November 19, 2004

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: ST. LUCIE PLANT, UNIT 2 - SECOND REQUEST FOR ADDITIONAL INFORMATION REGARDING PROPOSED LICENSE AMENDMENT, WCAP-9272 RELOAD METHODOLOGY AND IMPLEMENTING 30 PERCENT STEAM GENERATOR TUBE PLUGGING LIMIT (TAC NO. MC1566).

Dear Mr. Stall:

By letter dated December 2, 2003, Florida Power and Light Company (FPL) submitted an amendment request to revise the reload methodology and to allow operation of St. Lucie Unit 2 with a reduced reactor coolant system flow, corresponding to a steam generator tube plugging level of 30 percent per steam generator. The U. S. Nuclear Regulatory Commission (NRC) staff issued a Request for Additional Information (RAI) on June 21, 2004. This was discussed with members of the FPL staff in meetings on July 19 and 20, 2004, and FPL responded to the RAI by letter dated September 14, 2004.

The NRC staff has reviewed your submittals and has held additional discussions with your staff and finds that the additional information contained in the enclosed second RAI is needed before we can complete the review.

This request was discussed with Mr. George Madden of your staff on November 15, 2004, and it was agreed that a response would be provided by December 10, 2004. If the response cannot be provided by the agreed upon date, FPL should notify the NRC staff in writing. Upon written notification, a new date may be established with agreement from the NRC staff. If a satisfactory response is not provided in a timely manner, the NRC staff may proceed on your request consistent with Title 10, *Code of Federal Regulations*, Section 2.108, Denial of application for failure to supply information.

If you have any questions, please feel free to contact me at 301-415-3974.

Sincerely,

/RA/

Brendan T. Moroney, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No.: 50-389

Enclosure: As stated

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE NUCLEAR POWER PLANT, UNIT 2

DOCKET NUMBER 50-389

The U. S. Nuclear Regulatory Commission (NRC) staff issued a Request for Additional Information (RAI) on June 21, 2004. This was discussed with members of the Florida Power and Light Company (FPL) staff in meetings on July 19 and 20, 2004, and FPL responded to the RAI by letter dated September 14, 2004. Based on the review of your responses to the RAI and subsequent discussions with members of your staff, the NRC staff finds that the following information is needed before we can complete the review of your proposed license amendment.

1. The following issues should be addressed to support the acceptability of the proposed delay time for a loss-of-offsite power (LOOP) following turbine trip assumed in the Main Steam Line Break (MSLB) analysis.

    a. Provide an evaluation of the St. Lucie plant-specific design features that justify the use of the chosen time delay for the consequential LOOP. The following possibilities should be addressed for the St. Lucie site-specific electrical design: degraded switchyard voltage, spurious switchyard breaker-failure-protection-circuit actuation, automatic bus transfer failure, and startup transformer failure. One approach would be to address the events identified in Table G.5 of the "Technical Work to Support Possible Rulemaking for a Risk-Informed Alternative to 10 CFR [*Code of Federal Regulations, Section*] 50.46/GDC [General Design Criterion] 35" (ML022120661), which indicates that these are the likely causes of a consequential LOOP.

    b. Since the non-safety 6.9kV system does not have degraded voltage protection relays, please verify that if a degraded voltage condition were to occur as a result of the loss of the St. Lucie generator following a reactor coolant pump (RCP) shaft seizure event or a MSLB event, the 6.9kV loads, including the RCPs, would remain energized and would not trip due to some other protective system action such as overcurrent relaying or motor overload protection. The voltage used for this determination should be the lowest voltage that the grid surrounding St. Lucie 2 can support without becoming unstable or undergoing a voltage collapse. At this voltage also provide the speed and flow reduction that would occur on the 6.9kV motors, including the RCPs.

    c. For the MSLB event that involves the actuation of Emergency Core Cooling Systems (ECCS) it is also important to know the LOOP delay times (if any) that would occur on the 4.16kV safety-related system. Please provide an analysis on the 4.16kV safety-related system similar to that done for the non-safety 6.9kV system. The analysis should evaluate the consequential LOOP possibilities identified in question 1.a above and should provide the time delays (if any) associated with each.

