Mr. John Paul Cowan
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing (NA1B)
Crystal River Energy Complex
15760 W. Power Line Street
Crystal River, Florida 34428-6708

September 13, 2000

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING SPENT FUEL POOL STORAGE CAPACITY INCREASE (TAC NO. MA6754)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. DPR-72 for Crystal River Unit 3. This amendment is in response to a Florida Power Corporation (FPC) request dated September 16, 1999, as supplemented by letters dated May 3 and June 29, 2000. FPC proposed changes to the CR-3 Improved Technical Specifications to allow an increase in the spent fuel pool (SFP) storage capacity by replacing fuel racks in the B SFP with new high-density fuel racks.

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

Sincerely,

/RA/
L. Wiens, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures: 1. Amendment No. 193 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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

September 13, 2000

Mr. John Paul Cowan
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UNITED STATES
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FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION,
CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. **193**
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated September 16, 1999, as supplemented on May 3 and June 29, 2000, 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. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 193 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance to be fully implemented no later than September 1, 2001. While the spent fuel pool reracking modification is in progress, both the improved technical specifications issued through Amendment No.192, and those improved technical specifications being amended by Amendment No. 193 will be applicable.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: September 13, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 193

TO FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

Remove Page	Insert Page
3.7-28	3.7-28
3.7-30	3.7-30
3.7-31	3.7-31
3.7-32	3.7.32
3.7-33	3.7-33
3.7-33A	--------
4.0-2	4.0-2
4.0-3	4.0-3
B 3.7-69	B 3.7-69
B 3.7-71	B 3.7-71
B 3.7-72	B 3.7-72
B 3.7-73	B 3.7-73
B 3.7-74	B 3.7-74
B 3.7-75	B 3.7-75
B 3.7-76	B 3.7-76

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO. 3.7.14 The spent fuel pool boron concentration shall be ≥1925 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Verify by administrative means a Storage Pool A and Storage Pool B spent fuel pool verification has been performed since the last movement of fuel assemblies in the spent fuel pool.	Immediately

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Storage Pool A and Storage Pool B, shall be within the acceptable region of Figure 3.7.15-1 or Figure 3.7.15-2.

APPLICABILITY: Whenever any fuel assembly is stored in Storage Pool A or Storage Pool B of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------NOTE-------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to an acceptable configuration.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 or Figure 3.7.15-2.	Prior to storing the fuel assembly in Storage Pool A or Storage Pool B.

MINIMUM BURNUP REQUIRED FOR
"A" POOL STORAGE

NOTE: One-out-of-two checkerboard loading with empty cells allowed for any combination of enrichment and burnup.

