February 2, 2001

Mr. Thomas F. Plunkett
President - Nuclear Division
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE TURKEY POINT, UNITS 3 AND 4, LICENSE RENEWAL APPLICATION

Dear Mr. Plunkett:

By letter dated September 8, 2000, Florida Power and Light (FPL), submitted for the Nuclear Regulatory Commission's (NRC) review an application pursuant to 10 CFR Part 54, to renew the operating license for Turkey Point Nuclear Plant, Units 3 and 4. The NRC staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its safety review. Specifically, the enclosed questions relate to Section 3.2, "Reactor Coolant Systems."

Please provide a schedule by letter, electronic mail, or telephonically for the submittal of your responses within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with FPL prior to the submittal of the responses to provide clarifications of the staff's requests for additional information.

Sincerely,

/***RA***/

Rajender Auluck, Senior Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Program
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION
TURKEY POINT, UNITS 3 AND 4

Section 3.2 Reactor Coolant Systems

RAI 3.2-1: Identify all RCS components and subcomponents that are fabricated from either Alloy 600 base metal or weld metal fabricated using INCO 182/82, and are exposed to primary water. Describe the aging management programs used to manage the cracking due to stress corrosion cracking (SCC), in particular primary water SCC (PWSCC), in these items during the license renewal period, or provide the basis for not requiring management of this aging effect.

Section 3.2.1 Reactor Coolant Piping

RAI 3.2.1-1: Section 2.3.1.2.2 of the application indicates that the inner reactor vessel flange O-ring leak detection line tubing, fittings and valves, and the reactor vessel head vent piping, fittings and valves downstream of the restricting orifices are non-Class 1 reactor coolant components requiring aging management. Describe the environment of these components during an operational cycle, including refueling outages. Could these items have cyclic exposure to an aqueous environment followed by drying and resultant accumulation of corrosive products? If so, describe aging management of stress corrosion cracking for these items during the license renewal period.

Section 3.2.2 Regenerative and Excess Letdown Heat Exchangers

RAI 3.2.2-1: For the excess letdown heat exchangers, Section 3.2.2.2.2 of the LRA indicates that loss of material and loss of mechanical closure integrity of the bolting due to aggressive chemical attack will be managed by the boric acid wastage surveillance program. Describe any incidents to date where loss of material or mechanical closure integrity due to aggressive chemical attack have occurred for the excess letdown heat exchangers.

RAI 3.2.2-2: Appendix C, page C-22 of the LRA indicates that high yield stress materials and contaminants such as lubricants containing MoS_2 have caused cracking of bolting in the industry. Address how yield strength and elimination of contaminants will be addressed during the period of extended operation.

Section 3.2.3 Pressurizers

RAI 3.2.3-1: Discuss how the plant-specific water chemistry control programs provide for a sufficient level of hydrogen overpressure to support the conclusion in WCAP-14574 that hydrogen overpressure in the RCS will minimize the adverse effects of oxygen in the coolant and provide adequate protection against crevice corrosion in creviced geometries on the internal surfaces of the pressurizer.

ENCLOSURE

RAI 3.2.3-2: In order to take credit for the analysis in EPRI Report No. NP-5769 and the conclusion in WCAP-14574 that SCC is not an aging effect that needs to be managed for the SA193, Grade B7 low-alloy steel bolting materials, confirm that the yield strengths of record for the quenched and tempered, SA-193, Grade B7, low-alloy steel bolting materials in the Turkey Point pressurizers are within the range of 105-150 ksi.

RAI 3.2.3-3: In order to support the conclusion in WCAP-14574 that SCC would not be a problem in welded Type 304 stainless steel pressurizer supports if a reasonable justification could be made that the associated welds were not in the sensitized state, describe how the implementation of Turkey Point or FPL plant-specific procedures and quality assurance criteria for the welding and testing of austenitic welds, if any, provides a reasonable assurance that sensitization has not occurred in these welds.

RAI 3.2.3-4: In WCAP-14574, the WOG was not clear whether or not loss of material by erosion was a plausible aging effect for pressurizer surge nozzle thermal sleeves, surge nozzle safe ends, spray nozzle thermal sleeves, and spray nozzle safe ends in Westinghouse-designed plants. Analyze and discuss whether or not loss of material by erosion is a plausible aging effect for these components. If the analysis supports the conclusion that erosion is plausible within any of these components and that the corresponding components are within the scope of license renewal, modify Table 3.2-1 appropriately and propose an aging management program to manage this aging effect within the proposed extended operating terms for the Turkey Point units.

RAI 3.2.3-5: Propose an AMP to verify whether or not thermal fatigue-induced cracking in the pressurizer cladding has propagated through the clad into the ferritic base metal or weld metal materials beneath the clad.

Section 3.2.4 Reactor Vessels

RAI 3.2.4-1: In Table 3.2-1 of the LRA, the applicant indicated that cracking of the core support lugs will be managed by the Chemistry Control Program and ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection (ISI) Programs (Examination Category B-N-2). The staff does not believe that the VT-3 examinations are sufficient to detect cracking of the core support lugs. Therefore, the staff requests that the applicant provide details of a plant specific aging management program to detect cracking of the core support lugs.

