May 6, 1999

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: ST. LUCIE, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING SPENT FUEL POOL STORAGE CAPACITY; SOLUBLE BORON CREDIT (TAC NO. MA0666)

Dear Mr. Plunkett:

The Commission has issued the enclosed Amendment No. 101 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the St. Lucie Unit 2 Technical Specifications (TS) in response to your application dated December 31, 1997 as supplemented May 15, 1998, September 15, 1998, November 25, 1998, and January 28, 1999. The May 15, 1998 supplement was a result of an NRC request for additional information dated April 8, 1998.

This change modified the St. Lucie, Unit 2, TS to increase the capacity of the spent fuel storage pool, in part, by allowing a credit for a certain soluble boron concentration in the spent fuel pool. Following the receipt of the supplement dated November 25, 1998, and the staff's subsequent no significant hazards consideration determination, the supplement dated January 28, 1999, contained clarifying information that did not change the no significant hazards consideration determination.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

William C. Gleaves, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-389

Enclosures:
1. Amendment No. 101 to NPF-16
2. Safety Evaluation

cc w/enclosures: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 6, 1999

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

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Wm C Gl

William C. Gleaves, Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
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Docket No. 50-389

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. NPF-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company (FPL), dated December 31, 1997, and supplemented May 15, 1998, September 15, 1998, November 25, 1998, and January 28, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-16.

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 101 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specification.

3. This license amendment is effective as of its date of issuance and shall be implemented by the end of the next scheduled refueling outage, which is currently scheduled to begin in April of 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Sheri R. Peterson

Sheri R. Peterson, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 6, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 101

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following page of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Page	Insert Page
IX	IX
XXII	XXII
3/4 9-12	3/4 9-12
B 3/4 9-3	B 3/4 9-3
5-4	5-4
5-4A	5-4A thru 5-4F

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIST OF FIGURES (Continued)

3/4.9.11 SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The Spent Fuel Storage Pool shall be maintained with:

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

b. The fuel storage pool boron concentration greater than or equal to 1720 ppm.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

a. With the water level requirement not satisfied, immediately suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore fuel storage pool boron concentration to within the required limit.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

4.9.11.1 Verify the fuel storage pool boron concentration is within limit at least once per 7 days.

BASES

3/4.9.10 and 3/4.9.11 WATER LEVEL-REACTOR VESSEL and SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

The limit on soluble boron concentration in LCO 3/4.9.11 is consistent with the minimum boron concentration specified for the RWT, and assures an additional subcritical margin to the value of k_{eff} which is calculated in the spent fuel storage pool criticality safety analysis to satisfy the acceptance criteria of Specification 5.6.1. Inadvertent dilution of the spent fuel storage pool by the quantity of unborated water necessary to reduce the pool boron concentration to a value that would invalidate the criticality safety analysis is not considered to be a credible event. The surveillance frequency specified for verifying the boron concentration is consistent with NUREG-1432 and satisfies, in part, acceptance criteria established by the NRC staff for approval of criticality safety analysis methods that take credit for soluble boron in the pool water. The ACTIONS required for this LCO are designed to preclude an accident from happening or to mitigate the consequences of an accident in progress, and shall not preclude moving a fuel assembly to a safe position.

3/4.9.12 SPENT FUEL CASK CRANE

The maximum load which may be handled by the spent fuel cask crane is limited to a loaded multi-element cask which is equivalent to approximately 100 tons. This restriction is provided to ensure the structural integrity of the spent fuel pool in the event of a dropped cask accident. Structural damage caused by dropping a load in excess of a loaded multi-element cask could cause leakage from the spent fuel pool in excess of the maximum makeup capability.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,931 ± 275 cubic feet at a nominal T_{avg} of 572° F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel pool and spent fuel storage racks shall be maintained with:

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 520 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

3. A nominal 8.96 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1 b. Fuel placed in Region I of the spent fuel storage racks shall be stored in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fresh fuel shall have a nominal average U-235 enrichment of less than or equal to 4.5 weight percent.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

4. The reactivity effects of fuel assembly burnup and decay time may be considered as specified in Figures 5.6-1c through 5.6-1e.

5.6.1 c. Fuel placed in Region II of the spent fuel storage racks shall be placed in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fuel placed in Region II shall meet the burnup and decay time requirements specified in Figure 5.6-1a or 5.6-1b.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

CRITICALITY (continued)

5.6.1 d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U-235 enrichment less than or equal to 4.5 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