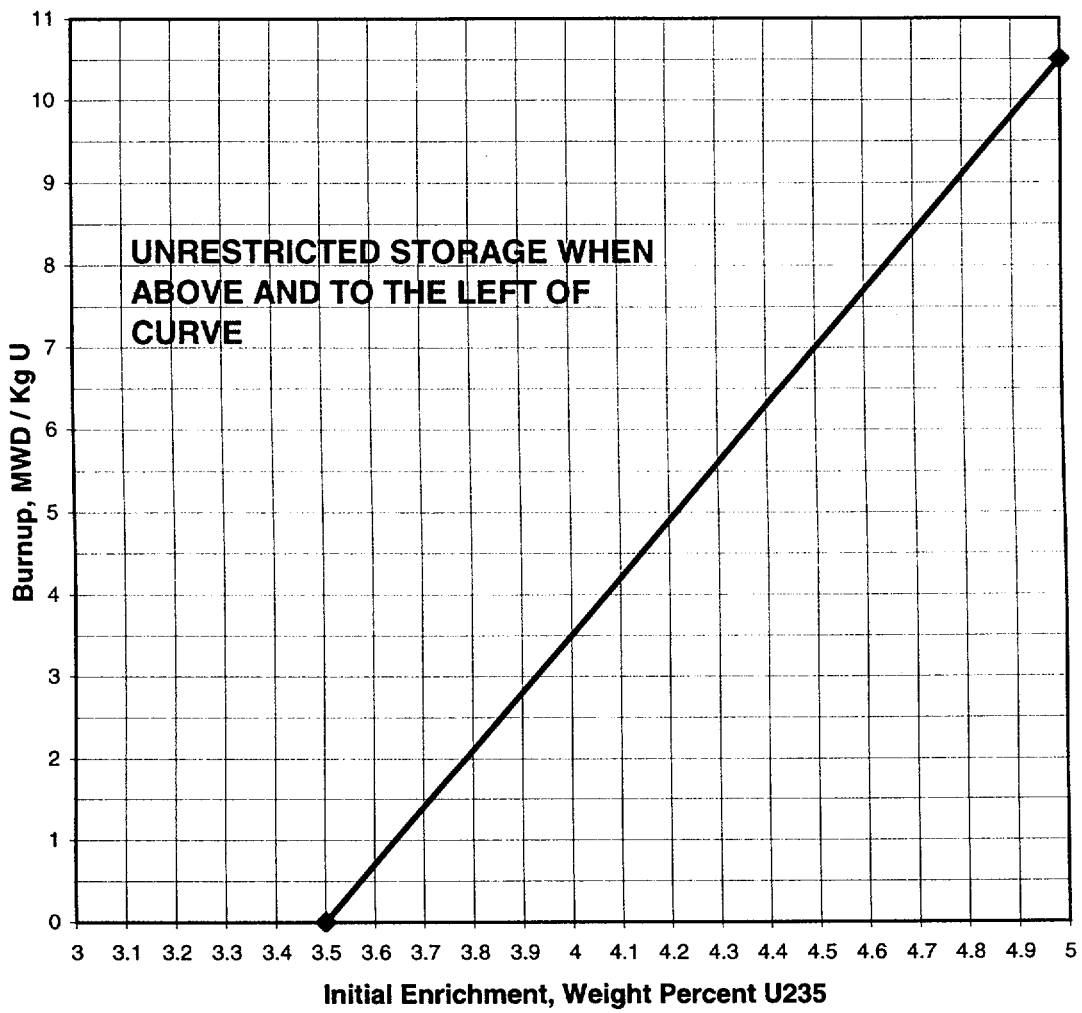

Figure 3.7.15-1
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool A

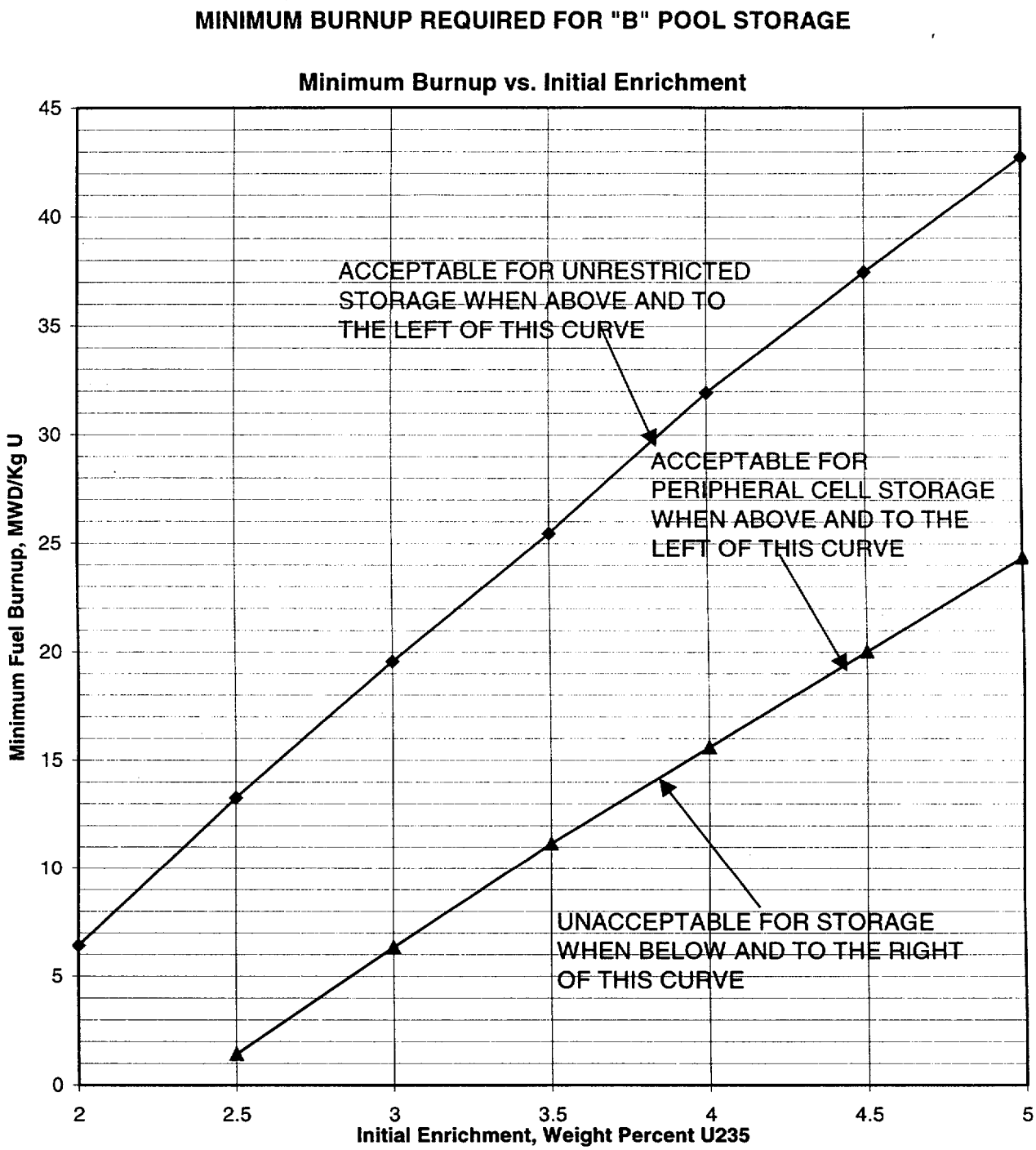

Figure 3.7.15-2
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool B

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.6 of the FSAR;

c. A nominal 9.11 inch center to center distance between fuel assemblies placed in the B pool;

d. A nominal 10.5 inch center to center distance between fuel assemblies placed in the A pool.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ is fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.6 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.6 of the FSAR; and

d. A nominal 21.125 inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

4.0 DESIGN FEATURES

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 138 feet 4 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1474 fuel assemblies and six failed fuel containers.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the Bases for LCO 3.7.15, "Spent Fuel Assembly Storage," fuel assemblies are stored in the high-density region of the spent fuel pool storage racks in accordance with criteria based on initial weight-percent enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations (criticality analysis) are conservatively developed without taking credit for the boron in the pool water.

APPLICABLE SAFETY ANALYSIS

The acceptance criteria for the fuel storage pool criticality analyses is that a k_{eff} of ≤ 0.95 must be maintained for all postulated events. The storage racks are capable of maintaining this k_{eff} with unborated pool water at a temperature yielding the highest reactivity (assuming the storage restrictions of LCO 3.7.15 are met). Most abnormal storage locations will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the mis-loading of an assembly with a burnup and enrichment combination outside the acceptable area in Figure 3.7.15-1 and 3.7.15-2, or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. For such events, credit is taken for the presence of boron in the pool water since the NRC does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in k_{eff}, caused by the boron more than offsets the reactivity addition caused by credible accidents.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The required concentration of dissolved boron in the fuel storage pool of ≥ 1925 ppm preserves the assumption used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

Operating experience has shown significant differences between boron measured near the top of the pool and that measured elsewhere. As long as this SR is met, the analyzed events are fully bounded. The 7 day Frequency is acceptable because no major replenishment of pool water is expected to take place over this period of time.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 With Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI-931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

This document describes the Bases for the Spent Fuel Assembly Storage which imposes storage requirements upon irradiated and unirradiated fuel assemblies stored in the fuel storage pools containing high density racks. The storage areas, which are part of the Spent Fuel System, governed by this Specification are:

a. Fuel storage pool "A" and
b. Fuel storage pool "B".

