August 16, 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 UNITS 1 AND 2 - CORRECTION/CLARIFICATION TO NRC SAFETY EVALUATION REGARDING CASK PIT RACK AMENDMENTS (TAC NOS. MB6627 AND MB6628)

Dear Mr. Stall:

By letter dated July 9, 2004, the U.S. Nuclear Regulatory Commission issued Amendment Nos. 192 and 135 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the Florida Power and Light Company (FPL) St. Lucie Plant, Units 1 and 2, respectively. These amendments revise Technical Specification Section 5.6, “Design Features - Fuel Storage,” for St. Lucie Units 1 and 2 to include the installation of a new cask pit spent fuel storage rack for each unit, and increase each unit’s spent fuel storage capacity by combining the cask pit rack and existing spent fuel pool storage rack capacities.

The FPL staff has informed the NRC of some inaccuracies in the safety evaluation (SE) supporting the amendments. We have resolved this by correcting and/or clarifying (as applicable) the SE. The corrected SE is included as an enclosure to this letter. Revisions are identified by a line in the margin. This letter with its enclosure should be attached to the subject SE to document the resolution of the inaccuracies. In this case, the inaccuracies do not result in a change to the staff’s conclusion in the subject SE.

The thoroughness of your staff in identifying these inaccuracies is appreciated, and is an important contribution in ensuring the accuracy of the SEs, which form the basis for approval of licensing amendments. If you or your staff have any questions concerning the resolution of this matter, please call 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

Enclosures: As stated

cc w/enclosures: See next page

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CORRECTED SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 192 AND 135

TO RENEWED FACILITY OPERATING LICENSES NOS. DPR-67 AND NPF-16

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNITS NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated October 23, 2002, as supplemented by letters dated August 28, 2003, December 11, 2003, February 3, 2004, and March 25, 2004, the Florida Power and Light Company (the licensee) requested an amendment to Facility Operating Licenses DPR-67 for St. Lucie Unit 1, and NPF-16 for St. Lucie Unit 2. The amendment proposes to revise Technical Specification (TS) section 5.6, "Design Features - Fuel Storage," for both St. Lucie units to include the design of a new cask pit spent fuel storage rack for each unit, and increase each unit's spent fuel storage capacity by combining the cask pit rack and existing spent fuel pool (SFP) storage rack capacities. The amendment also proposes to change the Unit 2 normal offload condition, from a full-core offload to a partial core offload, thus making the partial core offload as the normal offload condition for both Unit 1 and Unit 2. Additionally, the amendment supports reducing the core offload time after reactor shutdown from 168 hours to 120 hours.

The licensee's supplementary submittals dated August 28, 2003, December 11, 2003, February 3, 2004, and March 25, 2004, provided clarifying information that did not change the scope of the proposed amendment as described in the original notice of proposed action published in the *Federal Register* and did not change the initial proposed no significant hazards determination.

The licensee has designed the new storage racks to increase fuel storage capacity while maintaining criticality control. The Unit 1 pool is currently licensed to store a total of 1706 fuel assemblies. The new Unit 1 cask pit rack will increase spent fuel storage capacity by 143 fuel assemblies thus increasing the total spent fuel storage capacity for Unit 1 to 1849 fuel assemblies. The Unit 2 pool is currently licensed to store a total of 1360 fuel assemblies. The new Unit 2 cask pit rack will increase spent fuel storage capacity by 225 fuel assemblies thus increasing the total spent fuel storage capacity for Unit 2 to 1585 assemblies. For criticality control the new racks use a combination of geometric spacing and a fixed neutron absorber, Boral. Based on the current spent fuel storage capacities, Unit 1 will lose full-core offload capability in the year 2005, and Unit 2 will lose full-core offload capability in the year 2007. The additional storage capacity provided by the cask pit racks is expected to extend full-core offload capability until the year 2008 for Unit 1, and until the year 2012 for Unit 2.

2.0 REGULATORY EVALUATION

This amendment request has been evaluated from several aspects, each having unique regulatory requirements. The evaluation of these requirements is included in the appropriate section of the Technical Evaluation.

3.0 TECHNICAL EVALUATION

3.1 Cask Pit Storage Rack Design

3.1.1 Regulatory Evaluation

The storage expansion will add one 11 x 13 (143 total cells) Region 1 style storage rack to the Unit 1 Cask Pit and one 15 x 15 (225 total cells) Region 2 style storage rack to the Unit 2 Cask Pit. The new Cask Pit storage racks consist of fuel storage modules attached to a base plate and are freestanding and self-supporting.

The racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel (ASME) Code. The material procurement, analysis, fabrication, and installation of the rack modules conform to Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix B requirements. The license indicated that computer programs utilized to perform the analyses were benchmarked and verified, and that these programs were utilized by Holtec International (designer of the racks) in numerous license applications over the past decade.

For the seismic analysis, synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) were generated in accordance with the provisions of U. S. Nuclear Regulatory Commission (NRC) technical report NUREG-0800, Standard Review Plan, Section 3.7.1, “Seismic Design Parameters.” The structural damping values were based on the Updated Final Safety Analysis Report (UFSAR) for the respective units. A 3-dimensional dynamic rack model suitable for a time-history analysis, including fluid coupling interactions and mechanical coupling, nonlinear simulation was established. Analyses relating to various physical conditions, such as coefficient of friction and the number of cells containing fuel assemblies were performed. The applicable loads and their combinations in the seismic analysis of rack modules are in accordance with NRC Generic Letter (GL) 78-11, “Review and Acceptance of Spent Fuel Storage and Handling Applications” (April 14, 1978) and NRC Standard Review Plan, Section 3.8.4, “Other Seismic Category I Structures,” Rev. 2, 1989. Analysis results were compared with the allowable stresses of the ASME Code, Section III, Subsection NF.

