August 6, 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 UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK INFORMED INSERVICE INSPECTION RELIEF REQUEST THIRD 10-YEAR INSERVICE INSPECTION PROGRAM RELIEF REQUEST NO. 2 (TAC NO. MC0938)

Dear Mr. Stall:

By letter dated August 6, 2003, Florida Power and Light Company requested relief from the Inservice Inspection (ISI) requirements specified in the American Society of Mechanical Engineers (ASME) Code, for the third 10-year ISI interval at St. Lucie Unit 2. The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed your submittal for Relief Request No. 2 and 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 June 28, 2004, and Mr. George Madden indicated that a response would be provided by September 15, 2004.

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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Juno Beach, Florida 33408-0420

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REQUEST FOR ADDITIONAL INFORMATION

RISK-INFORMED INSERVICE INSPECTION RELIEF REQUEST

THIRD 10-YEAR INSERVICE INSPECTION PROGRAM RELIEF REQUEST NO. 2

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE NUCLEAR POWER PLANT, UNIT 2

DOCKET NO. 50-389

1. In Section 1.2 (page 3) core damage frequency (CDF) and large early release frequency (LERF) values for St. Lucie Unit 2 are summarized in the following statement: "The core damage frequency (CDF) based on 5.02-11 truncation and the large early release frequency (LERF) based on 0.01 conditional containment failure probability are 1.58E-05/Yr and 1.58E-07/Yr respectively." It is noted that the current submittal is based on a Probabilistic Safety Assessment (PSA) model updated in September 2001 (as compared to the 1999 PSA model update used for the original St. Lucie Unit 2 Risk-informed Inservice Inspection (RI-ISI) submittal of July 23, 2002). In the July 23, 2002 submittal, CDF and LERF information was provided as follows: CDF - 1.25E-05/yr and LERF - 6.00E-06/yr. This suggested a conditional containment failure probability (CLERP) of about 0.48, using the 1999 PSA models. Citing another datapoint, the St. Lucie Unit 1 RI-ISI submittal of July 30, 2003, also using a CDF model from 1999, but using a LERF model updated to 2001, provided the following data: CDF - 1.45E-05/yr and LERF - 3.43E-06/yr, suggesting a CLERP of about 0.24.

   The current submittal is different in that it appears that LERF is stated as a function of CLERP, rather than CLERP having been calculated from PSA model generated results of CDF and LERF. Please explain how you derived the CLERP value of 0.01, particularly in light of the suggestion that it is now much more optimistic than in the previous calculations.

   Also, please confirm that both the CDF and LERF PSA models were updated in September 2001. This is not clear from the text.

2. The discussion in support of showing that Identified IPE Weakness #2 (page 4) has no significant impact on this RI-ISI application, begins similarly to Florida Power and Light Company's (FPL's) response 11b in D.E. Jernigan to USNRC letter "Relief Request 29 Request for Additional Information Response", dated January 16, 2003. However, the current submittal omits a number of the supporting statements that were included in that RAI response. Please restore the supporting statements, or otherwise explain why the use of screening values for pre-initiator human actions in STL2's current PSA continue to have an insignificant impact on your RI-ISI application.

3. It was stated on page 7 that a Combustion Engineering Owner's Group (CEOG) peer review of a draft 2002 PSA update model was conducted in May 2002. Yet an older (September 2001) model was used in support of this RI-ISI program submittal. Please determine if any differences between the September 2001 model and the draft 2002

models might impact on the RI-ISI risk ranking and risk evaluation result, and provide the results of this determination.

4. WCAP 14572, Rev 1-NP-A, "Westinghouse Owners Group Application of Risk-informed Methods to Piping Inservice Inspection Topical Report," (p. 125) calls for an uncertainty analysis to ensure that no low safety significant segments could move into high safety significance when reasonable variations in the pipe failure and conditional CDF/LERF probabilities are considered. Yet, Section 3.5 (page 9) of your submittal indicates that based on insights gained from previous uncertainty analysis and based on sensitivity studies performed to address CEOG Peer Review Team comments, there was no 5$^{th}$ and 95$^{th}$ percentile uncertainty analysis performed.