In general, the function of the storage racks is to support and protect new and spent fuel from the time it is placed in the storage area until it is shipped offsite.

Spent fuel is stored underwater in either fuel storage pool A or B. Only fuel pool A has the capability to store failed fuel in containers. Spent fuel pool A features high density poison storage racks with a 10 1/2 inch center-to-center distance capable of storing 542 assemblies. Fuel pool A is capable of storing fuel with enrichments up to 5.0 weight percent U-235 (Ref. 1) without exceeding the criticality criteria of Reference 3 providing the fuel has sufficient burnup. New fuel will be placed into pool A only.

Spent fuel pool B also contains high density racks having a 9.11 inch center-to-center distance capable of storing 932 assemblies. Fuel pool B is capable of storing fuel with enrichments up to 5.0 weight percent U-235 (Ref. 2) without exceeding the criticality criteria of Reference 3, providing the fuel has sufficient burnup and required storage configuration. New fuel will not be placed into pool B.

It should be noted that the maximum enrichment limits are actually nominal values. The tolerance of fuel supplied by DOE is ± 0.013 weight percent. Thus, it is possible to have fuel with an initial enrichment slightly in excess of the stated limit. This is accounted for in the criticality analysis and is therefore acceptable.

(continued)

BASES

BACKGROUND (continued)

Both of the spent fuel pools are constructed of reinforced concrete and lined with stainless steel plate. They are located in the fuel handling area of the auxiliary building.

New fuel storage requirements are addressed in Section 4.0, "Design Features".

APPLICABLE SAFETY ANALYSES

The function of the spent fuel storage racks are to support and protect spent fuel assemblies from the time they are placed in the pool until they are shipped offsite. The spent fuel assembly storage LCO was derived from the need to establish limiting conditions on fuel storage to assure sufficient safety margin exists to prevent inadvertent criticality. The spent fuel assemblies are stored entirely underwater in a configuration that has been shown to result in a reactivity of less than or equal to 0.95 under worse case conditions (Ref. 1 and 2). The spent fuel assembly enrichment requirements in this LCO are required to ensure inadvertent criticality does not occur in the spent fuel pool.

Inadvertent criticality within the fuel storage area could result in offsite radiation doses exceeding 10 CFR 100 limits.

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

Limits on the new and irradiated fuel assembly storage in high density racks were established to ensure the assumptions of the criticality safety analysis of the spent fuel pools is maintained.

Limits on initial fuel enrichment and burnup for both new and for spent fuel stored in pool A have been established. Two limits are defined:

1. Initial fuel enrichment must be less than or equal to 5.0 weight percent U-235, and

(continued)

BASES

LCO (continued)

2. For new, low irradiation, and spent fuel with initial enrichment less than or equal to 5.0 weight percent and greater than or equal to 3.5 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-1. Figure 3.7.15-1 presents two areas of required fuel assembly burnup as a function of initial enrichment. For fuel with enrichment-burnup combinations in the area above the curve, there are no restrictions on where the fuel can be stored. For fuel with enrichment-burnup combinations below the curve, the fuel must be stored in a one-out-of-two checkerboard configuration with water cells that contain no fuel. The acceptability of storing this fuel in the checkerboard configuration is documented in Reference 6.

Fuel enrichment limits are based on avoiding inadvertent criticality in the spent fuel pool. The CR-3 spent fuel storage system was initially designed to a maximum enrichment of 3.5 weight percent. Enrichments of up to 5.0 weight percent are permissible for storage in spent fuel pool A as long as the fuel burnup is sufficient to limit the worst case reactivity in the storage pool to less than or equal to 0.95. Fuel burnup reduces the reactivity of the fuel due to the accumulation of fission product poisons. Reference 1 documents that the required burnup varies linearly as a function of enrichment with 10500 megawatt days per metric ton uranium (Mwd/mtU) required for fuel with 5.0 weight percent enrichment and 0 burnup required for 3.5 weight percent enriched fuel.

