December 17, 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: ST. LUCIE PLANT, UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK INFORMED INSERVICE INSPECTION (TAC NO. MB5698)

Dear Mr. Stall:

By letter dated July 23, 2002, Florida Power and Light Company requested relief from the Inservice Inspection (ISI) requirements specified in the American Society of Mechanical Engineers Code. The U.S. Nuclear Regulatory Commission staff has reviewed your submittal 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 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 feel free to contact Eva Brown at (301) 415-2315.

Sincerely,

***/RA by EBrown for/***
Brendan T. Moroney, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-389

Enclosure: As stated

cc w/encl: See next page

December 17, 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: ST. LUCIE PLANT, UNIT 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING RISK INFORMED INSERVICE INSPECTION (TAC NO. MB5698)

Dear Mr. Stall:

By letter dated July 23, 2002, Florida Power and Light Company requested relief from the Inservice Inspection (ISI) requirements specified in the American Society of Mechanical Engineers Code. The U.S. Nuclear Regulatory Commission staff has reviewed your submittal 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 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 feel free to contact Eva Brown at (301) 415-2315.

Sincerely,

***/RA by EBrown for/***
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
<u>DISTRIBUTION</u>:

| | | | | | |
|---|---|---|---|---|---|
| PUBLIC | EBrown | BMoroney | AHowe | PDII-2 R/F | LWert, RII |
| OGC | SDinsmore | PPatniak | MRubin | TChan | BClayton (Hard copy) |
| ACRS | SMalik | AKeim | | | |

**ADAMS ACCESSION NUMBER: ML023510620** * See previous concurrence

| OFFICE | PDII-2/PM | PDII-2/PM | PDII-2/LA | EMCB/SC | SPSB/SC | PDII-2/SC |
|---|---|---|---|---|---|---|
| NAME | BMoroney | EBrown | BClayton | TChan | MRubin | AHowe |
| DATE | 12/16/02 | 12/16/02 | 12/16/02 | 11/26/02* | 12/12/02* | 12/17/02 |

OFFICIAL RECORD COPY

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

**ST. LUCIE PLANT**

cc:
Senior Resident Inspector
St. Lucie Plant
U.S. Nuclear Regulatory Commission
P.O. Box 6090
Jensen Beach, Florida 34957

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

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

Mr. Douglas Anderson
County Administrator
St. Lucie County
2300 Virginia Avenue
Fort Pierce, Florida 34982

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

Mr. Donald E. Jernigan, Site Vice President
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. R. E. Rose
Plant General Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. Kelly Korth
Licensing Manager
St. Lucie Nuclear Plant
6351 South Ocean Drive
Jensen Beach, Florida 34957

Mr. William Jefferson
Vice President, Nuclear Operations Support
Florida Power & Light Company
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

Mr. J. Kammel
Radiological Emergency
Planning Administrator
Department of Public Safety
6000 SE. Tower Drive
Stuart, Florida 34997

REQUEST FOR ADDITIONAL INFORMATION (RAI)

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

FLORIDA POWER AND LIGHT (FPL)

ST. LUCIE UNIT 2

DOCKET NO. 50-389

1. Were all Class 1 piping segments and all B-J and B-F welds included in the evaluation? If not, please provide the following information.

   a) With reference to Table 3.1-1 “System Selection and Segment Definition for Class 1 Piping,” please provide your rationale for excluding shutdown cooling system from risk evaluation.

   b) Are all the piping and welds in the Chemical and Volume Control (CH), Reactor Coolant (RC), and Safety Injection (SI) system included in the scope of the RI-ISI submittal?

   c) What percentage of Class 1 welds are excluded from the scope of the RI-ISI submittal?

   d) What percentage of butt welds in the portions of the Class 1 piping excluded from the scope of the RI-ISI submittal will continue to be inspected with volumetric examination under the American Society of Mechanical Engineers (ASME) Section XI program?

2. In Table 3.4-1 “Failure Probability Estimates (without ISI),” please provide justification for not addressing stress corrosion cracking (SCC) as a potential failure mechanism for the charging, safety injection and shutdown cooling systems. How will the failure probability be affected when SCC is considered as a potential degradation mechanism in addition to fatigue and thermal transients?

3. The industry experience based on cracking of alloy 600 safe-end material (Inconel 82/182) at V.C. Summer attributes the degradation mechanism to be primary water stress corrosion cracking. This degradation mechanism has not been addressed in the Topical Report WCAP-14572, Rev 1-NP-A. As discussed on page 3-87 in the Electric Power Research Institute (EPRI) Topical TR-112657, Rev. B-A, the EPRI location selection process includes guidance that additional locations should also be chosen from segments that are susceptible to a degradation mechanism that is not otherwise inspected and the examination effort should be directed to detect flaws or relevant condition as a result of this degradation mechanism. In Table 5-1 “Structural Element Selection,” three welds are selected for volumetric examination in B-F examination category in the reactor coolant system. Are these welds made of Inconel 82/182?

Enclosure

4. Under “Deviations,” it is stated that the selection of elements is based on that described in EPRI Topical Report TR-112657 Rev. B-A. Are the examination volumes and methods for each of the welds, heat-affected zones and base metal, specific to each degradation mechanism in the affected segment as proposed in EPRI-TR?