The maximum computed value of displacement in the two horizontal directions is 0.396 inch. The minimum nominal gap between the rack and wall for both units is 3-15/16 inches. Therefore, the rack does not impact the adjacent walls at any time. The analysis results indicate that the racks will not tip over. Rack impact loads arise from rattling of fuel assemblies in the storage racks during earthquakes. For Unit 1 under the safe-shutdown earthquake (SSE) condition, the maximum calculated fuel/cell wall impact load is 419 lbs. while the allowable load is 3,423 lbs.; the maximum calculated rack/baseplate weld stress is 8,903 psi while the allowable stress is 35,748 psi; the maximum calculated baseplate/pedestal weld stress is 6,167 psi while the allowable stress is 35,748 psi; and the maximum calculated cell/cell weld stress is 1,946 psi while the allowable stress is 8,520 psi. For Unit 2 under the SSE condition, the maximum calculated fuel/cell wall impact load is 449 lbs. while the allowable load is 3,423 lbs.; the maximum calculated rack/baseplate weld stress is 11,048 psi while the allowable stress is 35,748 psi; the maximum calculated baseplate/pedestal weld stress is 13,780 psi while the allowable stress is 35,748 psi, and the maximum calculated cell/cell weld stress is 2,515 psi while the allowable stress is 8,520 psi.

The rack fatigue margins were calculated. The cumulative damage factor, which was calculated in accordance with ASME Code guidelines, for the combined effect of 1 SSE and 20 operating-basis earthquake events, is 0.196 for Unit 2 and less than 0.1 for Unit 1. The allowable cumulative damage factor is 1.0.

The floor of the cask pit in each unit is at a lower elevation than the floor of the SFP. In order to ensure that the top of any spent fuel rack placed in the cask pit is at the same elevation as racks in the pool, a cask pit rack platform will be installed in each cask pit to raise the top of the rack to the appropriate elevation. Maintaining the cask pit rack height the same as the existing racks facilitates fuel movement and precludes the need for changing height-related limits on the fuel handling cranes. The platforms were designed with the same code requirements as the supported racks. Analysis results demonstrate that the platform meets design code requirements.

Analyses were performed to evaluate the impact on fuel assembly racks under various fuel assembly drop scenarios. In a shallow drop event, a fuel assembly, along with the portion of the handling tool, which is severable in the case of a single element failure, were assumed to drop vertically and hit the top of the rack. For the shallow drop event, the dynamic analysis shows that the top of the impacted rack undergoes 12.5 inches localized plastic deformation. The allowable deformation limit of the rack is 35.75 inches. In a deep drop event, the fuel assembly, along with the portion of the handling tool, were assumed to fall through an empty storage cell and impact the fuel assembly support surface (i.e., rack baseplate). The deep drop event lowered the fuel assembly surface by 1.96 inches, which is less than the distance of 4.25 inches from the baseplate to the rack platform. The deep drop did not sever the baseplate/cell wall welds. The deep drop produces a maximum stress of 3,933 psi in the liner, which is less than the yield stress of the liner material. The deep drop also produces a maximum compressive stress of 1,260 psi in the concrete slab, which is less than the concrete strength of 3,000 psi. These analysis results demonstrate that the design of the rack meets the NRC position stated in GL 78-11.

The cask pit is adjacent to the SFP, and the two areas share common exterior walls of the Fuel Handling Building (FHB). The spent fuel cask handling crane outside the FHB will be upgraded to single-failure proof, which will result in new design tornado and seismic loads on some portions of the exterior walls. The reinforced concrete slab in the cask pit region and portions of the SFP exterior walls, which will be affected by the proposed fuel rack expansion and the upgrade of the crane, were evaluated for the new loads. Load combinations and structural capacity assessments for the analysis followed the requirements of the Unit 1 UFSAR and its design code, the American Concrete Institute (ACI) Code 318-63, and of the Unit 2 UFSAR and its design code, ACI 318-71. The analysis results demonstrate that adequate safety margins exist for all loading combinations against the appropriate structural design strengths of the affected FHB walls, the liners, and concrete slabs of the cask pit for both units.

3.1.2 Technical Evaluation

The licensee designed the racks in accordance with Section III, Division 1, Subsection NF of the ASME Code, which is consistent with industry practice and acceptable to the staff. The licensee committed to conform to 10 CFR Part 50 Appendix B requirements in the performance of activities relating to material procurement, analysis, fabrication, and installation of the rack modules. The licensee's seismic analysis was performed in accordance with the NRC Standard Review Plan, Section 3.7.1, "Seismic design Parameters," and its UFSAR commitments. The licensee used the loads and loading combination for the rack analysis in accordance with GL 78-11 and NRC Standard Review Plan, Section 3.8.4, "Other Seismic Category I Structures." The staff finds the loads and loading combination used by the licensee acceptable. The staff also finds the analysis results demonstrate that the rack module for both units possess adequate margins of safety with respect to stresses, stability, and fatigue.

The licensee's design of the platform used the same code requirements for the supported racks, which is also acceptable to the staff.

The drop events postulated for the cask pit were found to produce localized damage well within the design limits for the racks, baseplate, baseplate/cell wall welds, liners, and concrete slabs. Therefore, there will be no uncontrollable loss of water inventory in the cask pit. The fuel racks in the cask pit possess adequate margins of safety under the postulated mechanical accidents.

