the Turkey Point condition report (1992 through 2000) and metallurgical report (1986 through 2000) databases supports this conclusion in that no instances of bolting degradation due to SCC were identified. Therefore, cracking of bolting material due to SCC is not considered an aging effect requiring management at Turkey Point.

SECTION 3.3.2 CONTAINMENT SPRAY

RAI 3.3.2-1:
For all stainless steel pressure retaining components, such as, pumps, valves, piping, fittings, tubing, etc., in the containment spray system, where borated water is the internal environment, is crack initiation and growth due to stress corrosion cracking an applicable aging effect? It appears to be a possible effect according to some of the industry experience as listed in Section 3.3.3.1 of the Turkey Point license renewal application (LRA). Is there an inservice inspection aging management program in place to address this aging effect? If not, provide an explanation of why this aging effect is not addressed in the LRA or a discussion of the program that manages these aging effects.

FPL RESPONSE:
As discussed in LRA Appendix C, Section 5.2, page C-19, for austenitic stainless steels in treated water, the relevant conditions required for stress corrosion cracking (SCC) are the presence of halogens in excess of 150 ppb or sulfates in excess of 100 ppb, and elevated temperature. For Turkey Point treated water environments, a temperature criterion of greater than 140°F is utilized for susceptibility of austenitic stainless steels to SCC. Containment Spray (CS) operates at a temperature less than 140°F, therefore, cracking due to SCC is not an aging effect requiring management for CS components. This conclusion is supported by plant operating and maintenance experience.

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

RAI 3.3.3-1:
In Section 2.3.2.3 of the Turkey Point LRA, FPL states that all containment penetrations and associated containment isolation valves and components that ensure containment integrity, regardless of where they are described, require an aging management review. Provide a cross reference to sections in the LRA where aging effects and aging management programs for the containment isolation valves and associated piping are described and summarize the applicable aging effects and the programs used to manage them.

FPL RESPONSE:
The following table provides a list of systems/components having the intended function of containment isolation/containment integrity and identifies the corresponding table in the LRA that contains the aging effects/program information:

CONTAINMENT ISOLATION SYSTEMS and COMPONENTS	
SYSTEM	**LRA SECTION**
Auxiliary Feedwater and Condensate Storage	Table 3.5-3, pp 3.5-17 through 21
Breathing Air Systems	Table 3.3-3, p 3.3-17
Chemical & Volume Control	Table 3.4-4, pp 3.4-28 through 34
Component Cooling Water	Table 3.4-2, pp 3.4-19 through 23
Containment Post Accident Monitoring and Control	Table 3.3-7, pp 3.3-28 through 31
Containment Purge Systems	Table 3.3-3, p 3.3-16
Containment Spray	Table 3.3-2, pp 3.3-11 through 15
Emergency Containment Cooling (heat exchanger tubes)	Table 3.3-1, pp 3.3-9 through 10
Feedwater & Blowdown	Table 3.5-2, pp 3.5-11 through 16
Instrument Air	Table 3.4-8, pp 3.4-44 through 49
Main Steam and Turbine Generators	Table 3.5-1, pp 3.5-8 through 10
Nitrogen & Hydrogen Systems	Table 3.3-3, p 3.3-18
Normal Containment and Control Rod Drive Mechanism Cooling (heat exchanger tubes)	Table 3.4-9, pp 3.4-50 through 54
NSSS Sample System	Table 3.4-6, p 3.4-36 through 38
Primary Water Makeup	Table 3.4-5, p 3.4-35
Reactor Coolant Piping - Non Class 1 Components (RCP Bearing Oil Heat Exchanger Shell Side)	Table 3.2-1, pp 3.2-58 through 60

CONTAINMENT ISOLATION SYSTEMS and COMPONENTS (continued)	
SYSTEM	**LRA SECTION**
Reactor Coolant Pumps (Thermal Barrier Heat Exchanger Nozzles)	Table 3.2-1, pp 3.2-80 through 82
Regenerative and Excess Letdown Heat Exchangers (Excess Letdown shell side)	Table 3.2-1, pp 3.2-61 through 62
Residual Heat Removal	Table 3.3-5, pp 3.3-23 through 25
Safety Injection	Table 3.3-4, pp 3.3-19 through 22
Secondary Sample System	Table 3.4-6, pp 3.4-39 through 40
Steam Generators (shell side)	Table 3.2-1, pp 3.2-83 through 89
Waste Disposal	Table 3.4-7, p 3.4-41 through 43

Provided below is a summary of containment isolation/integrity component materials, aging effects requiring management, and credited aging management programs. For additional details see the applicable section of the LRA identified in the table above.