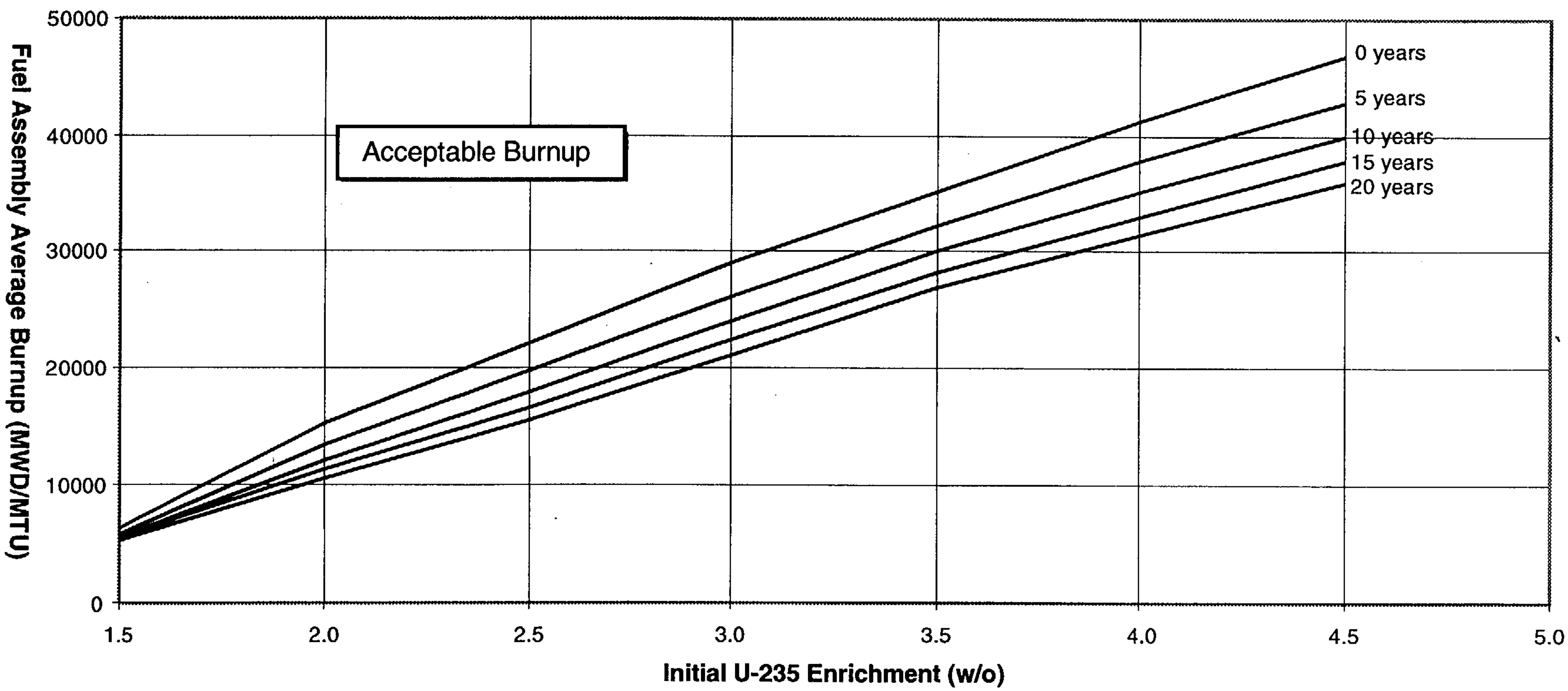
CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1360 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

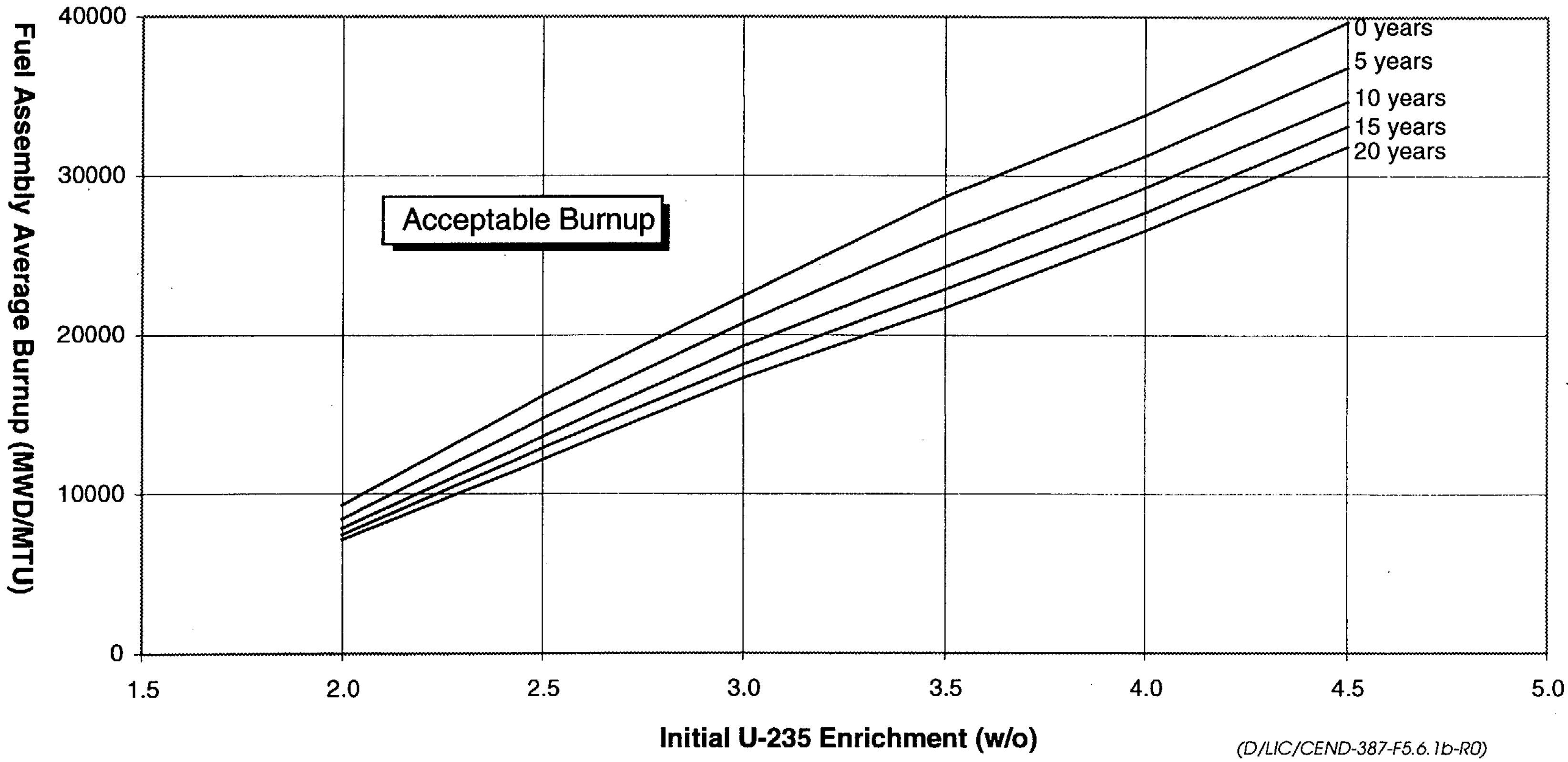
5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

Figure 5.6-1a
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region II, 1.3 w/o



