November 23, 2005

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 TO DEFINE THE DEPTH OF REQUIRED TUBE INSPECTIONS AND CLARIFY THE PLUGGING CRITERIA WITHIN THE TUBESHEET REGION OF THE ORIGINAL STEAM GENERATORS (TAC NO. MC5084)

Dear Mr. Stall:

By letter dated November 8, 2004, Florida Power and Light Company (FPL) submitted an amendment request to revise the Technical Specification (TS) Section 4.4.5.4 to modify the definitions of steam generator tube "Plugging Limit" and "Tube Inspection," as contained in TS Items 4.4.5.4.a.6 and 4.4.5.4.a.8, respectively, for St. Lucie Unit 2. The purpose of these modifications is to define the depth of the required tube inspections and to clarify the plugging criteria within the tubesheet region. The U.S. Nuclear Regulatory Commission (NRC) staff issued a Request for Additional Information (RAI) on December 16, 2004, and FPL provided a response by letter dated March 31, 2005.

The staff has reviewed your response to the RAI and has determined that additional information is needed before we can complete the review. This request was discussed with your staff on October 4 and November 15, 2005, and Mr. Michael O'Keefe indicated that a response would be provided by January 20, 2006. If you have any questions, please feel free to contact me at 301-415-3974.

Sincerely,

/RA/
Brendan T. Moroney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No.: 50-389

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE NUCLEAR PLANT, UNIT NO. 2

DOCKET NUMBER 50-389

1. In the March 31, 2005, response to request for additional information (RAI) number 12, all available data were used to support the analytical adjustment to account for the axial load resistance provided by the differential thermal expansion effects. However, it is not clear whether all of the available data was used to support the analytical adjustment to account for the axial load resistance provided by internal pressure. For example, specimens 8 and 12 from the Task 1154 program were run at room temperature with internal pressure; however, an analysis of this data (similar to what was done for the elevated temperature data point) was not provided. Please evaluate all data in which internal pressure (above ambient pressure) was applied to support the basis for the analytical adjustments to account for the internal pressure. With respect to the analysis of the pressure effects provided in your response, please provide additional details on how the axial force resistance due to the internal pressure of 1435 psi was calculated and discuss how the effect of the residual contact pressure was taken into account in your analysis. (The actual pullout force was nearly the same as the pullout resistance expected analytically from the internal pressure effects. As a result, if the residual contact pressure was not included in this assessment, it would appear that the analytical adjustments for internal pressure are too high.)

2. The U.S. Nuclear Regulatory Commission (NRC) staff is currently reviewing an amendment request to permit the installation of sleeves at St. Lucie Unit 2. In some cases, the sleeve joint may be established greater than 10.1 inches below the top of the tubesheet or the bottom of the expansion transition, whichever is lower. As such, Technical Specification (TS) 4.4.5.4.a.8 would no longer require the sleeve or tube to be inspected in this region. However, this is in potential conflict with a proposed requirement to inspect both the tube and sleeve over their full length. That is, it could lead to the incorrect interpretation that only the portion of the sleeve above 10.1 inches from the top of the tubesheet was required to be inspected. Similarly, there is a potential conflict of the tube plugging (or repair) limit in TS 4.4.5.4.a.6 since the plugging limit is not applicable below 10.1 inches from the top of the tubesheet despite the fact that the expectation is that any degradation in the pressure boundary portion of the sleeve/tube assembly below 10.1 inches from the top of the tubesheet is plugged on detection. Please correct these apparent conflicts/discrepancies in your proposed TSs.

3. In the March 31, 2005, responses to RAI 16 and 22, you discussed various data that was not included in Appendix B; however, some data in Appendix B was not included in Table 4-1 (which is used in determining the leak rate as a function of joint length). Please discuss the basis for not including all of the Appendix B data in Table 4-1. For example, was data not included in Appendix B when it was well outside the targeted temperatures or pressures? Furthermore, was data from Appendix B not included in Table 4-1 when steady state was never reached although the temperatures and pressures were within the desired range?

ENCLOSURE

4. In Attachment 1 to your November 8, 2004, submittal, it was indicated that (as part of another amendment request) the Limiting Condition for Operation (LCO) leakage rate was being reduced from 0.5 to 0.15 gallons per minute (gpm) per steam generator. You further stated that this modification will reduce the margin between the assumed primary to secondary leakage rate of Westinghouse Topical Report WCAP-16208-P (0.1 gpm) and the reduced LCO leakage rate utilized in the Updated Final Safety Analysis Report (UFSAR) accident analyses (0.15 gpm). The LCO leakage rate in your TSs limits the amount of primary-to-secondary leakage during normal operation (i.e., normal operating leak rate limit). Based on your statements, the NRC staff is inferring that your UFSAR accident analyses (e.g., steam line break) assumes that the amount of primary-to-secondary leakage during the accident is identical to your LCO leakage limit (i.e., 0.15 gpm). If this is correct, please address the following:

a. During a steam line break the differential pressure across the tubes is greater than the differential pressure during normal operation. As a result, the primary-to-secondary leakage may be greater during a steam line break than during normal operation. Since you could be operating with leakage as high as your normal operating leakage limit (0.15 gpm), the amount of leakage during a steam line break (or other postulated accidents) could be greater than that assumed in your accident analyses. If so, please discuss what controls are in place to ensure that you do not exceed your accident induced leakage limit simply as a result of normal operating leakage. In addition, discuss your plans for modifying your TS normal operating leakage limit to be consistent with your accident induced leakage limit assumed in your UFSAR accident analyses. Alternatively, discuss your plans for modifying your accident analyses to account for this phenomenon.

b. As part of the C* amendment, you will be assuming that there is 0.1 gpm accident induced primary-to-secondary leakage as a result of flaws within the tubesheet region. In addition, you may have accident induced leakage from other sources such as sleeves or other tube degradation. This latter amount of leakage will need to be limited to 0.05 gpm to ensure you do not exceed your accident induced leakage limits in your UFSAR. Since the source of any normal operating leakage is not known (i.e., it could be from sources other than the tubesheet or sleeves or other defects assumed to leak in your operational assessment) and it could be as high as 0.15 gpm (or even higher during some postulated accidents for the reason discussed above), it is not clear that you will be able to stay within your accident induced leakage limits unless you change your TS normal operating leakage limit or your UFSAR accident analysis leakage limit. Please discuss whether you will be able to stay within your accident induced leakage limit given your normal operating leakage limit and your proposed C* inspection requirements.

5. Please confirm that the hot-leg temperature at St. Lucie Unit 2 is greater than that assumed in the tubesheet deflection analysis (600 degrees Fahrenheit) and in determining the increase in contact pressure as a result of differential thermal expansion between the tube and the tubesheet.

6. Please clarify whether the load at first slip was reported and plotted in Figures 5-1 through 5-3 or whether the maximum load was plotted. If the load at first slip was not used in all cases,

please discuss the effect on the required inspection distance if the load at first slip was used. In addition, if the load at first slip was not used in your March 31, 2005, response to RAI 10, please confirm that the 10.1-inch proposed inspection distance is still bounded when the most limiting specimen (using load at first slip) is evaluated.

Mr. J. A. Stall
Florida Power and Light Company

ST. LUCIE PLANT

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