The licensee performed analysis for the cask pit and the affected walls in the FHB in both units for the new design tornado and seismic loads, as a result of upgrading the spent fuel handling crane to single-failure proof, in accordance with its respective UFSAR commitments (ACI 318-63 for Unit 1 and ACI 318-71 for Unit 2). The analysis results demonstrate that the affected FHB walls, the liners, and the concrete slabs of the cask pit are adequate to sustain the new loads. The staff finds the analysis for the cask pit and affected walls acceptable.

3.1.3 Summary

Based on the above discussion, the NRC staff concludes that the licensee used appropriate codes and standards to design and analyze fuel racks and concrete slabs in the cask pit and the affected FHB walls; the licensee made an appropriate commitment to 10 CFR Part 50 Appendix B requirements for its material procurement, analysis, fabrication, and installation of the rack modules; and the analysis results demonstrated that adequate margins of safety exist for the rack, cask pit concrete slabs, and the affected FHB walls. The NRC staff, therefore, concludes that the proposed changes to TS Section 5.6 are acceptable with respect to the cask pit storage rack design.

3.2 Criticality

3.2.1 Regulatory Evaluation

10 CFR Part 50 Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," provides a list of the minimum design requirements for nuclear power plants. According to GDC 62, "Prevention of Criticality in Fuel Storage and Handling," the licensee must limit the

potential for criticality in the fuel handling and storage system by physical systems or processes. The staff reviewed the amendment request to ensure that the licensee complied with GDC 62.

The NRC regulatory requirements for maintaining subcritical conditions in SFPs are provided in 10 CFR 50.68, “Criticality Accident Requirements.” Since 10 CFR 50.68 currently is the licensing basis for St. Lucie’s SFPs, the staff has reviewed the proposed changes against the appropriate parts of the section. For Unit 1, the acceptance criterion for prevention of criticality in the SFP is that the effective neutron multiplication factor (k_{eff}) shall be less than or equal to 0.95 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level. For Unit 2, the acceptance criterion for prevention of criticality in the SFP is that the effective neutron multiplication factor (Keff) shall be less than 1.0 if fully flooded with unborated water and less than or equal to 0.95 when credit is taken for soluble boron, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level.

The NRC defined acceptable methodologies for performing SFP criticality analyses in three documents:

1. NUREG-0800, Standard Review Plan, Section 9.1.2, “Spent Fuel Storage,” Draft Revision 4

2. Proposed Revision 2 to Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis”

3. Memorandum from L. Kopp (NRC) to T. Collins (NRC), “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants.”

The staff used the guidance contained in these documents to assist in its review of the licensee’s amendment request.

3.2.2 Technical Evaluation

In determining the acceptability of the licensee’s amendment request, the staff reviewed three aspects of the licensee’s analyses: (1) the computer codes employed, (2) the methodology used to calculate the maximum k_{eff}, and (3) the criticality analysis results. For each part of the review the staff evaluated whether the licensee’s analyses and methodologies provided reasonable assurance that adequate safety margins, in accordance with NRC regulations, were developed and could be maintained in the St. Lucie SFPs.

3.2.2.1 Computer Codes

The licensee performed the analysis of the reactivity effects for the St. Lucie cask pit storage racks with the MCNP4a code, a continuous energy three-dimensional Monte Carlo code. The MCNP4a code was benchmarked against criticality experiments under conditions which bound the range of variables in the rack designs. The critical benchmark experiments considered the effects of varying fuel enrichment, boron-10 loading, lattice spacing, fuel pellet diameter, and soluble boron concentration. The experimental data are sufficiently diverse to establish that the method bias and uncertainty will apply to St. Lucie storage rack conditions. The licensee determined the MCNP4a code calculation (methodology) bias is 0.0009 with a 95/95 bias uncertainty of +/- 0.0011.

In addition to using the MCNP4a code to perform the criticality analyses, the licensee employed the CASMO-4 code to perform the fuel depletion analyses. The licensee used this two-dimensional multigroup transport theory code to determine the isotopic composition of the spent fuel and determine the reactivity effect of the fuel and rack tolerances. From this code, the licensee determined the reactivity effect (delta-k) for each manufacturing tolerance of the fuel assemblies and storage racks.

The staff reviewed the licensee's application of the codes to determine whether each could reasonably calculate the appropriate parameters necessary to support the maximum k_{eff} analyses. The staff concludes that the licensee's use of the MCNP4a code for calculation of the nominal k_{eff} was appropriate since it was benchmarked against experimental data which bounds the proposed assembly and rack conditions for St. Lucie's SFPs. Additionally, the staff finds that the licensee's use of the CASMO-4 code was acceptable for determining the delta-k for each manufacturing tolerance and performing the fuel depletion analyses.

3.2.2.2 Maximum k_{eff} Calculation Methodology

In accordance with the guidance contained in NUREG-0800, Standard Review Plan, "Spent Fuel Storage," Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," and NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," the licensee performed criticality analyses of its SFP. As stated in its letter dated August 28, 2003, the licensee employed a methodology which combines a worst-case analysis based on the bounding fuel and rack conditions, with a sensitivity study using 95/95 analysis techniques. The major components in this analysis were a calculated k_{eff} based on the limiting fuel assembly, SFP temperature and code biases, and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances. The licensee provided an example calculation to demonstrate the methodology used to combine the data.

In performing its criticality analysis, the licensee first calculated a k_{eff} based on nominal conditions using the MCNP4a code. The licensee determined this k_{eff} from the limiting (highest reactivity) fuel assemblies stored in the SFPs. The licensee analyzed the three types of assemblies currently stored in the St. Lucie SFPs. These assemblies are the Combustion Engineering 14 x 14 (CE 14x14) assembly, the Framatome 14 x 14 (FR 14x14) assembly, and the Combustion Engineering 16 x 16 (CE 16x16) assembly. Only Unit 2's SFP contains the CE 16x16 fuel assemblies, while both units' pools contain the other two assembly types.

