December 16, 2002

Mr. J. A. Stall
Senior Vice President, Nuclear and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: TURKEY POINT PLANT, UNIT 4 - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK-INFORMED INSERVICE INSPECTION RELIEF REQUEST (TAC NO. MB5551)

Dear Mr. Stall:

By letter dated July 8, 2002, Florida Power and Light Company requested relief from the Inservice Inspection (ISI) requirements specified in the American Society of Mechanical Engineers Code. On October 23-24, 2002, the U.S. Nuclear Regulatory Commission (NRC) staff performed an audit of the probabilistic risk assessment analyses used to support the risk-informed ISI relief request. At the conclusion of the audit on October 24, 2002, the NRC staff discussed three questions that were identified during the audit that will require supplemental information to resolve.

Based on our review of your submittal and questions generated during the October 2002 audit, the NRC staff finds that a response to the enclosed request for additional information is needed before we can complete the review.

This request was discussed with your staff on December 16, 2002, and it was agreed that a response would be provided within 30 days of the issuance of this letter.

If you have any questions, please contact me at (301) 415-2315.

Sincerely,

/RA/

Eva A. Brown, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-251

Enclosure: Request for Additional Information

cc w/encl: See next page

December 16, 2002

Mr. J. A. Stall
Senior Vice President, Nuclear and
Chief Nuclear Officer
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: TURKEY POINT PLANT, UNIT 4 - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK-INFORMED INSERVICE INSPECTION RELIEF REQUEST (TAC NO. MB5551)

Dear Mr. Stall:

By letter dated July 8, 2002, Florida Power and Light Company requested relief from the Inservice Inspection (ISI) requirements specified in the American Society of Mechanical Engineers Code. On October 23-24, 2002, the U.S. Nuclear Regulatory Commission (NRC) staff performed an audit of the probabilistic risk assessment analyses used to support the risk-informed ISI relief request. At the conclusion of the audit on October 24, 2002, the NRC staff discussed three questions that were identified during the audit that will require supplemental information to resolve.

Based on our review of your submittal and questions generated during the October 2002 audit, the NRC staff finds that a response to the enclosed request for additional information is needed before we can complete the review.

This request was discussed with your staff on December 16, 2002, and it was agreed that a response would be provided within 30 days of the issuance of this letter.

If you have any questions, please contact me at (301) 415-2315.

Sincerely,
/RA/

Eva A. Brown, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-251

Enclosure: Request for Additional Information

cc w/encl: See next page

Distribution:

PUBLIC	EBrown	AHowe	BClayton (hard copy)	MRubin
PDII-2 (R/F)	SDinsmore	SMalik	DHarrison	RPalla
OGC	ACRS	AKeim	TChan	

ADAMS Accession No. ML023500398 *See Previous concurrence

OFFICE	PDII-2\PM	PDII-2\LA	SPSB\SC	EMCB\SC	PDII-2\SC
NAME	EBrown	BClayton	MRubin	TChan	AHowe
DATE	12/16/2002	12/16/2002	12/12/2002*	11/26 /02*	12/16/2002

OFFICIAL RECORD COPY

cc:
M. S. Ross, Attorney
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. John P. McElwain, Site Vice President
Turkey Point Nuclear Plant
Florida Power and Light Company
9760 SW. 344th Street
Florida City, FL 33035

County Manager
Miami-Dade County
111 NW 1 Street, 29th Floor
Miami, Florida 33128

Senior Resident Inspector
Turkey Point Nuclear Plant
U.S. Nuclear Regulatory Commission
9762 SW. 344th Street
Florida City, Florida 33035

Mr. William A. Passetti, Chief
Department of Health
Bureau of Radiation Control
2020 Capital Circle, SE, Bin #C21
Tallahassee, Florida 32399-1741

Mr. Craig Fugate, Director
Division of Emergency Preparedness
Department of Community Affairs
2740 Centerview Drive
Tallahassee, Florida 32399-2100

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

T. O. Jones, Plant General Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
9760 SW. 344th Street
Florida City, FL 33035

Walter Parker
Licensing Manager
Turkey Point Nuclear Plant
9760 SW 344th Street
Florida City, FL 33035

Mr. Don Mothena
Manager, Nuclear Plant Support Services
P.O. Box 14000
Juno Beach, FL 33408-0420

Mr. Rajiv S. Kundalkar
Vice President - Nuclear Engineering
Florida Power & Light Company
P.O. Box 14000
Juno Beach, FL 33408-0420

REQUEST FOR ADDITIONAL INFORMATION

RISK-INFORMED INSERVICE INSPECTION (RI-ISI) RELIEF REQUEST

TURKEY POINT PLANT UNIT 4

DOCKET NO. 50-251

1. Table 3.1-1 includes a note which states that 22 segments are categorized as “not used” for Unit 4. Explain what is meant by “not used” and why the segments are categorized this way.

2. Table 3.7-1 indicates that none of the segments with a risk reduction worth less than 1.005 were defined as high safety significant (HSS). Please describe the characteristics of Unit 4 and the RI-ISI evaluation that caused the expert panel to be satisfied that all HSS segments were identified by the quantitative calculations.

3. Are there any piping segments that include piping of a different diameter? If so how were the failure frequencies estimated for these segments? For segments including piping of a different diameter where the number of inspection locations was determined using the Perdue method, how were the number of locations to be inspected determined? How does the methodology for determining the failure frequency comport with the methodology described on page 71 of the Topical Report Westinghouse Commercial Atomic Power (WCAP) Report, WCAP-14572, Revision 2-NP-A? How does the methodology for determining the number of inspections comport with the methodology described on pages 170, 171, and 174 of the WCAP?