Similar types of restrictions have been established for Pool B.

1. Initial fuel enrichment must be ≤ 5.0 weight percent U-235, and

2. For fuel with initial enrichment ≤ 5.0 weight percent and ≥ 2.0 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-2.

(continued)

BASES

LCO (continued)

Fuel with burnup-enrichment combinations in the area above the upper curve has no restrictions on where it can be stored. Fuel with burnup-enrichment combinations in the area between the lower and upper curves must be stored in the peripheral cells of the pool. The peripheral cells are those that are adjacent to the walls of the spent fuel pool. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in Pool B, but must be stored in Pool A.

The LCO allows compensatory loading techniques, specified in the FSAR and applicable fuel handling procedures, as an alternative to storing fuel assemblies in accordance with Figures 3.7.15-1 and 3.7.15-2. This is acceptable since these loading patterns assure the same degree of subcriticality within the pool.

APPLICABILITY

In general, limiting fuel enrichment of stored fuel prevents inadvertent criticality in the storage pools. Inadvertent criticality is dependent on whether fuel is stored in the pools and is completely independent of plant MODE.

Therefore, this LCO is applicable whenever any fuel assembly is stored in high density fuel storage locations.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating LCO 3.0.3 does not apply. Since the design basis accident of concern in this Specification is an inadvertent criticality, and since the possibility or consequences of this event are independent of plant MODE, there is no reason to shutdown the plant if the LCO or Required Actions cannot be met.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1 or Figure 3.7.15-2, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance. The Immediate Completion Time underscores the necessity of restoring spent fuel pool irradiated fuel loading to within the initial assumptions of the criticality analysis.

(continued)

BASES

ACTIONS

A.1 (continued)

The ACTIONS do not specify a time limit for completing movement of the affected fuel assemblies to their correct location. This is not meant to allow an unnecessary delay in resolution, but is a reflection of the fact that the complexity of the corrective actions is unknown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Verification by administrative means that initial enrichment and burnup of fuel assemblies in accordance with Figure 3.7.15-1 and Figure 3.7.15-2 is required prior to storage of spent fuel in storage pool A or pool B (as applicable). This surveillance ensures that fuel enrichment limits, as specified in the criticality safety analyses (Ref. 1 and 2), are not exceeded. The surveillance Frequency (prior to storage in high density region of the fuel storage pool) is appropriate since the initial fuel enrichment and burnup cannot change after removal from the core.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 with Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI 931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

3. NUREG 0800, Standard Review Plan, Section 9.1.1 and 9.1.2, Rev. 2, July 1981.

4. 10 CFR 100.

5. CR-3 FSAR, Section 9.6.

6. Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5% Enriched Mark B-11 Fuel in Checkerboard Arrangement With Water Holes, S. E. Turner, Holtec Report HI-992285, August 1999.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By application dated September 16, 1999, as supplemented by letters dated May 3 and June 29, 2000, the licensee, Florida Power Corporation (FPC), requested an amendment to its facility Operating License No. DPR-72, for Crystal River Unit 3 (CR-3), in accordance with Title 10, *Code of Federal Regulations* (10 CFR) Section 50.90. The requested changes would allow the removal of the existing fuel storage racks from spent fuel pool (SFP) B and the installation of the new high-density racks in SFP B only. References to Region 1 and Region 2 storage racks in SFP B would be deleted since all SFP B racks would be of the same design after the rack replacement. In addition, the allowable fuel storage combination in SFP A would be changed to allow a checkerboard pattern to optimize overall combined fuel storage.

The May 3 and June 29, 2000, supplements did not affect the original no significant hazards determination, or expand the scope of the request as noticed in the FEDERAL REGISTER on December 8, 1999 (64 FR 68702).