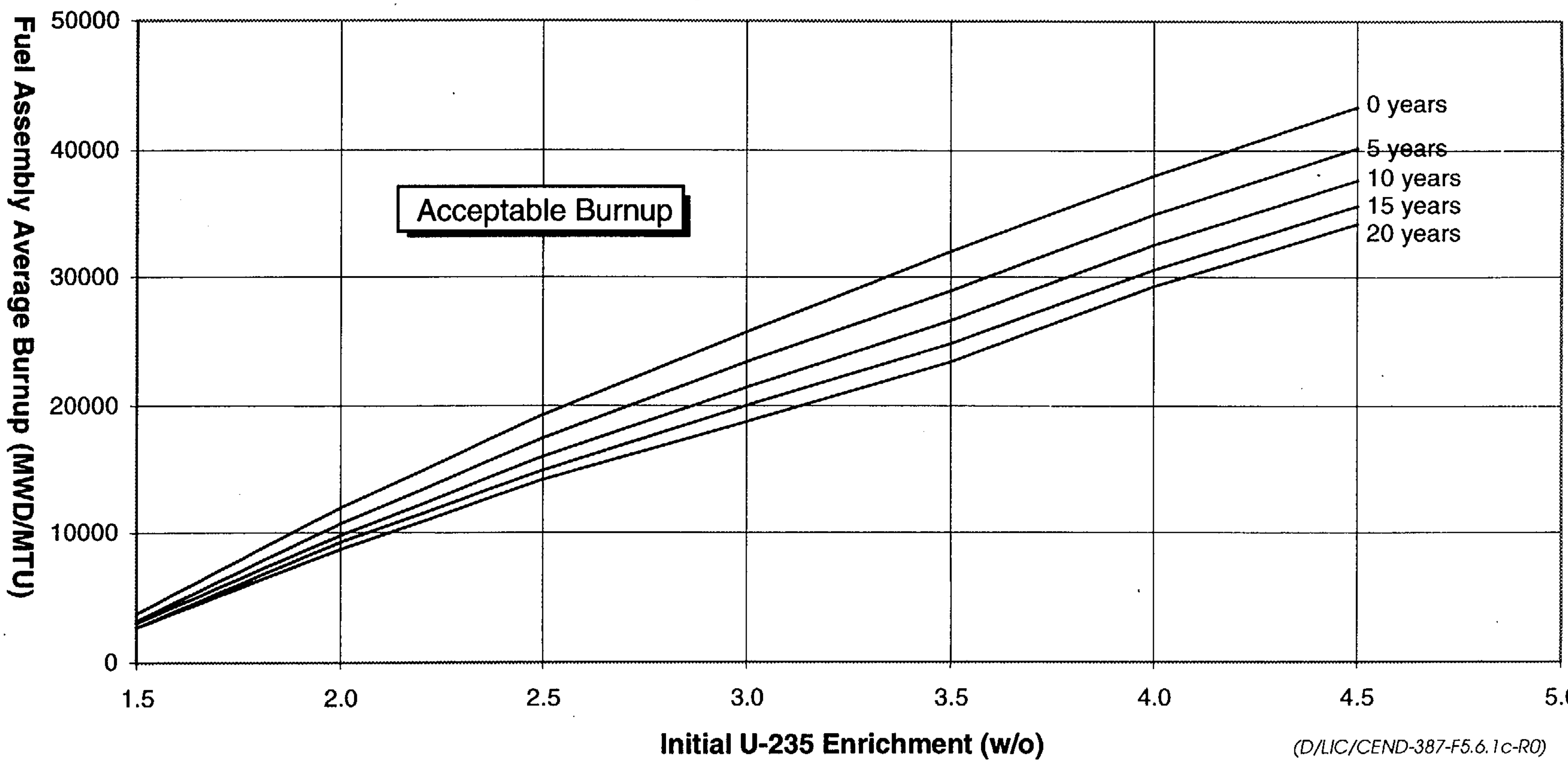
(D/LIC/CEND-387-F5.6.1a-R0)

Figure 5.6-1b
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region II, 1.5 w/o



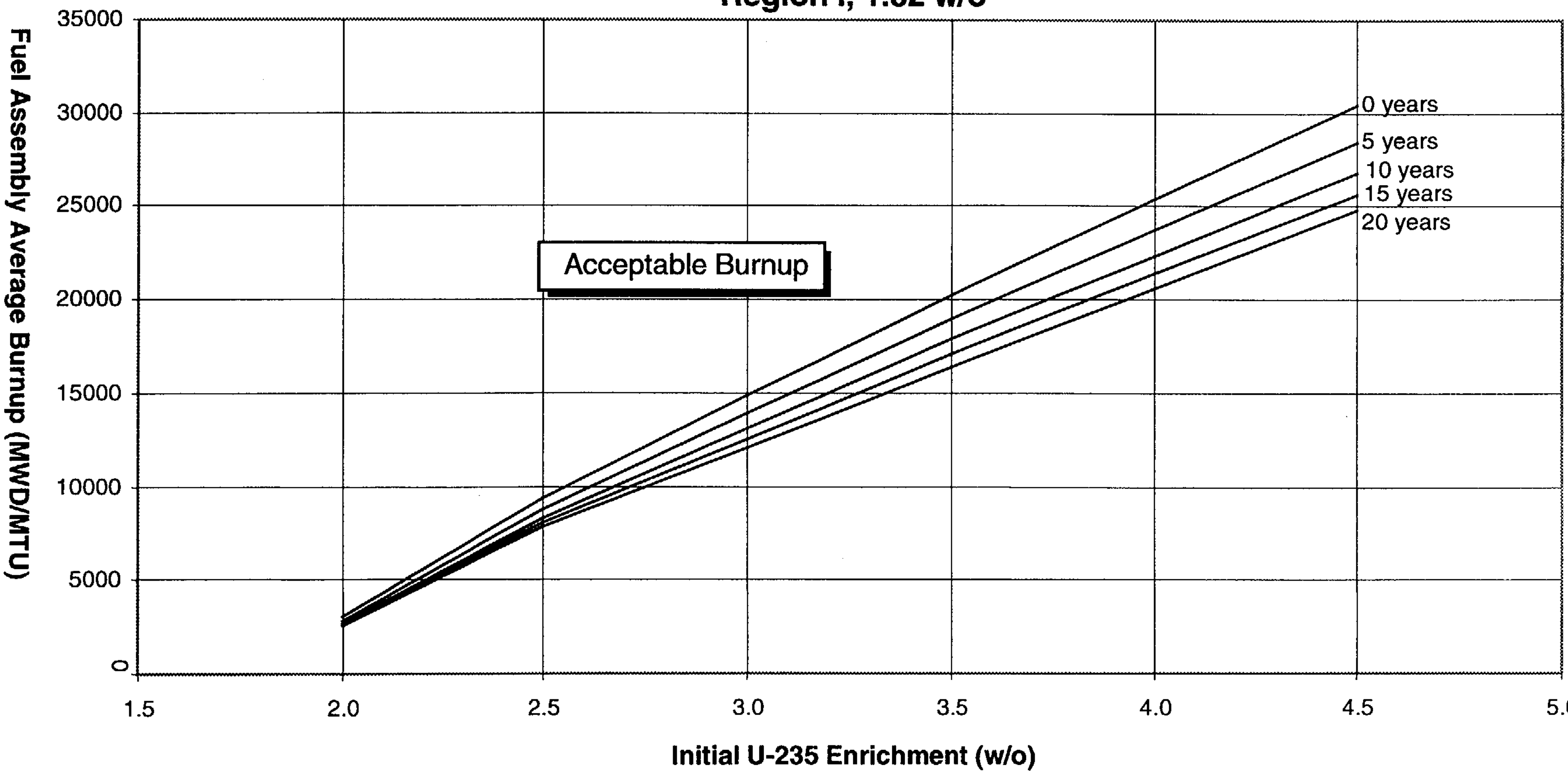
(D/LIC/CEND-387-F5.6.1b-R0)