Typical containment isolation/integrity components are valves, piping/fittings, tubing/fittings, and mechanical closure bolting associated with these components. For specific components associated with containment penetrations see License Renewal Boundary drawings supplied with LRA, identified in Tables 2.3-1, 2.3-4, 2.3-5 and 2.3-6. (Note the steam generator and Component Cooling Water (CCW) are closed systems inside containment that are credited for containment integrity.) The materials of construction for these components are carbon steel, galvanized carbon steel, stainless steel, or copper alloys. The external environments include containment air, outdoor, or indoor - no air conditioning. The aging effects requiring management for external surfaces are as follows:

Carbon steel: loss of material and loss of mechanical closure integrity
Stainless steel: none
Copper alloy: loss of material
Galvanized carbon steel: none

The programs credited for managing these effects are:

System & Structures Monitoring Program
ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program- (shell side steam generator only)
Periodic Surveillance and Preventive Maintenance Program
Boric Acid Wastage Surveillance Program

The internal environments include air/gas[1] (including nitrogen, dry air and ambient air) and treated water (including borated, primary and secondary). The aging effects requiring management for internal surfaces are as follows:

Carbon steel: loss of material
Copper alloy: loss of material
Stainless steel: loss of material, cracking[2]

The programs credited for managing these effects are:

Chemistry Control Program
Emergency Containment Coolers Inspection
Galvanic Corrosion Susceptibility Inspection Program[3]
Periodic Surveillance and Preventive Maintenance Program
Flow Accelerated Corrosion Program.

Based on the evaluations provided in Appendix B for the programs listed above, aging effects are adequately managed so that the intended functions of containment isolation/integrity are maintained consistent with the current licensing basis for the period of extended operation.

Notes:
1. Stainless steel, and galvanized carbon steel have no aging effects requiring management in an internal environment of air/gas.
2. Stress corrosion cracking is an aging effect requiring management for austenitic stainless steel at temperatures >140°F.
3. Excess Letdown Heat Exchanger, Control Rod Drive Mechanism Coolers, Reactor Coolant Pump Bearing Oil Heat Exchangers, and Normal Containment Coolers only

RAI 3.3.3-2:

For containment isolation, Section 3.3.2 of the Turkey Point LRA, states that the aging effect requiring management is loss of material for carbon steel components and the aging effect requiring management for carbon steel mechanical closure bolting is loss of mechanical closure integrity, clarify if these aging effects apply to all containment isolation valves and associated piping or is limited to the ones described in this section.

FPL RESPONSE:

As described in LRA Subsection 2.3.2.3 (page 2.3-15), "Containment Isolation," Breathing Air, Nitrogen and Hydrogen, and Containment Purge are the process systems whose only license renewal intended function is containment isolation. LRA Subsection 3.3.2, "Aging Effects Requiring Management," Containment Isolation refers to the three above mentioned systems. For aging effects applicable to other system components that perform a containment isolation function, see response to RAI 3.3.3-1.

RAI 3.3.4-1:
In Table 3.3-4 of the Turkey Point LRA, FPL indicates that the SI pump thrust bearing coolers and SI shaft seal heat exchanger shells and covers are fabricated from cast iron materials. Volume Four of the American Welding Society (AWS) *Welding Handbook* indicates that welding of cast iron materials tends to result in the precipitation of carbides and the formation of high-carbon martensitic material in the heat affected zones immediately adjacent to the weld metal. Since both of these materials are extremely brittle, the AWS states that welded cast iron materials may be susceptible to spontaneous cracking or cracking while in service. FPL has not identified cracking as an aging effect that could effect these components during the periods of extended operation for the Turkey Point units. Clarify whether or not the SI pump thrust bearing coolers and SI shaft seal heat exchanger shells and covers have any welds in them. If these components do contain welded regions, cracking must be added as an aging effect that needs managing in these components during the periods of extended operation for the Turkey Point units.

FPL RESPONSE:
The Safety Injection (SI) pump thrust bearing coolers and SI shaft seal heat exchangers do not contain any welds. The thrust bearing coolers are castings with threaded ports to accommodate once-through cooling. The shaft seal heat exchangers have cast bodies with gasketed and bolted covers.

RAI 3.3.4-2:
In Section 3.3.2 of the TP LRA, FPL identifies that cracking of stainless steel and Inconel components in the SI system is an aging effect that requires management during the periods of extended operation for the TP units. However, this aging effect was not included in Table 3.3-4 for stainless steel and Inconel components in the SI system. Clarify whether or not the operating conditions for the SI system, when taken into account with the environmental conditions for the system, will allow FPL to preclude cracking as a potential aging effect for the stainless steel and Inconel SI components. If cracking is an applicable aging effect for the stainless steel and Inconel SI components, then Table 3.3-4 should be modified accordingly.

