July 22, 2003

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: SAINT LUCIE PLANT, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUESTS FOR INCREASING SPENT FUEL STORAGE CAPACITY (TAC NOS. MB6627 AND MB6628)

Dear Mr. Stall:

By letter dated October 23, 2002, Florida Power and Light Company submitted a request to revise the St. Lucie, Units 1 and 2, Technical Specifications to include the design of a new cask pit spent fuel storage rack for each unit in order to increase each unit's spent fuel storage capacity.

The U.S. Nuclear Regulatory Commission staff has reviewed your submittal and finds that a response to the enclosed request for additional information (RAI) is needed before we can complete the review. This request was discussed with your staff on July 16, 2003, and Mr. Ken Frehafer agreed that a response would be provided by August 31, 2003.

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 Nos. 50-335 and 50-389

Enclosure: RAI

cc w/encl: See next page

July 22, 2003

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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, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUESTS FOR INCREASING SPENT FUEL STORAGE CAPACITY (TAC NOS. MB6627 AND MB6628)

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REQUEST FOR ADDITIONAL INFORMATION

ADDITION OF SPENT FUEL POOL CASK AREA RACK AMENDMENT

FLORIDA POWER AND LIGHT

SAINT LUCIE PLANT, UNITS 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1) Describe how the capability to remove fuel from the spent fuel pool (SFP) will be assured with licensed fuel storage in the new cask area fuel storage rack.

2) The submittal states that the proposed change in the design basis will be consistent with the Standard Review Plan (SRP), which requires a bulk spent fuel pool design temperature of 140°F to provide margin with a design basis of partial core offloads. The submittal assumes 150°F. Provide justification for the deviation.

3) The submittal proposes to change the design basis to be a partial core offload. It also proposes to reduce the core offload time after shutdown from 168 hours to 120 hours. A review of SRP Section 9.1.3.III.1.h indicates that spent fuel pool heat loads associated with both partial and full-core offloads are calculated based on 150 hours decay. At the reduced offload time the current licensing basis can be maintained for the partial-core offload, but may not be maintained in the case of a planned full-core offload. While not routine, full-core offloads are periodically necessary and are performed during planned outages. The current licensing basis for Unit 2 as stated in Updated Final Safety Analysis Report (UFSAR) Section 9.1.3.1, requires "outage-specific calculations to demonstrate that the spent fuel pool bulk water temperature will not exceed the St. Lucie Design-Basis temperature of 150°F with one Spent Fuel Cooling System pump and one heat exchanger in operation" for refueling evolutions that propose to utilize a full-core offload (UFSAR Section 9.1.3.1). Please explain how the design basis related to bulk pool temperature will be maintained for refueling evolutions that propose to utilize a full-core offload for Unit 1.

4) Explain if using a 90-day operation time is more conservative than the 36-day operation time used in the SRP, Scenario 3. If it is not conservative, provide the maximum bulk SFP temperature for an emergency full-core offload having 36 days operation time since the previous refueling outage for each unit.

5) Provide the flow rates for the SFP make-up sources.

6) The submittal states in several areas that certain crane features are in accordance with NUREG-0612 , "Control of Heavy Loads at Nuclear Power Plants." It also states several changes will be made, including increased crane capacity and replacement or upgrade to single failure proof cranes. Identify any upgrades or replacements necessary for this license amendment.

Enclosure

7) The submittal identified crane features that follow the approach of NUREG-0612. Please explain any deviations from the NUREG-0612 guidance.

8) The submittal states that safe-load paths will be established for loads, specifically the spent fuel racks in and out of the cask pits. Explain the guidelines for establishing safe-load paths. In particular, address if the load could travel over spent fuel, any safety-related equipment, or any part of the spent fuel pool (e.g., weir wall).

9. The proposed amendment described the methodology used to calculate the maximum effective multiplication factor (k_{eff}). The staff has outlined two acceptable methodologies to perform spent fuel pool criticality analyses in a letter entitled "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," from L. Kopp to T. Collins dated August 19, 1998. The two methodologies are (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff}, or (2) a sensitivity study of the reactivity effects of the tolerance variations. The proposed amendment is unclear on which methodology was used. Identify which methodology was employed to calculate the maximum k_{eff}.

10. The submittal indicates that maximum effective multiplication factors were calculated by statistically combining all of the tolerances and uncertainties for each of the St. Lucie cask pits with the new racks present and loaded. However, the submittal does not contain the equations used to calculate these values. Please provide the equations used to perform the k_{eff} calculations and a detailed quantitative example demonstrating how each of the tolerances and uncertainties were accounted for in the calculation. The response should include a detailed description of the statistical methods employed and the values used in the calculation of any statistical uncertainties.