2.0 BACKGROUND

The current licensed CR-3 combined SFP storage capacity is 1357 fuel assemblies. Based on expected spent fuel discharges, the licensee projects that full-core-discharge capability would be exceeded in the year 2013. The CR-3 operating license has an expiration date of December 3, 2016. Thus, the current combined capacity of SFP A and B is not adequate to allow all spent fuel discharged from the reactor to be stored onsite for the remainder of the CR-3 operating license.

The licensee has proposed to replace the racks in SFP B with new high-density racks which will provide an additional 117 storage locations. As a result, the combined SFP storage capacity would increase from the current 1357 to 1474 fuel assemblies. The increased storage capacity would provide adequate spent fuel storage capacity for the remainder of the CR-3 operating license.

The replacement racks use Boral as a neutron absorber material instead of the existing Boraflex. Boraflex degrades under long-term exposure to gamma radiation, resulting in water chemistry and water clarity problems. Replacement of the racks with racks using Boral would

be beneficial in maintaining proper water chemistry, as well as reducing the amount of radioactive waste due to increased water cleanup requirements.

3.0 EVALUATION

3.1 Criticality Considerations

3.1.1 Criteria

CR-3 has two SFPs designated as the "A" and "B" pools, which are physically joined together through a transfer canal. The A SFP has high-density storage rack modules which do not utilize Boraflex. The B SFP has eight high-density racks which are constructed with Boraflex. Fuel storage is divided into two regions within the B pool. Region 1 was designed to accommodate new (fresh) fuel assemblies or fuel which has not experienced sufficient burnup to be stored in Region 2. Region 2 was designed to accommodate less reactive irradiated fuel, determined by burnup calculations. The Region 1 racks have a double layer of Boraflex panels within each cell with a 1-inch water gap between each cell. The Region 2 racks have only a single layer of Boraflex.

Boraflex is known to degrade under the influence of gamma radiation and chemical reaction with free radicals in the pool water. Over the first few years of use, the Boraflex is expected to shrink, typically creating gaps distributed randomly in the axial direction. As the gamma dose increases, the Boraflex panels are expected to slowly begin to deteriorate, losing the neutron absorbing component B_4C. Therefore, the proposed reracking request would replace the existing Boraflex racks in Region 1 and Region 2 of pool B with a new set of racks all of the same design containing Boral.

General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50 requires the prevention of criticality in fuel storage and handling. For the normal spent fuel storage configuration, the pool water contains approximately 2000 parts per million (ppm) of soluble boron. This is usually sufficient to maintain the stored spent fuel assemblies at least 25 percent subcritical. However, to conform to the requirements of GDC 62 and to assure the criticality safety under all conditions, the criterion stated in U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) 9.1.2 must also be satisfied. This criterion states that the maximum reactivity of the storage racks shall not exceed a k_{eff} of 0.95 if fully flooded with unborated water. The maximum calculated reactivity includes a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed 0.95 at the 95/95 probability/confidence level.

3.1.2 Spent Fuel Pool A

The storage rack design in pool A is composed of a B_4C neutron absorber sandwiched between two 0.60-inch thick stainless steel boxes of 8.9375-inch inside dimension. The cells are arranged on a 10.50-inch lattice spacing with a 1.173-inch water gap between the storage cells so that there are actually two B_4C plates between each stored fuel assembly. The plates have a thickness of 0.075 inches and a nominal boron-10 (B-10) loading of 0.015 gm/cm^2.

Pool A is currently licensed for storage of irradiated fuel assemblies which conform to the initial enrichment versus burnup combinations specified in Technical Specification (TS) Figure 3.7.15-1. The figure illustrates that fuel with initial enrichments less than or equal to 3.5 weight percent (w/o) U-235 may be stored in the pool A storage racks independent of burnup. Fuel assemblies with initial enrichments between 3.5 and 5.0 w/o U-235 must meet the initial enrichment versus burnup requirements shown in Figure 3.7.15-1. A new criticality analysis for pool A is presented in Holtec International report HI-992285 (Reference 1) and was performed to evaluate the acceptability of storing fresh (unirradiated) fuel enriched to 5.0 w/o U-235 in a 1-out-of-2 checkerboard configuration.