Figure 5.6-1c
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region I, 1.4 w/o



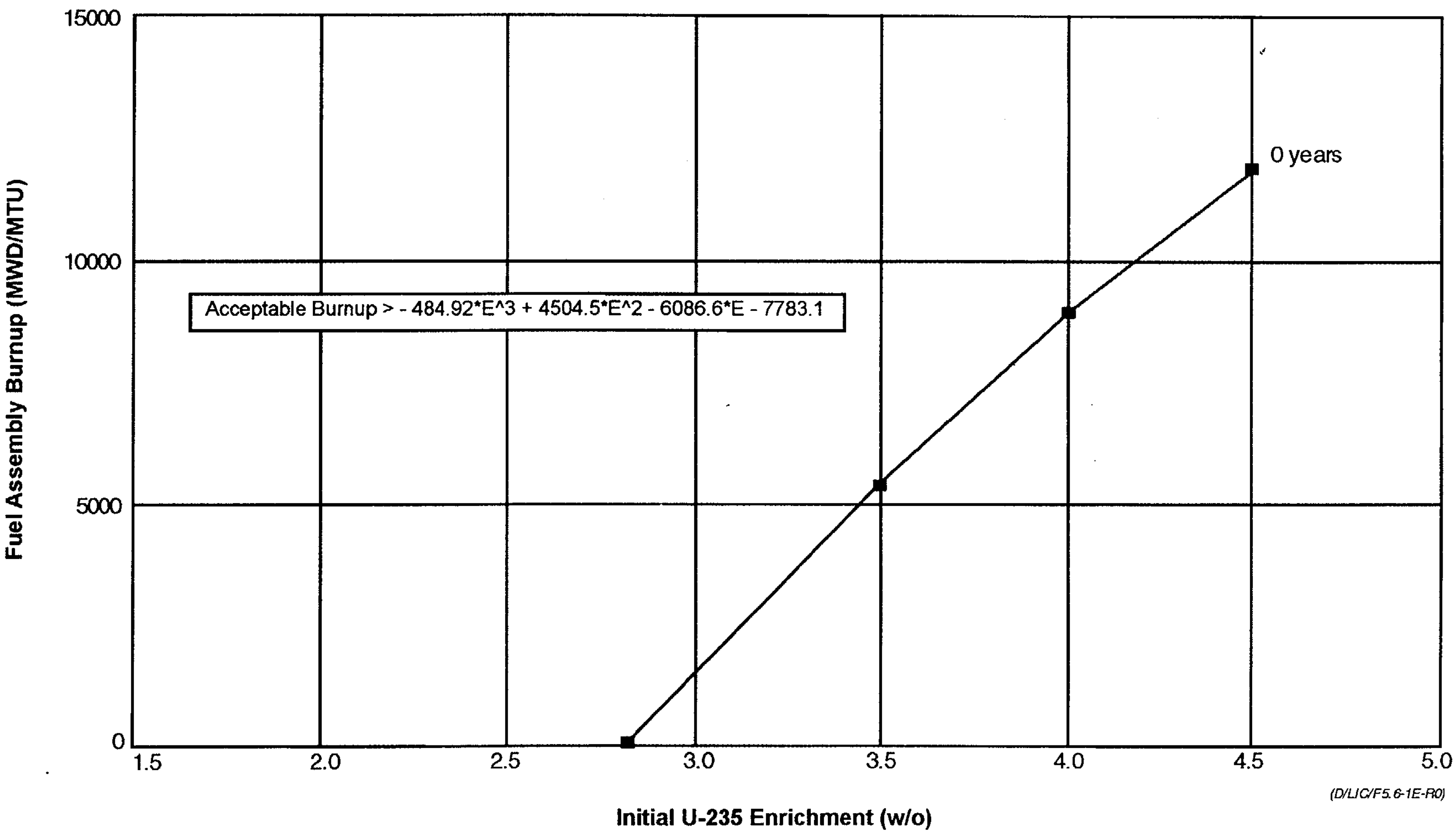
(D/LIC/CEND-387-F5.6.1c-R0)

Figure 5.6-1d
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region I, 1.82 w/o



(D/LIC/CEND-387-F5.6.1d-R0)

Figure 5.6-1e
Required Fuel Assembly Burnup vs Initial Enrichment
Region I, 2.82 w/o



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 101 TO FACILITY OPERATING LICENSE NO. NPF-16

REGARDING BORON CREDIT IN THE SPENT FUEL POOL

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

1.0 INTRODUCTION

In a letter of December 31, 1997 (Ref. 1), supplemented by letters dated May 15, 1998 (Ref. 2), September 15, 1998, November 25, 1998, and January 28, 1999, Florida Power and Light Company (FPL, the licensee) requested changes to the St. Lucie, Unit 2, Technical Specifications (TS) to allow the use of credit for soluble boron in the spent fuel pool criticality analyses. These criticality analyses were performed using methodology analogous to that developed by the Westinghouse Owners Group and described in WCAP-14416-NP-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology" (Ref. 3).