The licensee performed its reactivity analyses for various enrichments, cooling times, burnups, and the bounding cladding thicknesses. In performing these calculations, the licensee assumed appropriately conservative conditions such as an infinite radial checkerboard array and a 30-centimeter water reflector in both axial directions. The licensee identified the bounding assemblies as the FR 14x14 assembly for Unit 1 and the CE 16x16 assembly for Unit 2. Additionally, the licensee ensured it had identified the bounding assembly for Unit 2 by performing its depletion analyses with operating parameter assumptions that maximized the plutonium (i.e., provided the maximum residual reactivity) in the spent fuel assemblies. These parameters were average fuel pellet temperature, hot leg moderator temperature, and average core soluble boron concentration.

To the calculated k_{eff}, the licensee added the methodology bias as well as a reactivity bias to account for the effect of the normal allowable range of SFP water temperatures. As stated in the description of the MCNP4a code, the licensee determined the methodology bias from the critical benchmark experiments. For the proposed cask pit rack storage configuration, the licensee analyzed the reactivity effects of the SFP water temperature. The licensee determined that the SFP moderator temperature coefficient of reactivity is negative. The licensee calculated the reactivity bias associated with a decrease in water temperature to 10 degrees Celsius (°C) (50 degrees Fahrenheit (°F)). In its August 28, 2003, letter, the licensee provided additional information to justify not analyzing the temperature bias down to the maximum density of water (4 °C). The licensee stated that the ultimate heat sink for both units is the Atlantic Ocean for which the minimum annual temperature is always above 10 °C in the vicinity of the plant. Additionally, the heat input to the SFP from the spent fuel assemblies keeps the temperature of the pool water elevated. Therefore, the staff agrees that 10 °C is an appropriately conservative minimum temperature for the criticality analyses. The licensee added the temperature reactivity bias to the calculated k_{eff} to provide conservative margin in the calculation.

Finally, to determine the maximum k_{eff}, the licensee performed a statistical combination of the uncertainties and manufacturing tolerances. The uncertainties included the MCNP4a bias uncertainty and the MCNP4a uncertainty. The licensee determined both of these uncertainties to a 95/95 threshold, which is consistent with the requirements of 10 CFR 50.68. In its August 28, 2003, letter, the licensee provided, at the request of the staff, a comprehensive list of the manufacturing tolerances considered as well as the reactivity effect calculated for each. For each tolerance, the licensee used the CASMO-4 code to calculate a delta-k between the nominal condition and the most limiting tolerance condition. By using the most limiting tolerance condition, the licensee calculated the highest reactivity effect possible. This results in conservative margin since the tolerances will always bound the actual parameters. Once the reactivity effects for each of the tolerances were determined, the licensee statistically combined each of the manufacturing tolerances with the 95/95 uncertainties. The staff reviewed the licensee’s methodology for calculating the reactivity effects associated with uncertainties and manufacturing tolerances as well as the statistical methods used to combine these values. The staff finds the licensee’s methods conservative and acceptable.

3.2.2.3 Criticality Analysis Results

The primary purpose of the licensee’s amendment request was to gain the staff’s approval for the addition of the cask pit rack to each unit’s SFP. To demonstrate the acceptability of the proposed changes, the licensee provided the criticality analysis results for storage during both normal and accident conditions. The staff reviewed these results against the regulatory requirements described above.

The licensee’s proposed cask pit racks are defined as a Region 1 storage rack for Unit 1, capable of storing fresh and spent fuel regardless of burnup history and a Region 2 storage rack for Unit 2 which is capable of storing spent fuel meeting certain burnup and enrichment conditions. To demonstrate the acceptability of the proposed storage configurations, the licensee analyzed an appropriately conservative and bounding storage configuration for each unit. The Unit 1 analyzed configuration assumed a fully-loaded cask pit storage rack of fresh fuel (i.e., highest reactivity) assemblies at the maximum Uranium-235 enrichment of 4.5 weight percent. The Unit 2 analyzed configuration assumed a fully-loaded cask pit storage rack of spent fuel with the highest reactivity that meets the licensee's proposed burnup curves (Figure 5.6-1f of the proposed TSs). Additionally, the licensee made other conservative assumptions for each unit's analysis such as neglecting neutron absorption in minor structural members, assuming an infinite radial array, and neglecting the presence of burnable poisons in the fuel assemblies. The staff reviewed each of the assumptions used in the licensee's analyses and agrees that each provides conservative results as well as being consistent with the staff's guidance.

The licensee calculated nominal-condition maximum k_{eff} values for an unborated cask pit rack storage case for each unit. The licensee's results show a maximum Unit 1 k_{eff} of 0.9061 for an unborated case, which is below the licensing basis of 0.95. The licensee's results show a maximum Unit 2 k_{eff} of 0.9154 for an unborated case, which also is below the licensing basis of 1.0.