Section 3.2.5 Reactor Vessel Internals

RAI 3.2.5-1: In Section 3.2.5, FPL indicates that the Turkey Point RVI components with fluence greater than 10^{21} n/cm^2 do not include the lower support casting. The staff requests that FPL provide the maximum fluence expected for the lower support casting during the extended period of operation and the basis for that expectation.

RAI 3.2.5-2: The baffle assembly contains three categories of baffle bolts that are designated as, former/baffle bolts, barrel former/bolts and baffle/baffle bolts. The staff requests that FPL clarify or provide the basis for not including the baffle/baffle bolts in the baffle assembly bolting described in Subsections 3.2.5.2.2, 3.2.5.2.4 and Table3.2-1.

<u>RAI 3.2.5-3</u>: In Subsection 3.2.5.2.1, FPL indicates that; susceptibility has been observed at fluences as low as $1x10^{21}$ n/cm^2 in laboratory studies on Type 304 stainless steel in PWR environments, Type 316 stainless steel is less susceptible, and field information suggests that greater exposures are required for the development of susceptibility. The staff requests that FPL identify the field information that suggests that greater exposures are required for the development of susceptibility.

<u>RAI 3.2.5-4</u>: In Subsection 3.2.5.2.4, FPL indicates, in part, that significant data, information and industry experience relative to the aging of baffle bolting is provided in WCAP-14577 (Reference 3.2-7) and is not duplicated in the Subsection. Reference 3.2-7 is WCAP-14577, " License Renewal Evaluation: Aging Management for Reactor Internals," Revision 0, dated June 1997. This edition of WCAP-14577 predates NRC Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former bolts in Foreign Plants" and the WOG activities that developed the significant data, information and industry experience with regard to the baffle bolt cracking issue. Subsequent to the topical report submittal, the WOG had periodic meetings and interactions with the staff to present information, data and industry experience with regard to its ongoing baffle bolt program. The WOG program and activities included the development of an analytical methodology for minimum baffle bolting distributions under faulted conditions and plant baffle bolting inspections and bolting replacement activities.

The staff issued several RAIs (by letter from Raj K. Anand (NRC) to Roger A, Newton (WOG) dated June 14, 1999) with regard to updating WCAP-14577 Revision 0 and the WOG's plans for use of the results of the technical progress, the WOG's and licensee's commitments to participation and use of the industry's PWR Materials Reliability Project (MPR) initiatives with regard to the RVI aging management issues, conclusions and recommendations. WOG responses to the RAIs are contained in a letter from Roger A Newton (WOG) to Raj K. Anand (NRC) dated November 24, 1999

The staff requests that FPL review the staff RAIs, the associated WOG responses, and address the RAIs and their applicability and inclusion with regard to the Turkey Point Units 3 and 4 license renewal application. In addition, FPL&L is requested to provide responses to each of the renewal applicant action items provided in the final safety evaluation report issued by the staff for WCAP-14577 (these are repeated below for convenience):

Renewal Applicant Action Items from FSER for WCAP-14577

(1) To ensure applicability of the results and conclusions of WCAP-14577 to the applicant's plant(s), the license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant must commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the aging management programs described in this topical report as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, must be identified by the renewal applicant

and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

(2) A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs must be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).

(3) For the holddown spring, applicants for license renewal are expected to address intended function, aging management review, and appropriate aging management program(s).

(4) The license renewal applicant must address aging management review, and appropriate aging management program(s), for guide tube support pins.

(5) The license renewal applicant must explicitly identify the materials of fabrication of each of the components within the scope of the topical report. The applicable aging effect should be reviewed for each component based on the materials of fabrication and the environment.

(6) The license renewal applicant must describe its aging management plans for loss of fracture toughness in cast austenitic stainless steel RVI components, considering the synergistic effects of thermal aging and neutron irradiation embrittlement in reducing the fracture toughness of these components.

(7) The license renewal applicant must describe its aging management plans for void swelling during the license renewal period.

(8) Applicants for license renewal must describe how each plant-specific AMP addresses the following elements: (1) scope of the program, (2) preventative 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.

(9) The license renewal applicant must address plant-specific plans for management of cracking (and loss of fracture toughness) of RVI components, including any plans for augmented inspection activities.

(10) The license renewal applicant must address plant-specific plans for management of age-related degradation of baffle/former and barrel/former bolting, including any plans for augmented inspection activities.

(11) The license renewal applicant must address the TLAA of fatigue on a plant-specific basis.