The spent fuel pool (SFP) at St. Lucie Plant, Unit 2, contains 1584 spent fuel storage cells, of which 1076 are currently allowed for storage by the technical specifications. FPL estimates that by the year 2001, St. Lucie, Unit 2, will have the allowed storage cells filled, with the exception of those reserved for a full-core offload. The licensee proposed to increase the allowed storage capacity of the SFP to 1360 fuel assemblies and to credit the use of soluble boron in the SFP for criticality control. Accordingly, by letter dated December 31, 1997, the licensee requested an amendment to Facility Operating License No. NPF-16 for St. Lucie, Unit 2. The amendment, in part, proposed changes to the TS to move the boron concentration requirement from Section 5.6 to Section 3/4.9.11; to establish the SFP boron concentration surveillance requirement to reflect the credit for soluble boron in Section 3/4.9.11; and to reflect the increased SFP storage capacity in Section 5.6.3.

Following the receipt of the supplement dated November 25, 1998, and the staff's subsequent no significant hazards consideration determination (63 FR 69340), the supplement dated January 28, 1999, contained clarifying information that did not change the no significant hazards consideration determination. An additional notice was required, in accordance with 10 CFR 2.1107, due to an oversight (64 FR 16502, April 5, 1999).

2.0 EVALUATION

2.1 Criticality Evaluation

The St. Lucie, Unit 2, spent fuel storage racks were analyzed using the criticality methodology which has been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) (Ref. 3). This methodology takes partial credit for soluble boron in the fuel storage pool criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

1) k_{eff} shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95% probability, 95% confidence (95/95) level; and

2) k_{eff} shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level.

The analysis of the reactivity effects of fuel storage in the St. Lucie, Unit 2, spent fuel racks was performed with SCALE-PC, a personal computer version of the SCALE-4.3 code package (which includes KENO-Va, NITAWL, CSAS-2, and BON-AMI), with the updated 44-group ENDF/B-V cross section library. Since the KENO-Va code package does not have burnup capability, depletion analyses were made with the two-dimensional integral transport theory code, DIT, which uses an 89-group structure collapsed from the ENDF/B-VI library. The SCALE-PC models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the St. Lucie, Unit 2, racks are designed and have been found to adequately reproduce the critical values. The selected critical experiments included the Babcock & Wilcox experiments carried out in support of close proximity storage of power reactor fuel and the Pacific Northwest Laboratory program carried out in support of the design of fuel shipping and storage configurations. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to St. Lucie Unit 2 storage rack conditions. The DIT code and its cross section set have been used in the design of reload cores and extensively benchmarked against operating reactor history and test data. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the St. Lucie, Unit 2, storage racks with a high degree of confidence.

The St. Lucie, Unit 2, spent fuel storage racks have previously been qualified for storage of fuel assemblies with maximum nominal enrichments up to 4.5 weight percent (w/o) U-235. Region 1 of the pool consists of six rack modules containing a total of 448 storage cells with the assemblies arranged in a checkerboard pattern. Region 1 can be used to store fuel which has a U-235 enrichment less than or equal to 4.5 w/o. Region 2 consists of 13 modules containing a total of 1136 storage cells with the assemblies stored in a 3-out-of-4 pattern. Region 2 can be used to store fuel which has achieved the burnup requirements stated in the TS. Cell blocking devices are inserted to prevent inadvertent assembly storage in cells that are required to be empty. Thus, the existing racks contain a total of 1584 cells, of which 1076 are currently available for storage. The racks contain no absorber materials for reactivity holddown (e.g., Boraflex or boral).

The proposed Region 1 storage configuration is shown in Figure 9 of CENPD-387 (Ref. 4). Region 1 would continue to use flux traps to minimize k_{eff} by placing fuel assemblies next to regions of water. The proposed Region 2 storage configuration is shown in Figure 10 of CENPD-387, and would increase the storage density of the region to 95.4% from the current value of 75%. Region 2 storage would continue to require a minimum value of assembly burnup as a function of initial enrichment.

The following assumptions were used in the rack reactivity calculations to support an increase in the number of fuel assemblies that may be stored in the fuel pool from the current limit of 1076 to a new value of 1360. The fuel assemblies contain UO_2 over the entire length of each fuel rod. All fuel assemblies contain 236 fuel rods in a 16x16 fuel rod array. The moderator was assumed to be pure water (for comparison to the 1.0 k_{eff} limit) or water containing 350 ppm soluble boron (for comparison to the 0.95 k_{eff} limit) at a temperature of 50°F.

A methodology bias (determined from benchmark calculations) as well as a reactivity bias to account for the effect of the normal range of spent fuel pool water temperatures (50°F to 155°F) were included. Uncertainties due to tolerances in fuel enrichment and density, storage cell inner diameter, storage cell pitch, stainless steel thickness, assembly position, calculational uncertainty, and methodology bias uncertainty were accounted for. In addition, a control element assembly (CEA) uncertainty was included when the reactivity effect of CEAs placed in fuel assemblies was considered. Manipulation of both fuel and CEAs in the spent fuel pool is performed using the spent fuel handling machine hoist. The grapple tool used to handle CEAs has a different design than the grapple tool used to handle fuel assemblies. As a result, a grapple change must be made following any fuel manipulation in order to initiate a CEA repositioning campaign. Additionally, neither fuel assemblies nor CEAs stored in the pool may be handled or repositioned except as described in plant specific procedural guidance. Following the completion of fuel handling evolutions and inventory verification the handling machine is de-energized. The NRC, therefore, concludes that inadvertent CEA removal from stored fuel assemblies is precluded and finds the use of CEAs stored in fuel assemblies acceptable for reactivity holddown. These uncertainties were appropriately determined at the 95/95 probability/confidence level. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

The highest value of k_{eff} was obtained for the Region 2 configuration at 50°F with no soluble boron assumed in the pool water and was used as the base case for conservatism. The 95/95 k_{eff} was then determined by adding the temperature and methodology biases and the statistical sum of independent tolerances and uncertainties to the nominal k_{eff} values, as described in Reference 2. This resulted in a 95/95 k_{eff} of 0.99801. Since this value is less than 1.0 and was determined at a 95/95 probability/confidence level, it meets the NRC criterion for precluding criticality with no credit for soluble boron and is acceptable.