The reactivity calculations for a 1-out-of-2 checkerboard configuration of fuel in the spent fuel racks in pool A were performed by Holtec International with the NITAWL-KENO-5a Monte Carlo code using the 238-group SCALE cross section library and the Nordheim integral treatment for U-238 resonance shielding effects. This code has been acceptably benchmarked against numerous critical experiments which simulate the CR-3 storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, fuel rod size, fuel assembly spacing, and absorber reactivity worth. The Babcock & Wilcox 15 x 15 Mark B-11 fuel assembly at a maximum enrichment of 5.0 w/o U-235 was used as the reference design in the calculations. Previous calculations for CR-3 by Holtec, and approved by the NRC, have shown that the Mark B-11 fuel is the most reactive in pool A. The calculational bias and uncertainty derived from the benchmarks and the uncertainties due to manufacturing tolerances used in the previous approved Holtec analyses for CR-3 were included, as well as an assumed 15% loss in B-10 of the absorber panel to encompass possible degradation and other uncertainties including particle self-shielding. The uncertainties meet the 95/95 probability/confidence level requirement and are, therefore, acceptable.

Based on the above, the staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the storage racks in pool A with a high degree of confidence.

The resulting 95/95 k_{eff} was 0.8466, well below the regulatory limit of 0.95. Therefore, fresh (unirradiated) CR-3 fuel assemblies enriched to 5.0 w/o U-235 can be safely stored in the pool A storage racks in a 1-out-of-2 checkerboard arrangement (fresh fuel alternating with cells containing only water) and maintain at least a 5% subcriticality margin.

3.1.3 Spent Fuel Pool B

The proposed pool B storage racks would replace the current two-region racks with a new set of racks all of the same design. The racks' modules are composed of individual storage cells made of austenitic stainless steel and utilize a neutron absorbing material, Boral, which is attached to each cell wall. The cells are located on a lattice spacing of 9.11 inches. The Boral absorber panels have a thickness of 0.075 inches and a nominal B-10 loading of 0.0216 gm/cm^2. The Boral panels are 7.5 inches in width.

The analysis of the reactivity effects of fuel storage in the proposed CR-3 pool B spent fuel racks was also performed by Holtec International and the results are presented in HI-992128 (Reference 2). As for pool A, the analysis was performed primarily with the three-dimensional

NITAWL-KENO5a Monte Carlo code package. Verification calculations were made with the MCNP4A Monte Carlo code. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, CASMO4. The methods used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the CR-3 racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include close proximity storage and strong neutron absorbers. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the CR-3 pool B storage racks with a high degree of confidence.

For the nominal storage cell design, the racks were assumed to contain the most reactive fuel authorized to be stored without any control rods or burnable poison. These are the Babcock & Wilcox 15x15 Mark B-10F and Mark B-11 fuel. The moderator was assumed to be pure water at a temperature within the design basis range corresponding to the highest reactivity. No credit was taken for radial neutron leakage or for neutron absorption in minor structural members such as spacer grids. Uncertainties due to tolerances in U-235 enrichment and density, boron loading, Boral panel width, cell box inner diameter or lattice pitch, and stainless steel thickness were accounted for as well as a method bias and uncertainty. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. In addition, an allowance of 5 percent of the reactivity decrement from beginning of life to the burnup of interest was included for uncertainty in depletion calculations for those cases where burnup credit is used. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

Pool B was analyzed for fuel of various initial enrichment versus burnup combinations up to a maximum enrichment of 5.0 w/o U-235. Acceptable storage is shown in TS Figure 3.7.15-2. The upper curve represents the minimum burnup requirements for unrestricted storage in pool B. Fuel assemblies meeting these initial enrichment versus minimum burnup requirements will maintain k_{eff} less than or equal to 0.95 when stored anywhere in pool B. Acceptable storage of fuel of lower burnup in peripheral storage cells was also analyzed. This is shown in the lower curve of TS Figure 3.7.15-2. The peripheral cells are those that are adjacent to the walls of the SFP. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in pool B, but must be stored in pool A. Most of the fuel assemblies currently in storage may be safely stored in the unrestricted area of the storage racks, although a few (six at the present time) must be administratively restricted to peripheral cells.