Additionally, the licensee analyzed accident conditions to ensure regulatory limits were met. The licensee analyzed the following accident conditions: 1) temperature and water density effects, 2) assembly drop, and 3) abnormal location of a fuel assembly. The licensee did not analyze lateral rack movement accidents since the physical design of each SFP's cask area neutronically decouples it from the racks in the remainder of the SFP (Ref. 2). In all of the analyzed accidents, the regulations permit the licensee to credit the soluble boron for reactivity control, thereby offsetting any increase in reactivity. For the analysis of the effects of temperature and water density, the licensee considered an SFP heat-up event since its base case assumed water at a high density (low temperature). The results of the analysis showed that the negative SFP moderator temperature coefficient would cause the reactivity in the cask pit rack to decrease. Additionally, the licensee found that voiding would further reduce the reactivity in the rack. Next, the licensee analyzed both vertical and horizontal assembly drops. The licensee concluded that an assembly dropped horizontally on top of the cask pit rack would be neutronically decoupled from other assemblies in the rack due to a separation distance greater than 12 inches. For a vertical drop accident, the licensee found that minor base plate deformation was possible in the cask pit rack; however, the licensee's results show that the deformation would be minor and the reactivity effects would be statistically insignificant. Finally, the licensee analyzed the cask pit rack for a misloaded assembly. Since the licensee designed the Unit 1 rack to store fresh fuel at the maximum permissible enrichment, no credible misloading event would increase the reactivity of the cask pit rack. However, the Unit 2 cask pit rack is only designed to store spent fuel assemblies. The licensee analyzed the misplacement of a fresh fuel assembly in the Unit 2 cask pit area storage rack. The licensee calculated a maximum k_{eff} of 0.9417 including all biases and uncertainties. This meets the licensing basis requirement to limit the k_{eff} to less than 1.0 under accident conditions (with credit for soluble boron if necessary, pursuant to the Double Contingency Principle). The staff has reviewed the accidents analyzed by the licensee and concludes that the results obtained demonstrate that the cask pit racks are adequately designed to preclude a criticality event during accident conditions.

3.2.3 Summary

The staff reviewed the effects of the proposed changes using the appropriate requirements of 10 CFR 50.68 and GDC 62. The staff found that the licensee's amendment request provided reasonable assurance that under both normal and accident conditions the licensee would be able to safely operate the plant and comply with the NRC regulations. Therefore, the NRC staff concludes that the proposed revisions to TS 5.6 are acceptable with respect to criticality analyses.

3.3 Thermal-hydraulic Considerations

3.3.1 Regulatory Evaluation

The current SFP cooling system design basis requires that for Unit 1, the SFP bulk water temperature be maintained at less than 150 °F during a partial core offload with one operating cooling pump, and for Unit 2 that SFP bulk water temperature will not exceed 150 °F for both partial-core and full-core offloads with one operating cooling pump. As part of the current licensing basis a cycle-specific evaluation is required to be performed for full-core offloads to demonstrate that the SFP bulk temperature will not exceed the 150 °F limit with one cooling pump in operation. The licensee, in its October 23, 2002, submittal, proposed replacing the UFSAR commitment to perform cycle-specific evaluations for planned full-core offload cases with a new design basis, but later, in its August 28, 2003, letter, agreed to continue performing cycle-specific evaluations for planned full-core offloads.

Appendix A of 10 CFR Part 50, GDC 61, specifies, in part, that fuel storage systems shall be designed with residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat removal, and with the capability to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Attachment 2 to Matrix 5 of Section 2.1 of NRC Review Standard RS-001, Revision 0, provides the NRC staff review guidance used to determine the adequacy of SFP cooling capability. It will be reflected in a future update to the Standard Review Plan (NUREG-0800) Section 9.1.3. Section 3.1 of RS-001 states that "the licensee demonstrates adequate SFP cooling capacity by either performing a bounding evaluation or committing to a method of performing outage-specific evaluations." The analysis conditions to be assumed for bounding and cycle-specific analysis are given in section 3.1.1 of RS-001. Section 3.2 of RS-001 provides guidance regarding requirements for adequate makeup supply.

The SFP cooling system is described in Chapter 9 of the St. Lucie UFSAR. Section 9.1.3 provides design basis information for the SFP, including the SFP temperature limits for both normal (planned) and abnormal (emergency) refueling scenarios. The system description, along with the applicable design basis information included in Chapter 9, provides the criteria needed to evaluate the impact that the increased SFP heat load has on the ability of the SFP system to comply with the plant design basis and GDC 61. In meeting the GDC, the licensee should demonstrate that sufficient SFP cooling capacity and make-up sources are available during refueling and time is available prior to pool boiling to supply makeup water following a loss of forced cooling.

3.3.2 Technical Evaluation

The St. Lucie Unit 1 and Unit 2 SFP cooling systems are designed to assure adequate cooling to stored fuel, during routine operation and following normal planned offloads. The SFP cooling system design basis for both units requires that the SFP bulk water temperature be maintained at 150 °F or below (as applicable) following partial-core offloads, assuming a single failure of an active component coincident with a loss of offsite power, and following full-core offloads, with no single failure assumption.

The Unit 1 SFP cooling system is a closed loop system consisting of two full-capacity parallel cooling pumps discharging into a single full-capacity shell and tube heat exchanger. The fuel pool water is drawn from the fuel pool near the surface and is circulated by the fuel pool pumps through the fuel pool heat exchanger where heat is rejected to the component cooling water system. From the outlet of the fuel pool heat exchanger, the cooled fuel pool water is returned to the bottom of the fuel pool via a distribution header at the opposite end of the pool from the intake.

The Unit 2 SFP cooling system is a closed loop system consisting of two half-capacity parallel cooling pumps and two full-capacity shell and tube heat exchangers. The fuel pool water is drawn from the fuel pool near the surface and is circulated through one of the fuel pool heat exchangers where heat is rejected to the component cooling water system. From the outlet of the fuel pool heat exchanger, the cooled fuel pool water is returned to the bottom of the fuel pool via a distribution header.

3.3.2.1 Spent Fuel Pool Heat Up Analysis

The proposed increase in the storage capacity of the SFPs will result in an increase in the maximum decay heat loads for the bounding analysis cases that follow, and a corresponding increase in the SFP cooling loop heat load. The licensee performed some thermal-hydraulic analyses to evaluate the effect of the increased storage capacity on the SFP heat loads and the corresponding SFP water temperature. The following offload/cooling alignment scenarios were evaluated.