d. Because the MSLB event involves the actuation of ECCS, the consequences of the delayed LOOP on the performance of the electrical ECCS systems should be evaluated. The consequences of double sequencing and its associated vulnerabilities that would occur as the result of the delayed LOOP should be a part of this evaluation. These vulnerabilities include, but are not necessarily limited to: the consequences of starting large continuous-duty motors twice in quick succession with the first start under degraded voltage conditions and the second start with pump discharge valves open; the adequacy of the existing control logic to start loads on offsite power, shed those loads following the LOOP, and subsequently re-sequence those loads on the Emergency Diesel Generators (EDGs) with necessary delay to allow motor residual voltage to decay; interaction between the double sequencing and circuit breaker antipump logic that could lock out the breakers; the capability of the safety batteries to operate the necessary systems during an initial offsite power degraded voltage ECCS start, and subsequently restart the ECCS on EDGs; and the potential to trip motor overload protection or blow fuses as a result of a degraded voltage double sequencing scenario.

In discussions with with the NRC staff, FPL stated that a review of all Mode 1 MSLB analysis cases analyzed found that, in all cases where reactor trip/turbine trip ocurred, that the LOOP with the assumed 3-second delay occurred prior to the time where the Safety Injection (SI) signal would occur. Therefore, there is no direct double sequencing concern for these MSLB event analysis cases.

1) Rather than assuming a 3-second LOOP time delay, please provide a LOOP/Safety-Injection-Signal analysis using the LOOP time delays determined from your analysis that will be provided in response to question 1.c above. With regard to the degraded-voltage LOOP possibility, we note that for safety-related systems, one of the most likely times for separation of safety equipment from offsite power due to degraded voltage relay operation is during or immediately following ECCS energization on offsite power.

2) If your analysis still finds the SI signal will follow the LOOP, the consequences of the LOOP/delayed ECCS actuation on the performance of the electrical ECCS should be evaluated. The potential vulnerabilities that should be evaluated include, but are not necessarily limited to: the potential for overloading the EDGs as a result of simultaneously block loading or load sequencing LOOP loads and ECCS loads onto the EDGs, the potential for overloading the EDGs as a result of block loading or load sequencing ECCS loads onto operating EDGs that are powering LOOP loads, and the adequacy of existing control logic to power LOOP loads from the EDGs following the LOOP signal and then properly add ECCS loads to the already operating EDGs.

3) If your analysis finds that the LOOP will follow ECCS actuation, the consequences of the delayed LOOP on the performance of the electrical ECCS systems should be evaluated in accordance with the previously stated concerns regarding double sequencing.

e. During an MSLB event, the released steam causes a decrease in the reactor coolant system (RCS) temperature. In the presence of a negative moderator temperature coefficient, the decreased RCS temperature results in a positive reactivity addition. After the reactor trip, if the resulting positive reactivity is greater than the negative reactivity from the inserted control rods and the borated water from the SI system, the core will return to criticality for an MSLB post-trip core. Since the actual time of loss of grid or main generator will vary, please demonstrate that a LOOP at any time in excess of 3-seconds will not lead to insufficient borated water from the SI system that was credited in the proposed MSLB analysis. This should account for the possibility that SI pumps may have started on normal ac sources and then lost power, as the grid or main generator disconnected, until the EDGs start and load (the double sequencing phenomenon). The double sequencing of the SI pumps will delay the time of injection of SI flow into the core and can cause a reduction in the borated water injected from the SI system.

f. Similarly, the LOOP may occur near the maximum return to power (e.g., Core Average Heat Flux = 18.25% at 305.5 seconds for the hot zero power (HZP) case presented in submittal). An RCP coastdown initiated near the time of peak heat flux would further challenge the approach to departure from nucleate boiling (DNB) Specified Acceptable Fuel Design Limits (SAFDL). FPL is requested to expand its break size sensitivity study for an MSLB initiating from zero power without a LOOP to MSLB cases with and without a LOOP for power levels initiating from both full power and HZP levels, and provide the results of the limiting cases (in terms of break sizes and the time of LOOP in excess of 3-seconds) with consideration of these two issues. The results should demonstrate that the applicable acceptance criteria in the Standard Review Plan, Section 15.1.5 are met for the MSLB analysis.

g. How does the 3-second delay in LOOP affect the containment MSLB analysis? Provide an analysis that shows that the design pressure, design temperature and environmental qualification envelope are not exceeded and that the response to Generic Letter 96-06 remains valid. Also, address any other effects that changes in the containment analysis may have on other licensing basis considerations.