5. On page 2 of the submittal dated July 23, 2002, FPL stated that the methodology used deviated from the methodology presented in Westinghouse Owners Group WCAP-14572, “Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report” (WCAP) for selecting elements from Region 1B and 2 of the Structural Element Selection Matrix shown in Figure 3.7-1 of the WCAP. FPL indicated the use of the process described in EPRI TR-112657 to select elements from these regions instead of the one described in the WCAP. On page 10 of the submittal (Section 3.8, Structural Element and NDE [nondestructive examination] Selection), FPL stated that for the 205 piping segments that were evaluated, Region 1B contains 9 segments, Region 2 contains 2 segments, no segments are contained in Region 3, and Region 4 contains 194 segments. Explain why there are no parts of any segments in Region 1A (the placement of segments in Region 1A does not rely on the Perdue method). What failure probability value was used to distinguish elements between Region 1 and Region 2?

6. Are there any piping segments that include piping of different diameters? If so, how were the failure frequencies estimated for these segments? How does the methodology for determining the failure frequency comport with the methodology described on page 71 of the WCAP?

7. Table 3.4-1 of the submittal provides the Failure Probability Range for a Small leak probability. Provide the disabling leak failure probabilities, and a breakdown in small break loss-of-coolant-accident (LOCA), medium break LOCA, and large break LOCA failure probabilities.

8. Section 6 of the submittal, states that St. Lucie, Unit 2, was designed to ASME Section III and, therefore, has an improved level of fatigue analysis. Most plants were designed to ASME Section III and do not have leak probabilities as low as those presented in Table 3.4-1. Explain what physical characteristics of the St. Lucie reactor cause the estimation of small leak probabilities that are significantly lower than those given in other RI-ISI submittals.

9. Under what conditions would the RI-ISI program be resubmitted to the Nuclear Regulatory Commission (NRC) prior to the end of any 10-year interval.

10. Identify the version of the Probabilistic Safety Assessment (PSA) model that was used for the RI-ISI application and when it was last updated. Include when, and which version, of your PSA has been peer reviewed by the Combustion Engineering Owner's Group. Please also provide the Category A and Category B Facts and Observations obtained from your peer review and how these may impact your RI-ISI evaluation.

11. The safety evaluation (SE) report on the St. Lucie Individual Plant Examination (IPE), dated July 21, 1997, concluded that the IPE met the intent of Generic Letter 88-20. The SE also stated that "the staff identified weaknesses in the front-end, HRA [human reliability analysis] and back-end portions of the IPE which, we believe, limit its future usefulness." The weaknesses stated in the SE are briefly outlined below. Explain how each of the weaknesses has been removed by modifications to the PRA or otherwise addressed during the RI-ISI evaluation.

    a) Some initiating event frequencies appeared low and some initiating event frequencies which relied on generic values should have received a plant-specific analysis.

    b) Some preinitiator human actions appeared in dominant accident sequences, with unexpected and uncommon results. It appears that a more detailed analysis of preinitiator human actions may appropriately reduce the human error probabilities (HEPs) for these events, thus reducing the likelihood that excessively conservative HEPs may distort the risk profile.

    c) It was not clear what basis was used to determine which postinitiator human actions were quantified with a time-independent technique and those postinitiator actions that were quantified with a time-dependent technique. Three postinitiator human actions (initiating once-through cooling, manually initiating recirculation actuation components following loss of the automatic signal, and securing the reactor coolant pumps after loss of seal cooling) are relatively short timeframe events. Failure to consider time in these events might lead to unrealistic values.

    d) The time-dependent human actions used likelihood indices at their default values. Therefore, the resulting human error probabilities may be generic rather than plant-specific.

    e) An additional sensitivity analysis should have been performed regarding the probability of in-vessel recovery since the licensee assumed a very high probability of in-vessel recovery due to ex-vessel cooling.

12. On Table 3.5-1 of the submittal, you present the piping risk contribution by system without ISI. Explain how the risk contribution can be so low in the SI system given that the system contains 59 segments. Also, please provide the conditional core damage probability for small, medium, and large break LOCAs.

13. Table 3.10-1 presents a comparison of core damage frequency/large early release frequency for current Section XI and RI-ISI programs. Page 214 of the WCAP presents criteria for the evaluation of these results.

    a) Criterion 1 states that the total change in piping risk should be risk neutral or a risk reduction. The risk from the SI and RC systems remain the same but the risk from the CH system, although small, increases. Was an analysis performed of the dominant system and piping segment contributors to the RI-ISI? If so, provide any added inspection locations.

b) Criterion 2 states that any system that contributes greater than 10 percent of the total risk for RI-ISI should be examined when the overall risk has not changed. The criteria further state that any such identified systems should be reevaluated in an attempt to identify additional examinations which would reduce the risk for these systems and, thus, the overall risk. It appears from Table 3.10-1 that RC system contributes greater than 10 percent of the total risk for RI-ISI and the overall risk when implementing RI-ISI has not changed. Explain why additional examinations were not added to reduce risk in the RC system and thus reduce overall risk.

14. Section 3.5 discusses an uncertainty analysis done in support of the submittal.

    a) Explain how this uncertainty analysis comports or differs from the uncertainty analysis required on page 125 of the WCAP.

    b) Explain how the 5 percent and 95 percent percentile bounds were developed for the piping failure frequency and provide some representative ranges.

15. 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.