4. Will the proposed RI-ISI program be implemented during the current third 10-year ISI interval? What interval and period will the program be implemented?

5. How will the RI-ISI program be implemented?

6. For the current interval, how much of the American Society of Mechanical Engineers (ASME) Code ISI program has been completed? How much will be covered by the RI-ISI program? How many RI-ISI examinations will be performed?

7. Will the RI-ISI program be updated every 10 years and submitted to the U.S. Nuclear Regulatory Commission (NRC) consistent with the current ASME Section XI requirements?

8. Under what conditions will the RI-ISI program be resubmitted to the NRC before the end of any 10-year ISI interval?

9. Version 0 of the Probability Risk Assessment was used to support the Unit 4 RI-ISI submittal, and credited the use of the Unit 3 reactor water storage tank (RWST) as a back-up water supply to the Unit 4 RWST. This mode of operation was credited for small-small loss of coolant accidents (LOCAs) (3/8 to 2 inches) and for small LOCAs

Enclosure

(2 to 6 inches). A probability of about 2E-4/demand was used as the probability that the operators would fail to properly align the Unit 3 RWST.

The refill of the RWST using water sources and paths other than the Unit 3 RWST is included in the emergency operating procedures. Version 0 did not, however, credit these other sources of water. Use of the Unit 3 RWST to refill the Unit 4 RWST is implied but not defined in the site's severe accident management guidelines (SAGs). The SAGs are to be initiated when the core exit temperature is high enough to indicate that core damage has begun. The SAGs only include identification of all potential water sources (of which the Unit 3 RWST is one) and then directs the operator to provide sufficient cooling water to the reactor vessel. The individual steps of the process are left to the operator.

The "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Application," ASME RA-S-2002, April 5, 2002, allows crediting, "those actions performed by the control room staff either in response to procedural direction or as skill-of-the craft to recover a failed function, system or component that is used in the performance of a response action in dominant sequences (e.g., manual start of a standby pump following failure of auto-start)." As a result, the failure probability of 2E-4/demand for the non-proceduralized use of the Unit 3 RWST to refill the Unit 4 RWST in order to prevent core damage following a small-small and a small LOCA is in question.

Please re-evaluate the modeling of the RWST refill, and determine the impact of the re-evaluation on the proposed RI-ISI program. Provide the specific changes made to the model to credit the refilling of the RWST and submit sufficient information to allow the staff to review the changes. The submitted information should include a description of each RWST refill source and associated path (including all human actions) credited in the evaluation. For each RWST refill source and for each human action provide, as appropriate, the success criteria, the logic models, the input values, and the quantitative results. Identify and discuss the impact of these changes on the conditional core damage probabilities (CCDP) used to support the RI-ISI relief request, and on the selection of inspection locations in the proposed RI-ISI program.

10. LOCA size definitions are defined based on the functional requirements that would prevent core damage for the given rate of primary coolant loss. Florida Power & Light (FPL) defines four LOCA sizes whereas other licensees normally only define three. The Unit 4 small-small LOCA corresponds to the size that other licensees label as a small LOCA. For Unit 4, the small LOCA corresponds to the size that other licensees label medium LOCA. Other licensees label all LOCAs with a 6 inch or greater (equivalent) diameter as a large LOCA.

    For Unit 4, FPL divides LOCAs with a 6 inch or greater, diameter into a medium LOCA between 6 and 13 ½ inches, and a large LOCA greater than 13 ½ inches. FPL stated that the introduction of a medium LOCA between 6 and 13 ½ inches is necessary because thermal-hydraulic evaluations indicate that at least one train of high-pressure injection is required for LOCAs in this size range; other licensees' success criteria require only low-pressure injection for all LOCAs greater than 6 inches. It was noted by FPL that this was more "conservative" than requiring only low-head injection for LOCA sizes between 6 and 13 ½ inches.

During the audit, the staff observed in the RI-ISI submittal documentation that the CCDP for the medium LOCA (about 0.03/demand) was about five times smaller than the CCDP for the large LOCA (about 0.15/demand). The staff further noted that the peer review final report stated that the peer reviewers could not locate the thermal-hydraulic analyses supporting introduction of a fourth LOCA size for Unit 4. Is the CCDP for the medium LOCA smaller than for the large LOCA? If so, explain why the medium LOCA (that apparently requires more equipment to operate) has a smaller CCDP than the large LOCA. Also provide the criteria used to identify core damage initiation. Include a discussion of the thermal-hydraulic analyses and the results justifying the introduction of an additional LOCA size between 6 and 13 ½ inches. The justification should include results from the bounding size compared with the criteria used to identify core damage initiation.

11. During the audit, a description of the individual changes to the probabilistic risk assessment was provided in the documentation. Some contradictory statements were found regarding whether one or two high pressure injection trains were modeled as success criteria for a small-small LOCA.

    Identify the appropriate success criteria for high-pressure injection following a small-small LOCA and confirm that this success criteria is accurately reflected in the final proposed RI-ISI program.

12. Section 3.8 of the licensee's submittal addresses additional examinations. It states, "The evaluation will include whether other elements on the segment or segments are subject to the same root cause and degradation mechanism. Additional examinations will be performed on these elements up to a number equivalent to the number of elements initially required to be inspected on the segment or segments. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same service related root cause conditions or degradation mechanism."

    ASME Code directs licensee's to perform these sample expansions in the current outage that the flaws or relevant conditions were identified. Verify in what time frame the sample expansions will be completed.