RAI 3.2.5-5: In Subsection 3.2.5.2.6, FPL discusses RVI material dimensional changes and cites references that indicate the material may be subject to various levels of dimensional changes resulting from void swelling under certain conditions. One reference cited in the discussion concludes that at the approximate RVI end-of-life dose of 100 displacements per

atom, swelling would be less than 2% at irradiation temperatures between 572°F and 752°F. In the discussion FPL indicates that, field service experience in PWR plants have not shown any evidence of swelling, and at present there have been no indications from the different reactor vessel internals bolt removal programs, or from any of the other inspection and functional evaluations (e.g., refueling), that there are any discernible effects attributable to swelling.

The staff requests that FPL identify specific examples of field service experience, bolt removal programs, and other inspections and functional evaluations with detailed descriptions of the examinations, inspections and evaluations that have been performed to support the conclusion that there is not any evidence of, or any discernible effects attributable to swelling.

The staff requests that FPL describe the change in loading on the baffle bolt and its impact on the bolt integrity that would occur if the thickness of the baffle material located under the bolt head were subjected to a 2% or less dimensional change due to swelling.

RAI 3.2.5-6: The application uses 1 x 10^{21} n/cm^2 (E > 0.1 MeV) as a fluence threshhold for neutron embrittlement of stainless steel used to fabricate internal components. Provide data to support this position, or revise the application to expand the list of potentially susceptible components to include those at lower fluences.

RAI 3.2.5-7: In Section 3.2.5 of the application, FPL states that, "Turkey Point's TLAA identification effort also identified fatigue as the only TLAA applicable to the reactor vessel internals. Fatigue of the reactor vessel internals is addressed in Subsection 4.3.1."

The staff requests that FPL provide a list of the TLAAs associated with fatigue used in verifying that the structural integrity of the reactor vessel internals were evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Section 3.2.7 Steam Generators

RAI 3.2.7-1: In Section 3.2.7.2.2 (Loss of Material) of the license renewal application, the aging mechanisms that can cause loss of material for the steam generators are listed. However, industry operating experience indicated that erosion (aging mechanism) could cause the loss of section thickness (aging effect) of a component, and this aging effect is not addressed in the application. One example of this aging effect is the loss of section thickness of the feedwater impingement plate supports in the Harris Nuclear Plant steam generators. Provide the plant specific aging management program for this aging effect in general for the steam generators and other components in the plant within the scope of license renewal for the period of extended operation.

RAI 3.2.7-2: Table 3.2-1 states that the applicable AMP for steam generator internals (e.g., the steam generator tube support plates) is the chemistry control program. However, FPL's response to Generic Letter (GL) 97-06, "Degradation of Steam Generator Internals," indicates that significantly more activities are undertaken to manage the potential degradation of steam generator internal components. In addition, the scope of the steam generator integrity program AMP includes steam generator secondary-side integrity inspections. Identify any additional AMPs (e.g., steam generator integrity program) that are applicable to steam generator internals in Table 3.2-1.

<u>RAI 3.2.7-3</u>: FPL identified a "loss of mechanical closure integrity" as the aging effect requiring management for primary bolting. Section 3.2.7.2.3 of the LRA identifies stress relaxation and/or aggressive chemical attack as two potential causes of a loss of mechanical closure integrity. However, industry operating experience indicates that a loss of mechanical closure integrity can also result from stress corrosion cracking.

> A) Section 5.4 of Appendix C of the LRA discusses the "loss of mechanical closure integrity" aging effect. The last paragraph of section 5.4 briefly discusses stress corrosion cracking. Describe, more thoroughly, the actions taken by FPL (e.g., the use of non-susceptible material and/or the use of non-aggressive lubricants) to prevent stress corrosion cracking in primary bolting. Operating experience has shown that some alloy steels with lower yield strengths are susceptible to stress corrosion cracking. Identify the range of yield strengths used at Turkey Point Units 3 and 4 and the susceptibility of those material strengths.
>
> B) Several NRC generic communications (e.g., NRC IE Bulletin 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants" and NRC Generic Letter 91-17, "Bolting Degradation or Failure in Nuclear Power Plants") provide information on industry operating experience associated with the degradation of primary bolting, but are not referenced by FPL in Section 3.2.7.3.1 of the LRA. Explain why these generic communications were not identified as reference documents and whether the information contained within was assessed for Turkey Point Units 3 and 4.

<u>RAI 3.2.7-4</u>: Table 3.2-1 lists the ASME Section XI, Subsections IWB, IWC and IWD Inservice Inspection Program as an AMP for steam generator U-tubes. ASME Examination Categories B-Q and B-P are identified as applicable to U-tubes. Explain the type of inspections associated with examination categories B-P and B-Q that are applicable to steam generator U-tubes, plugs and sleeves (if installed in the future).

<u>RAI 3.2.7-5</u>: NRC Information Notice (IN) 97-88, "Experiences During Recent Steam Generator Inspections," was not identified as a reference in Section 3.2.7.3.1 of the LRA. Discuss why the IN was not listed as a reference for the Turkey Point Units 3 and 4 LRA.

<u>RAI 3.2.7-6</u>: Feedwater nozzle safe ends and steam outlet nozzle safe ends were not identified in Table 3.2-1 as components requiring an aging management program. Explain why they were not identified.

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