Soluble boron credit is used to provide safety margin by maintaining k_{eff} less than or equal to 0.95 including 95/95 uncertainties. The pool water was assumed to be borated to 350 ppm. As previously described, the individual tolerances and uncertainties, and the temperature and methodology biases, were added to the calculated nominal k_{eff} to obtain a 95/95 value. The resulting 95/95 k_{eff} was 0.94797. Since k_{eff} is less than 0.95 with 350 ppm of boron and uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion for precluding criticality is satisfied. This boron value is well below the minimum spent fuel pool boron concentration value of 1720 ppm required by proposed TS 3.9.11.b (previously TS 5.6.1.a.3) and is, therefore, acceptable.

The concept of reactivity equivalencing due to fuel burnup was used to define the conditions under which fresh and irradiated fuel assemblies are interchangeable on an overall reactivity basis. The NRC has previously accepted the use of reactivity equivalencing to equate an array of fresh fuel assemblies and their enrichments, that have been shown to be acceptable for storage, into an array of irradiated assemblies with different initial enrichments, decay times, and burnable absorber concentrations. To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ for storage of fuel assemblies with enrichments higher than those acceptable for storage of fresh assemblies, a series of reactivity calculations were performed to generate a set of enrichment versus fuel assembly discharge burnup ordered pairs that all yield an equivalent k_{eff} when stored in the spent fuel storage racks. For conservatism, the nuclide inventory assumed no Xe-135, peak Sm-149, and peak Pu-239, thereby maximizing the assembly reactivity.

The analysis also included spent fuel decay time credit, which results from the radioactive decay of isotopes in the spent fuel to daughter isotopes. The loss in reactivity due to the radioactive decay of the spent fuel results in reducing the minimum burnup needed to meet the reactivity requirements. The reactivity of an irradiated fuel assembly will decrease following its discharge from the reactor and the decay of short lived fission products due to the decay of actinides and long half-life fission products. For long cooling periods, the decay of Pu-241, with a half-life of approximately 14 years, to Am-241 is the most important contribution to a reduction in fuel assembly reactivity. The minimum required burnup for initial assembly enrichments between 1.5 and 4.5 w/o U-235 for each assembly type in the spent fuel pool are shown in TS Figures 5.6-1a thru 5.6-1e. These burnup-enrichment pair data also include credit for actinide decay from 0 to 20 years.

Uncertainties associated with burnup credit include a reactivity uncertainty of 0.005 Δk at 30,000 MWD/MTU applied linearly to the burnup credit requirement to account for calculational and depletion uncertainties. Benchmarking of the design codes for several operating cycles demonstrated a small reactivity bias, that is bounded by the bias of 0.005 Δk per 30,000 MWD/MTU. An uncertainty of 5% was also applied to the calculated burnup to account for burnup measurement uncertainty. The NRC staff concludes that these uncertainties conservatively reflect the uncertainties associated with burnup calculations and are acceptable.

Fuel assemblies discharged from St. Lucie, Unit 2, have nearly symmetric axial burnup distributions with a slight top-peaked bias. As a result, there are minor differences between axially uniform isotopic distributions and the actual assembly nuclide distributions. A slightly top-peaked Pu-241 distribution will decay into a slightly top-peaked Am-241 distribution. This non-uniformity in isotopic distribution results in a small non-conservative reactivity effect relative to the uniform axial distributions that were assumed in the decay calculations. However, the assumption of a constant 1000 ppm soluble boron and 1200°F fuel temperature used in the depletion analyses results in a hardened spectrum, that leads to an overprediction of the fissile content and to an overprediction of the reactivity in the spent fuel pool configuration. These conservative assumptions embedded in the model more than compensate for any postulated axial isotopic distribution effect within the maximum exposure of 40,000 MWD/MTU considered.

The amount of additional soluble boron that is needed to account for these reactivity equivalencing uncertainties is 170 ppm. Adding this to the soluble boron credit of 350 ppm required for k_{eff} to be less than or equal to 0.95 results in a total soluble boron credit of 520 ppm. This value is well below the minimum spent fuel pool boron concentration value of 1720 ppm required by proposed TS 3.9.11.b (previously TS 5.6.1.a.3) and is, therefore, acceptable.

The reactivity analysis described above was performed assuming that all fresh fuel contained no burnable absorbers and that the maximum fresh fuel enrichment is 4.5 w/o U-235. If the reactivity hold-down due to the presence of burnable absorbers such as gadolinia is considered, then the fresh fuel enrichment can be increased above 4.5 w/o until the reactivity matches that of an unshimmed 4.5 w/o assembly. FPL has presented analyses showing that the reactivity of a gadolinium-bearing assembly enriched to 5.0 w/o U-235 is always equal to or lower than that of a fresh 4.5 w/o unshimmed assembly. Therefore, a gadolinium assembly can be stored in the locations reserved for fresh 4.5 w/o U-235 unshimmed assemblies. However, FPL is not requesting an increase in the spent fuel pool TS enrichment limit above 4.5 w/o U-235 at this time.

Although most accidents will not result in a reactivity increase, two accidents can be postulated that could increase reactivity beyond the analyzed conditions. The first would be a loss of fuel pool cooling system and a rise in pool water temperature from 155°F to 240°F. The second accident would be a misload of an assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied. A comparison of reactivity values for these two types of accidents have shown that the potential increase in k_{eff} due to a misloaded fuel assembly is substantially greater than the increase in k_{eff} due to a loss of all fuel pool cooling. Calculations have shown that the misloading of a fresh, unrodded and unshimmed 4.5 w/o fuel assembly into a cell location required to contain water (empty cell) results in the highest reactivity increase. The reactivity increase requires an additional 746 ppm of soluble boron to maintain $k_{eff} \leq 0.95$. However, for such events, the double contingency principle can be applied. This states that the assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident. Therefore, the minimum amount of boron required by proposed TS 3.9.11.b (previously TS 5.6.1.a.3) (1720 ppm) is more than sufficient to cover any accident and the presence of the additional boron above the concentration required for normal conditions and reactivity equivalencing (520 ppm maximum) can be assumed as a realistic initial condition since to assume it is not present would be to assume a second unlikely event.