3.3.2.1.1 Planned Partial-core Offload with a Single Active Failure

The licensee analyzed the bounding case for the partial-core offload. In this scenario it is assumed that 105 assemblies are offloaded to the SFP, completely filling all storage locations, starting at 120 hours after reactor shutdown. The minimum decay time for the previously offloaded fuel is taken to be 18 months, based on a nominal operating cycle lenght. A cooling system configuration that includes a single active SFP cooling system component failure was considered.

In evaluating the maximum SFP bulk water temperature for Unit 1, the licensee's analysis was based on a cooling configuration that credits operation of one SFP cooling pump and one SFP heat exchanger. The loss of one SFP cooling pump is taken as the single active failure. The licensee used Holtec's Quality Assurance (QA) validated LONGOR computer code to calculate the steady-state decay heat load from previously offloaded fuel, and Holtec's QA-validated BULKTEM computer program to calculate the transient decay heat loads and SFP bulk temperatures. An SFP bulk water temperature of 134.47 °F was calculated for this offload scenario, with a corresponding coincident net heat load of 21.31 MBtu/hr occurring at 137 hours after reactor shutdown, which is below the design basis water temperature limit of 150 °F during refueling.

In evaluating the maximum SFP bulk water temperature for Unit 2, the licensee's analysis was based on a cooling configuration that credits operation of one SFP cooling pump and one SFP heat exchanger. The loss of one SFP cooling pump is taken as the single active failure. An SFP bulk water temperature of 139.58 °F was calculated for this offload scenario, with a corresponding coincident net heat load of 22.20 MBtu/hr occurring at 140 hours after reactor shutdown, which is below the design basis water temperature of 150 °F during refueling.

3.3.2.1.2 Refueling Full-core Offload with a Single Active Failure

The licensee analyzed the bounding case for the planned full-core offload. In this scenario it is assumed that 217 assemblies are offloaded to the SFP, completely filling all storage locations, starting at 120 hours after reactor shutdown. The 217 offloaded assemblies are assumed to consist of 73 assemblies with 4.5 years of irradiation at full power, 72 assemblies with 3 years of irradiation at full power and 72 assemblies with 1.5 years of irradiation at full power. The minimum decay time for the previously offloaded fuel is taken to be 18 months. A cooling system configuration that includes a single active SFP cooling system component failure was considered.

In evaluating the maximum SFP bulk water temperature for Unit 1, the licensee's analysis was based on a cooling configuration that credits operation of one SFP cooling pump and one SFP heat exchanger. The loss of one SFP cooling pump is taken as the single active failure. The licensee used Holtec's QA-validated LONGOR computer code to calculate the steady-state decay heat load from previously offloaded fuel, and Holtec's QA-validated BULKTEM computer program to calculate the transient decay heat loads and SFP bulk temperatures. An SFP bulk water temperature of 161.19 °F was calculated for this offload scenario, with a corresponding coincident net heat load of 37.17 MBtu/hr occurring at 137 hours after reactor shutdown. This temperature exceeds the design basis limit of 150 °F for bulk pool temperature.

The licensee states in the August 28, 2003, supplement, that "prior to commencing a full-core offload, an outage specific engineering evaluation will be performed to evaluate the spent fuel pool coolant temperature while assuming a cooling capacity equivalent to a single train of the SFP cooling system. The procedure will outline the need for contingency actions such as supplemental cooling requirements or fuel offload stop-temperature if this temperature objective cannot be met for the full core offload scenario."

The licensee also evaluated the maximum bulk temperature for Unit 2 for the scenario described above. An SFP bulk water temperature of 165.89 °F was calculated for this offload scenario, with a corresponding coincident net heat load of 36.30 MBtu/hr occurring at 159 hours after reactor shutdown. This temperature exceeds the design basis limit of 150 °F for bulk pool temperature. The current licensing basis for Unit 2 as stated in UFSAR Section 9.1.3.1 requires an "outage-specific calculation to demonstrate that 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. The licensee's August 28, 2003, supplement confirmed continued compliance with this commitment.

3.3.2.1.3 Refueling Full-core Offload with Maximum Cooling

The licensee analyzed the bounding case for the refueling full-core offload. In this scenario it is assumed that 217 assemblies are offloaded to the SFP, completely filling all storage locations, starting at 120 hours after reactor shutdown. The 217 offloaded assemblies are assumed to consist of 73 assemblies with 4.5 years of irradiation at full power, 72 assemblies with 3 years of irradiation at full power and 72 assemblies with 1.5 years of irradiation at full power. The minimum decay time for the previously offloaded fuel is taken to be 18 months. A cooling system configuration that credits flow from both pumps is modeled. No component failures were assumed. For this offload scenario, an SFP bulk water temperature of 125.01 °F, with a corresponding coincident net heat load of 39.38 MBtu/hr occurring at 128 hours after reactor shutdown was calculated for Unit 1 and an SFP bulk water temperature of 142.87 °F, with a corresponding coincident net heat load of 39.81 MBtu/hr occurring at 133 hours after reactor shutdown was calculated for Unit 2. The resulting SFP temperature for both units is below the design basis temperature of 150 °F.

3.3.3 Effects of SFP Boiling

In the unlikely event there is a complete loss of cooling, the SFP bulk water temperature will begin to rise and will eventually reach the boiling temperature.