11. The submittal identifies the worst possible moderation condition as a spent fuel pool temperature of 50°F (10°C) due to a negative moderator temperature coefficient. The maximum density of water occurs at 39.2°F (4°C). Provide justification for selection of a higher temperature (i.e., lower density) as the worst possible moderation condition.

12. Section 4.2.4.3 of the supporting Holtec report provided a table identifying the principal core operating parameters used to analyze the burn-up calculations for the St. Lucie, Unit 2, spent fuel assemblies. The submittal did not identify how these values represented the most limiting conditions (resulting in the highest residual reactivity) of the spent fuel to be stored in the cask pit racks. Please provide a detailed justification for each of the values provided, demonstrating how they result in the most limiting reactivity conditions of the spent fuel.

13. The submittal described a limitation of the Monte Carlo N-Particle Transport Code (MCNP) calculations that prevented modeling some fission product nuclides in the criticality analyses, and described a process to calculate an equivalent amount of boron that provides nearly the same reactivity in MCNP as the CASMO4 results. The submittal stated that this process would compensate for the inability to model these nuclides. Please provide detailed technical information demonstrating that this alternate methodology is conservative or provides bounded results.

14. The submittal identified the most limiting postulated accident condition as the misplacement of a fresh fuel assembly into a location intended for storage of a spent fuel assembly. The description of the analysis only includes a discussion of the maximum effective multiplication factor obtained. Please provide a detailed description of the following: (1) the assumptions used in the analysis, (2) how each assumption represents the most bounding or limiting condition, (3) the biases and uncertainties included in the analysis, (4) how each bias or uncertainty was accounted for, (5) how the data was evaluated, and (6) how this analysis varies from the currently licensed worst-case misplacement accident in the spent fuel pool.

15. A 5 percent uncertainty in fuel density was assumed when performing the Unit 1 cask pit criticality analysis; however, only 1 percent uncertainty was assumed in the Unit 2 analysis. The reduced uncertainty will result in lower residual reactivities in the spent fuel. Please provide a detailed justification for the values assumed in each of the analyses and the basis for the differences.

16. The references ([1], [2],...) located within the licensee's submittal Sections 4.1 and 4.2 do not correlate to the references listed in section 4.3. Provide a revised list of references to correctly identify the appropriate documents.

17. The current St. Lucie, Unit 2, spent fuel racks have a nominal 8.96-inch center-to-center distance between the fuel assemblies; however, the proposed amendment calls for a nominal 8.80-inch center-to-center distance between fuel assemblies placed in the cask pit storage rack. Please describe the basis for the reduced spacing and discuss how the change was accounted for in the reactivity calculations.

18. Section 3.2 of the amendment request states, "...the Unit 1 rack cells employ Boral neutron absorber panels mounted on the outside faces of stainless steel boxes... (except cells on the rack periphery which contain no Boral panel on the outer face)...." The licensee's criticality analysis assumed an infinite array of storage cells. This array assumed the presence of two Boral panels between adjacent assemblies. The lack of a Boral panel on the outside periphery of the rack may result in greater neutron coupling between the cask pit racks and adjacent spent fuel pool racks. Please provide a detailed description, explaining why greater neutron coupling will not occur with racks adjacent to the cask pit racks, or submit a criticality analysis to evaluate this condition. Additionally, please perform the same evaluation and provide the same requested information for the Unit 2 spent fuel pool.

19. The criticality analysis assumed a minimum Boral density as an uncertainty of the analysis. Please provide a detailed description of the surveillance program that will be used to monitor the installed Boral panels to verify they will continue to meet this limit in the future. Additionally, demonstrate that the surveillance program schedule will be sufficient to identify the depletion of the Boral panels before the boron density decreases below the value assumed in the criticality analysis.

20. The application indicates that materials containing boron will be part of the design of the spent fuel storage racks that will be installed in the cask area. The application does not address the potential increase in tritium that might be produced by neutron capture of boron-10 and released in liquid effluent pathways from the plant. Identify how much

additional tritium is expected to be produced and released. Describe the significance of any estimated increase.

21. The additional stored spent fuel will increase the amount of heat being removed from the water in the spent fuel pool and cask area. Specify how much additional heat will be released to the cooling canal. Explain the significance of any estimated increase.

22. According to Section 9.6 of the application, all spent fuel and spent fuel storage racks will be removed from the cask pit before a cask is brought into the pit. State whether this restriction is formalized in an administrative control. If so, describe how will this restriction be formalized.

23. Section 3.7, "Radiological Considerations," indicates that the radiological consequences of a fuel-handling accidents are discussed in the Unit 1 and 2 UFSAR fuel-handling accident analyses. In Table 15.4.1-7 of the Unit 1 UFSAR, the control room thyroid dose resulting from the fuel-handling accident is different from the dose stated in the analyses supporting the requests for amendment dated October 30, 2000 (Amendment 172), and May 23, 2002 (Amendment 184). Please clarify.

cc:
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