FPL proposes to modify TS 5.6 to permit an increase in the storage capacity of the spent fuel pool storage racks from 1076 to 1360 assemblies. TS Figure 5.6-1 would be removed and new Figures 5.6-1a through 5.6-1e would be added to describe the assembly burnup requirements for Region 1 and 2 of the spent fuel pool. The TS changes proposed are consistent with the revised criticality analysis evaluated above. Based on this consistency and on the use of approved methodology, the staff finds these TS changes acceptable.

2.2 Spent Fuel Storage Capacity Expansion Evaluation

2.2.1 Spent Fuel Pool Cooling System

The SFP cooling and purification system (SFPCS) is designed to remove the decay heat generated by stored spent fuel assemblies and to clarify and purify the water in the SFP. The cooling portion of the SFPCS contains two full-capacity heat exchangers and two half-capacity pumps that operate in parallel. Each pump is powered from a separate motor control center and has a design flow rate of 1500 gpm. Normally one heat exchanger and one pump are in service. The licensee considers the cooling capacity of the normal in-service equipment to be the equivalent of one train of the SFPCS. The design heat removal for a SFP heat exchanger with both pumps operating is 32.0E+06 Btu/hr. Suction for the system is taken from near the top of the pool and is returned near the bottom. The licensee states in the submittal that normal refueling outages will employ full-core offloads. As such, the St. Lucie, Unit 2, design basis for the SFP bulk water temperature is that it is maintained at or below 150°F given that one pump and one heat exchanger are operating. The purification portion is designed to remove soluble and insoluble foreign matter from the fuel pool water and dust from the pool surface. This maintains the pool water purity and clarity, permitting visual observation of underwater operations.

2.2.2 Decay Heat Load Limits

As a result of an increase in the spent fuel assemblies (SFA) planned to be stored in the SFP, the decay heat load will increase. While a full core off-load is considered to be normal, the licensee performed a bounding heat load calculation for a partial-core offload that assumed 96 SFA were discharged after 5 days of decay and 1394 previously discharged SFA were stored in the SFP. The licensee bounded the heat load by using an end-of-life scenario and assumed the heat load from 1492 SFA in the SFP, which exceeds the proposed capacity of 1360 SFA. The licensee calculated a heat load of 19.76E+06 Btu/hr for a partial-core offload. With one pump and one heat exchanger operating, the peak SFP bulk temperature was calculated to be 139.8°F. With both cooling portions or equivalent trains operating, the peak SFP bulk temperature was calculated to be 130.8°F. These temperatures are below the design temperature and meet the guidance of Standard Review Plan (SRP) Section 9.1.3 for the cooling system and remain below the St. Lucie design basis for the SFP bulk water temperature of 150°F.

The staff performed confirmatory decay heat load calculations. The staff's confirmatory calculations verified the SFP bulk water temperature would remain below 150°F assuming the operation of one SFPCS pump and one heat exchanger for the partial-core offload scenario.

Since the licensee considers that normal refueling practices are full-core offloads, the SFP bulk water temperature needs to be maintained below its design-basis temperature of 150°F given one SFPCS pump and one heat exchanger are operating. To ensure this for full-core offloads, the licensee has committed in a letter dated January 28, 1999, to perform outage-specific calculations to demonstrate that the SFP bulk temperature will not exceed the St. Lucie design-basis temperature of 150°F given the operation of one SFPCS pump and one heat exchanger.

Based on its review, the staff finds that the design and operation meets the guidance of the SRP for SFPs and the January 28, 1999 commitment; therefore, the proposed increase in spent fuel capacity is acceptable.

2.2.3 Effects of SFP Boiling

In the event that there is a complete loss of cooling capability of using SFPCS heat exchangers to remove heat from the SFP, the SFP water temperature will begin to rise and eventually will reach the boiling temperature. In the SFP boiling analysis in Final Safety Analysis Report (FSAR) Table 9.1-8, the licensee states that the time to reach boiling for a full-core offload is greater than 2.9 hours with a boil-off rate of 66.3 gpm. Two permanent, non-seismic pool makeup systems are provided. The first can use the fuel pool purification pump to draw water from the refueling water tank at a rate of 150 gpm. The second, the primary water system, can provide makeup at a rate of 100 gpm. A seismic Category I backup system is available. A hose connection is provided on each header for the intake cooling water system, and a seismic standpipe is located in the fuel handling building on the fuel handling deck. The intake cooling water system via this hose connection can supply sea water at a rate of 66.3 gpm.

The SFP boiling analysis for the full-core offload has not changed from the current licensing basis as stated in the FSAR. The licensee will ensure it remains within its heat load removal capacity for a full-core offload by performing outage-specific calculations. While the partial-core offload heat load will increase, the full-core offload heat load bounds the heat load for the partial-core offload; therefore, the result of the current boiling analysis is acceptable.

2.3 Boron Dilution Analysis

The licensee followed the methodology in accordance with the NRC Safety Evaluation (Ref. 5) of the Westinghouse methodology described in WCAP-14416-A (Ref. 6) to demonstrate the use of soluble boron credit in the SFP. In following this methodology, the licensee performed this analysis to ensure that sufficient time is available to detect and mitigate the dilution prior to exceeding the 0.95 k_{eff} design basis. The analysis was provided on December 31, 1997, and supplemental information was provided on May 15, September 15, November 25, 1998, and January 28, 1999. Potential plant events were quantified to show that sufficient time is available to enable adequate detection and suppression of any dilution event.