The licensee has performed analyses to demonstrate minimum time-to-boil and the maximum boil-off rate. Two SFP loss of forced cooling scenarios were evaluated for each unit. The first scenario assumed a partial-core offload, and the second scenario assumed a full-core offload. Both scenarios assumed an instantaneous core offload rate to maximize the decay heat load and minimize the time-to-boil for a loss of forced cooling. The full-core offload scenario is the most limiting case for both units. For Unit 1, the calculated minimum time from loss-of-pool cooling at peak SFP temperature until the pool boils based on the heat load for the full-core offload scenario is 3.33 hours. The corresponding maximum boil off rate is 76.23 gallons per minute (gpm). For Unit 2, the calculated minimum time from loss-of-pool cooling at peak SFP temperature until the pool boils based on the heat load for the full-core offload scenario is 3.1 hours. The corresponding maximum boil off rate is 84.74 gpm.

The licensee states that there is adequate time to align and supply sufficient water, from a variety of sources, to the SFP prior to the time to boil. Makeup sources include, the refueling water storage tank (from the fuel pool purification pump) and the primary water tank (from the primary water pumps). These sources are capable of providing makeup flows of at least 100 gpm. In the event the above freshwater sources of makeup are unavailable, a backup source of makeup water from the essential service water system is also available via a hose connection. Based on the above, the staff finds the time-to-boil analysis acceptable.

3.3.4 Summary

The NRC staff concludes that there is adequate cooling water flow to the SFP heat exchangers to remove the decay heat generated by the spent fuel in the pool during normal and abnormal offload conditions. The staff also finds that the licensee has sufficient time and capability, prior to the onset of boiling, to align makeup water to the pool, and provide makeup at a rate in excess of the boil-off rate, thus satisfying GDC 61. Therefore, the NRC staff concludes that the proposed revisions to TS 5.6 are acceptable regarding thermal-hydraulic considerations.

3.4 Handling of Heavy Loads

3.4.1 Regulatory Evaluation

NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” provides guidelines and recommendations to assure safe handling of heavy loads by prohibiting, to the extent practicable, heavy load travel over stored spent fuel assemblies, fuel in reactor core, safety-related equipment, and equipment needed for decay heat removal.

NUREG-0612 endorses a defense in-depth approach for the handling of heavy loads near spent fuel and safe shutdown systems. General guidelines for overhead handling systems that are used to handle heavy loads in the area of the reactor vessel and SFP are given in Section 5.1.1 of NUREG-0612. They are as follows: (1) definition of safe load paths; (2) development of procedures for load handling operations; (3) training and qualification of crane operators in accordance with Chapter 2-3 of American National Standards Institute (ANSI) B30.2-1976; (4) use of special lifting devices that meet guidelines in ANSI N14.6-1978; (5) installation and use of noncustom lifting devices in accordance with ANSI B30.9-1971; (6) inspection, testing, and maintenance of cranes in accordance with Chapter 2-2 of ANSI B30.2-1976; and (7) design of crane in accordance with Chapter 2-1 of ANSI B30.2-1976 and Crane Manufacturers Association of America document CMAA-70. Section 5.1.2 of NUREG-0612 provides additional guidelines for control of heavy loads in the SFP area of Pressurized-Water Reactors.

3.4.2 Technical Evaluation

The installation of each cask pit rack and platform into the units flooded cask pit will involve handling heavy loads in the vicinity of the SFP. NUREG-0612 provides recommendations and guidelines to assure safe handling of heavy loads in proximity to or over safe shutdown equipment or irradiated fuel in the spent fuel area. The guidelines are meant to ensure that either (1) the potential for a load drop is extremely small, or (2) the consequences of a postulated accidental load drop do not result in the violation of radiological or criticality limits, or compromise safe shutdown.

The licensee states that the spent fuel cask handling crane will be used for lifting the new rack and platforms into the respective Fuel Handing Building. Each unit has an overhead bridge-type crane that is located outdoors at the north end of its respective FHB. An L-shaped door in the FHB roof is located directly over the cask pit area. The spent fuel cask handling crane will be used to lower the platform and rack vertically through the L-shaped door directly into the cask pit. Dry weights for the empty cask pit racks are approximately 17 tons (Unit 1) and 15 tons (Unit 2). The platforms to be installed under the racks weigh approximately 5 tons. The weights of the empty cask pit racks and platforms to be installed are less than the weights of the spent fuel casks (25 tons for Unit 1 and 100 tons for Unit 2) discussed in each unit’s UFSAR.

The licensee stated in its October 23, 2002, submittal that, prior to the cask pit rack installation, both cask handling cranes would be upgraded to designs meeting NUREG-0554, “Single-Failure-Proof Cranes for Nuclear Power Plants.” This would meet the criteria in section 5.1.2 of NUREG-0612, hence eliminating the need to analyze for the consequences of a crane heavy load drop of a rack or platform. In a telephone call with the licensee on September 9, 2003, the