5. Section 3.8 (page 11) indicates that there are 202 piping segments in the scope. Yet it breaks down the allocation into the different regions as follows: Region 1B - 9 segments, Region 2 - 2 segments, and Region 4 - 194 segments. This adds up to 205 segments total, different from the 202 segments noted. However, Tables 3.1-1 and 3.7-1 support the notion of a total of 202 segments in the scope.

   a. Please confirm that there are only 191 segments in Region 4, rather than the 194 segments noted in Section 3.8.

   b. Please explain why 3 segments were removed from the scope between the previous St. Lucie Unit 2 RI-ISI submittals and this one.

   c. Please confirm that the segments identified as High Safety Significant (HSS) in the previous St. Lucie Unit 2 RI-ISI submittal are the same as those identified as HSS in the current submittal, or otherwise explain what has changed and why.

   d. In connection with (c) above, please explain why, despite the fewer number of reactor coolant (RC) system segments, there are, per Table 3.7-1, now 3 segments with risk reduction worth (RRW) between 1.005 and 1.001 where there used to be only 2 such segments. Also, how was it decided that this additional "gray area" segment was not to be placed in the HSS category, particularly if the Expert Panel was not convened, as noted in Section 3.6? WCAP-14572, Rev. 1-NP-A, denotes that the Expert Panel is to make the categorization decisions.

6. Table 3.10-1 (page 17) shows that the RI-ISI alternative inspection program will be slightly beneficial overall from a risk perspective, relative to the traditional American Society of Mechanical Engineers (ASME) Code, Section XI program. Within this overall perspective, however, the table shows a noticeable fractional increase in risk from the charging and from the safety injection systems, with a slight fractional decrease in risk from the reactor coolant system. Contained in D.E. Jernigan to USNRC letter "Relief Request 29 Supplemental Request for Additional Information", dated March 26, 2003, is Attachment 2, which is identified in this letter as Revision 3 to Risk-Informed Inservice Inspection Plan for St. Lucie Unit 2, the most recent revision on the docket that pertains to the second inspection interval at St. Lucie Unit 2. Table 3.10-1 therein indicates absolutely no change in risk in any system between the traditional Section XI and the Risk-Informed alternative programs. Please explain these apparently contrasting results, particularly in light of the changes in the population of HSS piping segments

noted in the above question. Also, if relevant, please indicate how any changes in selected weld locations between the second and third interval risk-informed programs are influencing these “new” differences in risk.

7. In D.E. Jernigan to USNRC letter “Relief Request 29 Supplemental Request for Additional Information”, dated March 26,2003, FPL Response 9 and FPL Clarification to Response 9 provided FPL’s commitment to RI-ISI program situations requiring NRC notification and approval. Please restate FPL’s commitment to these situations requiring NRC notification and approval as part of this submittal.

8. The industry experience based on cracking of dissimilar metal welds, such as safe-end welds at V. C. Summer, Three Mile Island, and Ringhals 3 and 4, attributes the degradation mechanism to be primary water stress corrosion cracking involving Alloys 82 and 182. This degradation mechanism has not been addressed in the Topical Report WCAP-14572, Rev 1-NP-A. Please indicate if this recent industry experience was taken into account when selecting dissimilar metal welds in B-F and B-J categories for volumetric examination.

9. In Table 5-1, “Structural Element Selection,” none of the Category B-F welds have been selected for volumetric examination. Please discuss the basis for how these welds were excluded from volumetric examination, including selection criteria and a description of each weld (size, materials (including weld metal), location and operating temperature). Also, provide the same information for B-J category dissimilar metal welds that will not receive a volumetric examination per the St. Lucie Unit 2 RI-ISI program.