2. In discussions with the NRC staff, FPL stated that the analysis of the potential for LOOP scenarios on the non-safety 6.9kv RCP buses indicated that the immediate loss of one 6.9 kV bus and the associated two RCPs due to plant-centered failures following a reactor/turbine/generator trip is possible as a result of a plant-centered component failure. The staff notes that the licensing report used to support the 30 percent steam generator tube plugging application credited the LOOP delay time of 3-seconds in the MSLB, Feedwater Line Break (FWLB) and locked rotor analyses.

   Address the effect of the immediate loss of two RCPs due to a plant-centered component failure on the results of the analyses for MSLB, FWLB and locked rotor events in terms of fuel failures from experiencing DNB, and confirm that the cases identified in the analyses provided in the licensing report are the limiting cases.

3. The following questions are related to the response to questions 20.a and 20.b of the initial RAI:

   a. Describe the model used for the analysis of the boron dilution event. Also, provide the definition for the RCS volumes of 3412 ft^3, 3712 ft^3, and 7368 ft^3 assumed in the analysis.

   b. Clarify the following statement in response to question 20.a:

      > The number of operating charging pumps, operable shutdown cooling system (SCS) and RCPs are all modeled consistent with the Technical Specification.

      Provide a table showing the SCS and RCP assumed to be in operation for each Mode in the analysis, and confirm that the assumptions are consistent with the Technical Specification (TS) requirements.

4. The following questions relate to Pre-Trip MSLB Issues:

   a. At the July 2004 meeting at the NRC Headquarters (HQ), the staff stated that an MSLB with coincident LOOP (LOAC [loss-of-ac-power], at T=0 sec) would need to be evaluated. In the past, this case was always bounded by the LOAC occurring at reactor trip breaker opening (RTBO). Since this submittal credits a 3+ second delay for LOAC, the coincident LOAC scenario now needs an evaluation. Provide justification that the Pre-Trip MSLB with coincident LOAC does not violate SAFDLs and is bounded by the case presented in the submittal.

   b. At the July 2004 meeting at NRC HQ, the staff stated that an MSLB with Failure of a Fast Bus Transfer (FFBT) would need to be evaluated. This case results in a two-RCP coastdown at reactor/turbine trip. St. Lucie Unit 2 Updated Final Safety Analysis Report, Section 15.1.4.3 documents the MSLB with FFBT event and lists 3.7 percent fuel failure. Note that recent St. Lucie Unit 2 core reloads may not have analyzed this case since it was bounded by the MSLB with LOAC at RTBO scenario (which lists 33 percent fuel failure). With the 3+ second delay in LOAC, the new analysis exhibits no fuel failure. Therefore, the 15.1.4.3 MSLB with FFBT case may now be more limiting. The submittal does not address this case. Therefore, the staff requests that FPL submit the limiting MSLB with FFBT case clearly defining inputs and assumptions and demonstrate that this scenario does not violate SAFDL or provide an associated dose calculation.

   c. Both the Pre-Trip MSLB and Feedwater Line Break (FWLB) analyses credit a 0.25-second delay between the RTBO signal and the turbine trip. Any safety grade actuation which provides a credit to mitigating the consequences of a transient must have a firm bases backed by surveillance requirements. The staff is unaware of any surveillance requirements on the link between reactor trip and turbine trip.

      1) The response to the previous RAI question 13.a states: “Assuming 3.0 seconds for the loss of offsite power delay and a 0.25 second delay

for the turbine is bounded by the 3.3 seconds justified for the loss of offsite power.” Please clarify your position with regard to both the loss of offsite power delay being credited in your submittal (3.0 versus 3.3 seconds) and a justifiable turbine trip delay.