FPL RESPONSE:
For Safety Injection (SI), LRA Section 3.3.2, page 3.3-3, states "The aging effects requiring management are cracking for certain stainless steel components; and loss of material and fouling for Inconel heat exchanger tubing…." Table 3.3-4, "Safety Injection," page 3.3-22, component/commodity group "Piping/fittings, (large bore, thin wall outdoors, and outdoors in trenches)," lists cracking as an external aging effect requiring management. As described in Section 5.2 of Appendix C to the LRA, operating experience at FPL has demonstrated that SCC is a potential aging mechanism for the non-stress relieved heat affected zones of weld joints on the external surfaces of large bore, thin wall stainless steel piping in trenches and outdoors. Therefore, SCC is listed as an aging effect requiring management for these components. The Systems and Structures Monitoring Program is credited with managing this effect. Cracking is not an aging effect requiring management for any other stainless steel or Inconel SI components. As discussed in LRA Appendix C, Section 5.2 (page C-19), for austenitic stainless steels and nickel-based alloys in treated water, the relevant conditions required for SCC are the presence of halogens in excess of 150 ppb, or sulfates in excess of 100 ppb, and elevated temperatures (>140° F). The SI system is normally in standby at temperatures less than 140° F, therefore it is not susceptible to cracking. A review of plant operating and maintenance history supports this conclusion in that no instances of internal SCC in the SI system were identified. The statement in Section 3.3.2 does not identify cracking as an aging effect requiring management for Inconel in the SI system. LRA Table 3.3-4, page 3.3-19, correctly identifies loss of material and fouling as aging effects for the Inconel tubes of the SI pump shaft seal heat exchangers. The Chemistry Control Program is credited with managing these aging effects.

RAI 3.3.4-3:
In Section 3.3.2 and Table 3.3-4 of the TP LRA, FPL identifies that the SI pump shaft seal heat exchanger tube shields, which are fabricated from brass material, may be effected by loss of material. Since Section 5.2 of Appendix C does not discuss whether cracking of brass materials is an aging effect requiring management, discuss your basis for not including it as a plausible aging effect for the SI pump shaft seal heat exchanger tube shields.

FPL RESPONSE:
Cracking in tube shields of heat exchangers can be the result of flow-induced vibrational fatigue or stress corrosion cracking.

Cracking due to flow-induced vibration is precluded by proper design. Fatigue and vibration in general represent only about 5% of the total number of nuclear industry reported heat exchanger failures. The time span between the onset of high cycle vibration induced fatigue and the failure of the component is relatively short. If a particular heat exchanger in a plant is subject to flow induced vibration and fatigue, failures should have been already reported during the early part of the original license period of the plant. The operating history review performed as part of the aging management review for these heat exchangers did not identify any instances of cracking of the tube shields. Therefore, cracking in tube shields due to flow-induced vibrational fatigue is not an aging effect requiring management during the period of extended operation.

Copper alloys exhibit excellent resistance to stress corrosion cracking[1] in treated water, i.e., water that is free from any pollutants known to cause SCC, such as nitrates and ammonia, therefore this aging effect was not considered to require management for copper alloys.

Reference 1: "Corrosion of Metals in Marine Environments," J. A. Beavers, G. H. Koch, W. E. Berry, Metals and Ceramics Information Center Report, July 1986.

SECTION 3.3.5 RESIDUAL HEAT REMOVAL SYSTEM

RAI 3.3.5-1:
For those RHR components identified in Table 3.3-5 as being within the scope of license renewal and for whom cracking has been identified as a potential aging effect, justify why the chemistry control program is sufficient by itself as the basis for managing this aging effect.

FPL RESPONSE:
As listed in LRA Table 3.3-5, pages 3.3-23 and 3.3-24, cracking has been identified for certain stainless steel RHR components exposed to treated water - borated. As discussed in LRA Appendix C, Section 5.2, page C-19, for austenitic stainless steels in treated water, the relevant conditions required for stress corrosion cracking (SCC) are the presence of halogens in excess of 150 ppb or sulfates in excess of 100 ppb, and elevated temperature. For Turkey Point treated water environments, a temperature criterion of greater than 140°F is utilized for susceptibility of austenitic stainless steels to SCC. Portions of RHR operate at a temperature greater than 140°F during refueling outages. However, the chemistry of RHR water is controlled such that halogens and sulfates are maintained below 150 and 100 ppb respectively. The Chemistry Control program as described in LRA Appendix B, Subsection 3.2.4, page B-47 monitors halogens and sulfates in systems with an internal environment of treated water - borated to insure that they are maintained within specified limits. The effectiveness of the Chemistry Control Program in managing cracking for the RHR system is supported by a review of plant operating and maintenance history, including condition reports and metallurgical laboratory reports. This review indicated no instances of SCC or age related cracking in the RHR system components.