FPL

January 15, 2003
10 CFR 50.55 (a)

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

Re: Turkey Point Unit 4
Docket Nos. 50-251
Response to Request for Additional Information Regarding Risk-Informed Inservice Inspection (RI-ISI) Relief Request for Turkey Point Unit 4

By letter L-2002-022, dated July 8, 2002, Florida Power & Light (FPL) submitted a request to revise the Turkey Point Unit 4 Inservice Inspection Program for Class 1 piping. By letter dated December 16, 2002, the U.S. Nuclear Regulatory Commission Staff requested additional information regarding the above referenced FPL submittal.

The response to the request for Additional Information is attached.

If you have any questions on this request, please contact Walter Parker at (305) 246-6632.

Sincerely,

J.P. McElwain
Vice President
Turkey Point Plant

SM

Attachment

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point

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.

   Response to Question # 1

   The 22 segments identified as "not used" were in reference to Unit 3 segment numbers. These segment numbers were not used in Unit 4; new segment numbers were used. Where the segment definitions for both units were identified as having significant differences, these numbers were not utilized e.g. Unit 3 location of the RHR suction line is Loop A versus Unit 4 location of the RHR suction line is Loop C. Section 6, Summary of results and conclusions of the RI-ISI submittal provides a detailed summary of segment differences, between the two units.

2. Table 3.7-1 indicates that none of the segments with a risk reduction worth <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.

   Response to Question # 2

   The methodology identified in WCAP-14572 Revision 1-NP-A was used for determining the safety significance of the segments in the Turkey Point Unit 4 RI-ISI program. Both quantitative results and deterministic considerations were used in determining the safety significance of the segments. For the segments which were quantitatively ranked low risk reduction worths (RRW < 1.001) or medium (RRW < 1.005 and $\geq$ 1.001), there were no quantitative results or deterministic reasons for making these segments high safety significant. As an additional consideration, the Turkey Point Unit 4 Expert Panel compared the ranking of segments in Turkey Point Unit 4 with similar segments in Turkey Point Unit 3 to assure consistency in the categorization of segments between the two units.

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 were 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?

Response to Question # 3

The Turkey Point Unit 4 Class 1 RI-ISI program has no segments that include piping of different diameters.

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?

Response to Question # 4

Yes. The program will be implemented in the 3rd period of the 3rd interval.

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

Response to Question # 5

The population of B-F and B-J welds that will be included under the risk informed program will be subdivided into three periods, with the third period examinations of the new program scheduled to be performed to close out the current third ten year interval.

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?

Response to Question # 6

Turkey Point Unit 4 has completed the examinations scheduled for the second period of the third ten year interval with 55.5% and 57.1% of the Class 1 B-F and B-J welds, respectively. The maximum percentage that will be credited for the risk informed program during the third period will be 34% of the B-F and B-J risk informed population. A minimum of 18 examinations will be performed during the third period of the third interval.

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?

Response to Question # 7

Yes. The RI-ISI program will be updated every 10 years and it will be 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?

Response to Question # 8

Changes to the RI-ISI program, prior to the ten year interval, will be resubmitted following the guidance of WCAP-14572, Revision 1-NP-A. During the monitoring process, on a period basis, plant design changes, plant procedure changes, equipment performance changes, and examination results (flaws, or leaks) will be factored into the risk informed program, as appropriate. If these changes decrease the percentage of examinations required for the 10-year interval under the proposed RI-ISI program, the revised program would then be resubmitted to the NRC for review and approval.

9. Version 0 of the PRA 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 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 (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.

Response to Question # 9

The text of this question represents the modeling of the recovery of using the opposite unit RWST for injection as a re-alignment of the Unit 4 RWST to refill the Unit 3 RWST. This is not an appropriate description of the recovery. The HHSI systems for the Turkey Point Units are cross-tied. When, for example, Unit 3 experiences a small LOCA, all 4 HHSI pumps (two Unit 3 pumps and two Unit 4 pumps) start and inject to the affected unit. Once it is determined that the pumps are running, the opposite unit pumps (in this example, the Unit 4 HHSI pumps) are stopped. Later in the sequence, when the RWST is nearing depletion and attempts to implement recirculation and RWST refill from other sources have failed, the operating crew merely has to start a Unit 4 HHSI pump to begin adding water from the Unit 4 RWST. This is a single action from the control room. No re-alignment is necessary. In the version of the model that was used to calculate the segment risk importances for the Unit 4 RI-ISI submittal, this human action had a probability of 2E-4. In the latest revised model, which uses a more current human reliability analysis algorithm, this human action is assigned a failure probability of 5E-2. The details of this recovery action are given below.