2) For any situation where there is no firm basis, the analyst should select a conservative value, which is both reasonable and provides little mitigation to the transient. In past submittals, some indicated that a delay of 0 second was used when it made the event worse and others indicated that a delay of 3.0 seconds was used when it made the event worse. For both MSLB and FWLB events, discuss the impact of either a 0-second delay or a 3.0-second delay on the nuclear steam supply system response.

d. With regard to Variable High Power - Excore Power Signal, the response to the previous RAI question 13.d.3 states that "the trip signal is only assumed to be operable for 60 seconds after the break initiation . . . ." For an inside containment break, containment would quickly experience an increase in temperature, humidity, pressure, and radiation levels (small increase due to secondary side only). This submittal credits a limited availability of this instrumentation. Please provide further information on the environmental qualification (EQ) status for harsh environment of instrumentation and cables supporting this trip function.

e. For the inside containment MSLB and FWLB events, the new methodology credits a Low RCS Flow reactor trip function. Even though containment would quickly experience an increase in temperature, humidity, pressure, and radiation levels (small increase due to secondary side only), the submittal states, “. . . there would be insufficient time for the adverse environment to affect the setpoint modeled for the low flow trip." Please provide further information on the EQ status for harsh environment of instrumentation and cables supporting this trip function.

f. The Combustion Engineering (CE) methodology recognizes that the decreasing temperature will produce a change in local power peaking. In past analyses, the local peaking factors have increased during this cooldown event. The response to previous RAI question 13.e lacks any quantification of this effect. Please discuss the change in methodology that allows the exclusion of temperature effects and quantify any change in power distributions.

5. The following questions relate to the Post-Trip return to power (R-t-P) MSLB:

a. The submittal follows established Westinghouse methodology in evaluating only the HZP case without LOAC. This methodology identifies several conservative aspects which it claims make HZP inherently more severe than the hot full power (HFP) case. However, for the CE fleet, the HFP case may approach and sometimes become more limiting than the HZP case. In many cases, fuel management guidelines require preserving control element assembly scram worth (N-1) greater than TS shutdown margin requirements. Further, the R-t-P

case is time/path dependent, being influenced by the rates of cooldown, depletion of secondary inventory, and SI boron entry to the core. After reviewing the qualitative responses to previous RAIs, the staff is still not convinced that the HFP case will not challenge SAFDLs for all future cycles. Therefore, the staff requests that FPL submit an HFP MSLB case clearly defining inputs and assumptions along with minimum HFP–1 scram worth and maximum power peaking factors that will be validated against future core reloads.

b. FPL requests a change from the current licensing basis for the timing of LOAC. The response to previous RAI question 8.e states that the accident analyses consider the possibility of the LOAC occurring "simultaneously with the pipe break," "during the accident," and "offsite power may not be lost." In the submittal and in response to RAIs, FPL provides justification that the Post-Trip MSLB without LOAC case is more limiting than Post-Trip MSLB with a coincident LOAC. However, the submittal provides no justification for the case where LOAC occurs during the accident. The staff requests that FPL submit justification that a LOAC occurring beyond 3.3 seconds post-trip (since this time interval has been addressed separately) would not further challenge the SAFDLs. The sensitivity study should include a LOAC occurring near the peak R-t-P heat flux.

6. The following questions relate to the FWLB Event:

a. At the July 2004 meeting at NRC HQ, the staff stated that a FWLB with FFBT would need to be evaluated. This case results in a two-RCP coastdown at reactor/turbine trip. The submittal does not address this case. Therefore, the staff requests that FPL submit the limiting FWLB with FFBT case clearly defining inputs and assumptions.

b. During an inside containment FWLB event, a Safety Injection Actuation Signal (SIAS) may be generated on high containment pressure. Based on several recent submittals, the staff is now aware of a potential limiting case whereby this SIAS further challenges peak pressure and the requirements of the Three-Mile Island (TMI) Action Plan, item II.D (i.e., preclude liquid discharge from pressurizer safety valves). Specifically, all charging pumps start on an SIAS and this liquid mass addition into the RCS increases pressurizer liquid level. Coupled with a decrease in heat removal (due to the FWLB event), the pressurizer may go solid and/or liquid may be discharged from the pressurizer safety valves. The staff does not believe that this scenario has been properly addressed for St. Lucie Unit 2. Further, the staff believes that compliance to both peak pressure criteria and TMI requirements needs to address the engineered safety features actuation system actuations, even when those actuations are not beneficial (as is the case when SIAS starts charging pumps). As such, the staff requests that the limiting FWLB scenario for peak pressurizer liquid level and TMI compliance be identified and analyzed. Clearly define initial conditions, assumptions, operator actions, and modeling techniques employed in this case. Consider the most limiting single failure (e.g. failure of steam driven or motor driven Auxiliary Feedwater pumps), a LOOP, and the potential for starting charging pumps on an SIAS on high containment pressure.