Based on the analysis, the licensee determined that a soluble boron concentration of 520 parts per million (ppm) was required to maintain k_{eff} below 0.95. Additionally, deterministic dilution event calculations were performed for St. Lucie, Unit 2, to define the dilution times and volumes necessary to dilute the SFP from the minimum TS boron concentration of 1720 ppm to a soluble boron concentration of 520 ppm for the SFP water inventory of 300,070 gallons. Assuming a well-mixed pool, the volume required to dilute the SFP from the TS limit of 1720 ppm to 520 ppm was determined to be 358,959 gallons. The various events that were considered included dilution from the primary water system, fire protection system, component cooling water system, intake cooling water system, demineralized water system, service water system, and other events that may affect the boron concentration of the pool, such as a seismic event, pipe break, and loss of offsite power.

The largest possible flow rate for dilution of the SFP is 300 gpm due to a pipe rupture of the primary water system. In this case, offsite power must be available for this flow rate to occur. If offsite power is not available, the primary water pumps would not be available, and a dilution would not occur. The primary water system tank has a capacity of 150,000 gallons with automatic makeup from the site water treatment facility. At a flow rate of 300 gpm, over 19 hours would be needed to dilute the SFP from the TS minimum of 1720 ppm to 520 ppm. In this time, two operator rounds, which occur every 8 hours, would have occurred. If the event occurred due to a seismic event, personnel rounds are required within 2 hours following the seismic event. Additionally, alarms such as the SFP level and the primary water tank level would alert operators to this potential dilution event.

The only dilution source large enough to dilute the SFP without replenishment would be the service water tanks, of which St. Lucie has two, each with a capacity of 500,000 gallons. Two-inch service water piping enters the fuel handling building but the lines are not located in the vicinity of the pool. Additionally, the service water tank does not replenish the primary water tank for St. Lucie Unit 2, which has connections to the SFP. Therefore, the service water system is not a viable dilution source for the St. Lucie, Unit 2, SFP.

Based on the evaluation, the licensee determined the most rapid dilution event would occur in approximately 19 hours due to a pipe rupture of the primary water system. All other evaluated dilution events would take longer to reach the minimum boron concentration; therefore, these events would be detected by plant personnel during required rounds every 8 hours. To detect low flow, long term dilution events, the licensee samples its SFP every 7 days. This frequency is consistent with the standard TS for Combustion Engineering and Westinghouse plants and is considered appropriate for this plant.

The licensee also considered an alternate, infrequent SFP configuration that is more limiting than the normal configuration. This configuration occurs with the cask loading area isolated and results in a SFP volume of 250,404 gallons. However, even for a dilution event with the largest flow rate it would take over 16 hours to reach the 520 ppm level. Although less time is needed for the dilution to occur, the conclusion remains the same for the normal pool configuration.

The licensee concluded that an unplanned or inadvertent event that would dilute the SFP boron concentration from 1720 ppm to 520 ppm is not credible for St. Lucie, Unit 2. The staff finds that the combination of the large volume of water required for a dilution event, the TS-controlled SFP concentration and 7-day sampling requirement, and plant personnel rounds would adequately detect a dilution event prior to k_{eff} reaching 0.95 (520 ppm) and, therefore, the analysis and proposed technical specification controls are acceptable for the boron dilution aspects of the request.

Additionally, the criticality analysis for the spent fuel storage pool show that k_{eff} would remain less than 1.0 at a 95/95 probability/confidence level even if the pool were completely filled with unborated water. Therefore, even if the spent fuel storage pool were diluted to zero ppm, the spent fuel is expected to remain subcritical.

3.0 STAFF CONCLUSION

Based on the review described previously, the staff finds the criticality aspects of the proposed St. Lucie, Unit 2, license amendment request are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling.

Following the receipt of the supplement dated November 25, 1998, and the staff's subsequent no significant hazards consideration determination (63 FR 69340), the supplement dated January 28, 1999, contained clarifying information that did not change the no significant hazards consideration determination. An additional notice was required, in accordance with 10 CFR 2.1107, due to an oversight (64 FR 16502, April 5, 1999).

Based on the review of the licensee's cooling analysis and commitment for full-core offload outage-specific calculations described above for the SFP expansion, the staff concludes that the proposed TS changes to increase the SFP storage capacity from 1076 to 1360 fuel assemblies at St. Lucie with respect to the SFP cooling capacities are acceptable.

Based on the review of the boron dilution analysis described above for the soluble boron credit, the staff finds the boron dilution aspects of the proposed St. Lucie, Unit 2, license amendment request acceptable. The TS boron concentration of 1720 ppm and 7-day surveillance requirements are acceptable to ensure that sufficient time is available to detect and mitigate a dilution event prior to exceeding the design basis k_{eff} of 0.95.

4.0 STATE CONSULTATION

By Letter dated March 8, 1991, Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, informed Deborah A. Miller, Licensing Assistant, U.S. NRC, that the State of Florida did not desire notification of issuance of license amendments. Thus, the State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment has been published (64 FR 23133) in the Federal Register on April 29, 1999. Accordingly, the Commission has determined that the issuance of this amendment will not result in any environmental impacts other than those evaluated in the Final Environmental Statement. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 49936, 63 FR 69340, and 64 FR 16502).

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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Date: May 6, 1999

7.0 REFERENCES

1) J. A. Stall, FPL, letter to U. S. Nuclear Regulatory Commission, "St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, SFP Storage Capacity; Soluble Boron Credit," December 31, 1997.

2) R. S. Kundalkar, FPL, letter to U. S. Nuclear Regulatory Commission, "St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment: SFP Storage Capacity; Soluble Boron Credit (TAC No. MA0666), Response to Request for Additional Information," May 15, 1998.

3) Newmyer, W. D., "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Westinghouse Electric Corporation, WCAP-14416-NP-A, November 1996.

4) "St. Lucie Unit 2 Criticality Safety Analysis for the Spent Fuel Storage Rack Using Soluble Boron Credit," ABB Combustion Engineering Nuclear Operations, CENPD-387, October 1997.

5) NRC letter to Mr. T. Greene, Westinghouse Owners Group, dated October 25, 1996, Enclosure: NRC Safety Evaluation of WCAP-14416-P.

6) Westinghouse report WCAP-14416-A, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology."

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