Following discovery of a LOCA, the operating crew proceeds to EOP-E-1. Step 16 of EOP-E-1 directs the crew to verify cold leg recirculation capability. If it cannot be verified, the crew is transferred to EOP-ECA-1.1, Loss of Emergency Coolant Recirculation. If recirculation capability is verified, the operator continues on in the procedure. At Step 23, if RWST level is less than 155,000 gallons, the operator is transferred to EOP-ES-1.3, Transfer to Cold Leg Recirculation. If recirculation cannot be established using this procedure, the operator is transferred to EOP-ECA-1.1. Therefore, at the latest, the operator will transferred to EOP-ECA-1.1 when the RWST reaches the 155,000 gallon level, and, at the earliest, at Step 16 of EOP-E-1.

Step 2 of EOP-ECA-1.1 instructs the operator to add makeup to the RWST. Step 7 of ECA-1.1 directs the operator to Step 31 if RWST level is less than 60,000 gallons. Steps 31 and beyond are concerned with various things such as charging flow, depressurization, accumulator inventory injection, etc., but they do not mention HHSI injection using the opposite unit's RWST. At Step 40, b., the operator is told to consult the TSC staff to determine if the RHR system should be placed in service. At Step 43, the operator is simply told to consult the TSC staff. At one of these points, it is judged that the TSC personnel will be very concerned with the impending RWST depletion, and will be considering the possibility of using the opposite unit's RWST for injection. Discussions with different operators and operating instructors at Turkey Point support this judgment.

When the RWST level reaches 60,000 gallons, the RWST lo-lo-level annunciator alarms. Annunciator Response Procedure ARP-097.CR operator actions for this alarm direct the operator to verify that the SI pumps are aligned to take suction from the opposite unit's RWST if EOPs are no longer in use. At this point, the operators should be out of the EOPs.

If the operators do not use the opposite unit RWST for injection at this time, and no functions have been restored, the TSC personnel use the Diagnostic Flow Chart (DFC) to determine which Severe Accident Guideline to use. If the core temperature is above 700 degrees F, the TSC personnel are directed to SAG-3, where Step 1 lists all of the different equipment which can be used to inject coolant to the RCS. The opposite unit's RWST is listed as one of these sources.

Since this is an accident sequence of long duration (it takes many hours for the RWST to deplete), the TSC and EOF will have been staffed and activated for some time prior to RWST depletion. The issue of the viability of the continuation of injection using the opposite unit's RWST was discussed with the Turkey Point Plant General Manager, who also serves as one of the Emergency Coordinators in the Emergency Plan. He said that the operators or the nuclear shift supervisor would not hesitate to use the opposite unit's RWST when it became apparent that EOP measures were not working. They would not wait until the in-core thermocouples reached a temperature of 700 degrees F. The TSC and EOF personnel would also not hesitate to recommend the action given the same circumstances.

Despite the viability of using the opposite unit's RWST in this situation, this recovery was removed from the model; RWST refill using the method described in the EOPs (using the primary water and boric acid systems to serve as the source of borated water) was added to the model; and the RI-ISI cases were re-run to determine what the effect would be on the RI-ISI calculations. The details of the RWST refill recovery are given below.

In the event of a small-small LOCA, the safety injection system would initially take suction from the refueling water storage tank (RWST). At some point before the RWST inventory was depleted, it would be necessary to establish a mode of long-term cooling. This could be either by cooling down and entering normal shutdown cooling via the residual heat removal (RHR) system, or by establishing recirculation from the containment sump. If neither option were available (e.g., because the RHR system, which is needed for both, is unavailable), the operators could extend the time essentially indefinitely by making up to the RWST.