staff requested the licensee to provide a letter certifying that the new cask handling cranes being replaced in accordance with 10 CFR 50.59, “Changes, Tests and Experiments,” would be designed, installed, and load-tested to the single-failure-proof criteria of NUREG-0554. In response to the staff’s request, the licensee stated in a letter (L-2003-310) dated December 11, 2003, that the new cask handling cranes had been installed and the new cranes satisfy the regulatory guidance for single-failure-proof handling systems found in NUREG-0612, Section 5.1.6 and NUREG-0554. Additional information on the cranes and their certification is contained in a letter (L-2003-309) dated December 11, 2003, which was submitted by the licensee for license Amendments 190 (Unit 1) and 134 (Unit 2) dated April 28, 2004.
The staff reviewed the information about the new cranes that the licensee supplied in its December 11, 2003, letters, along with relevant information in the licensee’s October 23, 2002, submittal describing how the cranes are to be used in the installation of the cask pit storage racks. In its October 23, 2002, submittal, the licensee stated, “to prevent submerging the crane’s main hook during rack installation, a temporary hoist with the appropriate capacity will be attached to the main hook, and a Holtec designed lifting rig will be used.” The staff found that the combination of a single-failure-proof crane with a temporary hoist attached to the main hook does not meet the criteria for a single-failure-proof hoisting system. The staff discussed this issue with the licensee, and requested clarification on the proposed use the temporary hoist with the new cranes. In a letter dated March 25, 2004, the licensee stated that they had determined that the temporary hoist was not needed to prevent submerging the main hook. Thus, the lifting rig will be attached directly to the main hook and a temporary hoist will not be used during installation of the cask pit racks for St. Lucie Units 1 and 2. Upon review of the information provided on the new cranes and how they will be used in the cask installation, the staff finds that credit taken for use of a single-failure-proof crane is acceptable.

3.4.3 Summary

Based on the review of the licensee’s submitted information concerning the handling of heavy loads associated with this amendment request, the staff finds the licensee has provided adequate assurance that its planned actions for the installation of the cask pit storage racks are consistent with the “defense-in-depth” approach to safety in the handling of heavy loads described in NUREG-0612. The staff believes that the use of the spent fuel cask handling crane, coupled with the lifting rig, will enable the licensee to maintain safety during the handling of heavy loads associated with the installation of the cask pit storage racks. Therefore, the NRC staff finds the amendment request acceptable regarding the handling of heavy loads.

3.5 Radiological Consequences

Section 15.4 of the St. Lucie Unit 1 UFSAR and Section 15.7.4 of the Unit 2 UFSAR describe the design basis fuel handling accident (FHA) at St. Lucie Units 1 and 2, respectively, and evaluate the radiological consequences of a single fuel assembly drop inside the FHB, which includes the cask pit area. The staff evaluated the potential effect of the cask pit rack addition on the FHA radiological consequence. The new racks do not require any changes in the fuel assembly design. The assumptions and parameters including FHA source term remain the same. There are no new fuel movement pathways created by the addition of the cask pit racks, since fuel assembly movement to and from the new cask pit rack will take essentially the same path as fuel movement required to load a spent fuel transfer cask. The frequency of fuel assembly movement will be essentially the same as without the cask pit rack up to the time that the current pool capacity to accommodate a full core offload expires. The addition of the new racks will extend the full-core offload capability by approximately 3 years, allowing additional fuel storage movement rather than fuel transfer movement to other storage facilities. However, the frequency of fuel movement would be approximately the same during this period for either the storage or fuel transfer scenario.

Therefore, the probability of an FHA will not be increased by the addition of the cask pit racks. The staff concludes that the installation and use of the cask pit racks will not change the probability, frequency or radiological consequences of the FHA described in the UFSAR. The staff finds the proposed amendments are acceptable.

The staff notes that subsequent to submitting the proposed amendments, the licensee conducted control room leakage testing in September 2003 in response to GL 2003-01, “Control Room Habitability.” In a letter dated December 9, 2003, the licensee reported that the testing indicated that unfiltered in-leakage exceeded the values assumed in the UFSAR analyses for both units. This would result in a higher radiological dose to control room operators following an accident.

In its letter, the licensee also stated that it had performed an operability assessment pursuant to GL 91-18, “Information to Licensees Regarding Resolution of Degraded and Nonconforming Conditions and On Operability,” which concluded that the radiological doses to the control room operators due to the higher in-leakage would not exceed the dose acceptance criteria specified in GDC 19, “Control Room.” Thus, the control rooms are considered degraded but operable, and operation may continue while corrective actions are taken. As part of its corrective actions, the licensee submitted license amendment requests to implement full-scope alternative source term in order to revise design basis accident radiological consequence analyses (FHA included) for St. Lucie Units 1 and 2 and to bring the plant into alignment with the dose analysis in regard to as-found control room envelope unfiltered air in-leakage. These license amendments are currently under staff review. The staff concludes that the higher radiological consequences are the result of a degraded condition of the control rooms and are not related to the installation of the cask pit racks. The condition does not change the acceptability of the proposed amendments.

3.6 Maintaining Spent Fuel Pool Offload Capability

The purpose for installing the cask pit racks is to provide additional storage capacity for spent fuel to allow refueling outage fuel offloads and non-outage fuel shuffles. In addition, for Unit 1, the rack would be used to temporarily stage new fuel prior to a refueling outage. In Section 1.0 of the October 23, 2002, license amendment request, the licensee stated, “because the cask pits will eventually be needed for loading fuel into transfer casks, the cask pit racks will be removed, cleaned, and stored in an alternate location prior to any spent fuel cask loading operations.” The staff requested additional information from the licensee as to what measures will be taken to ensure that the capability to unload spent fuel to a cask is not lost. In its letter dated August 28, 2003, the licensee said it would impose a restriction that will assure the capability to unload and remove the cask pit racks when cask loading operations are necessary. The licensee agreed to the incorporation of an appropriate license condition and suggested the following wording:

> The licensee shall restrict the combined number of fuel assemblies loaded in the SFP storage racks and cask pit rack to no more than the capacity of the SFP

> storage racks at all times except during a reactor offload/refueling condition. This restriction will assure the capability to unload and remove the cask pit rack when cask loading operations are necessary.

The NRC staff concludes that the proposed license condition provides adequate assurance that the ability to offload spent fuel to a cask is maintained. Approval of the proposed amendment requests will be subject to the above condition.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has previously been prepared and published in the *Federal Register* on July 9, 2004 (69 FR 131). Accordingly, based on the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect upon the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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