The initial response to a small-small LOCA would entail proceeding through EOP-E-0, the emergency operating procedure (EOP) for a reactor trip or safety injection. Step 29 of this procedure requires assessment of RCS conditions, and eventually instructs the operators to transfer to the EOP for LOCAs, EOP-E-1. At step 16 of this procedure, the operators would be instructed to verify that equipment needed to support cold-leg recirculation was available. If the capability to accomplish recirculation could not be verified, the operators would then be directed to EOP-ECA-1.1, the procedure for loss of coolant recirculation.

The first step of this procedure would require further investigation of the availability of cold-leg recirculation. The second step would require the initiation of makeup to the RWST to extend the availability of suction to the safety injection pumps.

For a small-small LOCA, the RWST should provide adequate suction for at least 10 hours. Based on input from operator training instructors, it is estimated that the initial procedural response would be completed and the point at which the capability for cold-leg recirculation would be checked would be reached within about 20 minutes. It is estimated that another 5 minutes might be spent in investigating the availability of cold-leg recirculation before the decision was made to implement makeup to the RWST. The execution time (i.e., to align the makeup system (primary water and boric acid) and initiate proper flow) could be as long as 10 minutes, including local actions. The failure probability calculated for this human action is 2.1E-4. RWST refill was added to the model as an additional success path for long-term cooling for small-small LOCAs. One of two primary water pumps and one of two boric acid pumps were modeled as being required for successful RWST refill. Operator failure to implement RWST refill and flow path valve failures were included as single order failure paths for the RWST refill recovery.

As mentioned earlier, despite the strong arguments for the operator's use of the opposite unit's RWST, the recovery was removed from the model; RWST refill using the method described in the EOPs was added to the model; and the cases re-run to determine what the effect would be on the RI-ISI calculations. The CDFs/CDPs and LERFs/LERPs from these calculations were input to the Excel spreadsheets used to calculate the risk reduction worths (RRW) for each piping segment in the RI-ISI submittal (CDF/LERF, with and without operator action). These results were then compared to the results from the original submittal. It was found that there was little change. The segments are given different risk categories based on the RRW: low (RRW < 1.001), medium (1.001 ≤ RRW < 1.005), and high (RRW ≥ 1.005). None of the segments changed from a low or medium risk category to a high risk category. Seven segments changed from a low to a medium risk category, but the increases in RRW were small, from 1.000 to 1.001 or 1.002. A few segments dropped to a lower risk category. While the relative order of importance changed somewhat, the population of segments with a RRW ≥ 1.005 was the same.

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 Turkey Point Unit 4 (TP4) 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 1/2 inches, and a large LOCA greater than 13 1/2 inches. FPL stated that the introduction of a medium LOCA between 6 and 13 1/2 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 1/2 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 your thermal-hydraulic analyses and the results justifying the introduction of an additional LOCA size between 6 and 13 1/2 inches. The justification should include results from the bounding size compared with the criteria used to identify core damage initiation.

Response to Question #10

The CCDP for a medium LOCA is less than the CCDP for a large LOCA because the CCDP for both initiators is dominated by the sequence where the operating crew fails to successfully implement low pressure recirculation. There is considerably less time to perform this operation for a large LOCA than there is for a medium LOCA. Therefore, the probability of the operating crew failing to perform the action is significantly higher.

The criteria for core damage for these sequences was core uncovery. Thermal-hydraulic calculations for the Turkey Point success criteria for medium LOCAs show that 2 accumulators, 1 HHSI train, and 1LHSI train are needed for mitigation of LOCA sizes of 6 and 8 inches in diameter.

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.

Response to Question #11

The appropriate success criterion for high pressure injection following a small-small LOCA is one HHSI train. This reflects the findings of the thermal-hydraulic calculations and is modeled this way in the Turkey Point PSA model used for the RI-ISI calculations.

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.

Response to Question #12

WCAP-14572 Revision 1-NP-A, Section 4 Inspection Program Requirements, Subsection 4.5, Implementation and Program Monitoring, Paragraph 4.5.3 Use of Corrective Action Programs, states " Acceptable Sampling such as those required in ASME Section XI under IWB-2430 shall be used." Turkey Point Unit 4 Risk Informed Inservice Inspection Program will follow the requirements of 1989 ASME Section XI Code in regards to additional examinations and complete those sample expansions within the outage they were identified.