It is possible to postulate events which could lead to an increase in reactivity. However, since the fuel is normally about 25 percent subcritical when stored with at least 1925 ppm of soluble boron in the pool water, as required by TS 3.7.14, (the actual boron concentration is expected to be at least 2000 ppm at all times), a complete loss of soluble boron represents the largest possible increase in reactivity achievable by a single accident (approximately 20 percent Δk). Any other events which could lead to an increase in reactivity, such as fuel assembly misplacement or pool water temperature changes, would result in significantly lower reactivity increases (i.e., on the order of only a few percent Δk). The complete loss of soluble boron is considered to be a highly unlikely event for the following reasons. The racks and pool structure

are designed to Seismic Class I requirements and therefore, pool liner integrity would be assured even in the event of a Safe Shutdown Earthquake (SSE). In addition, a high-water-level alarm at 6 inches above normal pool level would alert the operators to an increasing volume in the pool. If the alarm does not function, or the operators failed to notice the increase, water would overflow the pool and drain to the auxiliary building sump. Frequent operation of the sump pump would be readily noticed well before any substantial reduction in pool soluble boron concentration occurs. A low-level alarm at 2 feet below normal pool level is also available to alert operators to a loss of pool water. Both the high-level and low-level alarms annunciate in the main control room. The pool level is also monitored and recorded every shift by the main control room staff and increasing or decreasing levels in the pool would be readily detected. The addition of pool makeup water requires manipulation of valves and the only personnel authorized to operate these valves are licensed operators. These operators are aware of the need to keep the SFPs borated and the need to control addition of unborated water. Because of the large volume of water required to completely dilute the pool, plant personnel would have a sufficiently long time period to detect the event and stop the dilution. Periodic sampling of the boron concentration every 7 days, as required by TS 3.7.14, also verifies that the level of soluble boron remains consistent with the approved design and would detect any credible dilution events on a timely basis. Therefore, this represents the bounding reactivity accident since, as recommended in the NRC guidance sent to all power reactor licensees on April 14, 1978 (Reference 3), the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The licensee has shown that a 5 percent subcriticality margin (k_{eff} less than or equal to 0.95) is maintained even with no soluble boron in the pool water, thus meeting the NRC subcriticality criterion as well as GDC 62 requirements.

3.1.4 Summary

Based on the review described above, the staff finds the criticality aspects of the proposed license amendment for CR-3 are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The acceptability of revisions to TS Sections 3.7.14 and 3.7.15 is addressed in section 3.8 below.

3.2 Spent Fuel Pool Cooling

The SFP cooling system (SFPCS) which provides cooling to both SFPs consists of two cooling trains, each equipped with one pump and one heat exchanger. Heat is removed from the SFPCS heat exchangers by the nuclear services closed cycle cooling system. In addition, the decay heat removal (DHR) system, which has a higher heat removal capacity, serves as a back-up system to the SFPCS when the reactor vessel is unloaded.

The SFPCS is designed to maintain the SFP water at or below 160°F[1] with both cooling trains operating in parallel to remove a full core discharge heat load of 29.6 x 10^6 Btu/hr. One DHR[2] train alone is capable of maintaining the SFP water temperature at or below 160°F during a planned or an unplanned full-core discharge.

The increased SFP storage capacity from 1357 fuel assemblies to 1474 fuel assemblies would result in a slight increase of SFP decay heat load for any specific fuel discharge scenario; however, the increase would be negligible. The licensee stated that the limiting SFP heat load is from the combined stored spent fuel assemblies and a full core off-load. The full core off-load accounts for approximately 90 percent of the heat load. The increase of less than 10 percent (approximately 8.6 percent) in stored fuel assemblies is comprised of fuel that has been stored in the SFPs the longest, resulting in less decay heat. Thus, the impact of the increased spent fuel storage capacity on the total heat load is less than 1 percent.

Based on the review of licensee's rational and evaluation, we find that the proposed SFP rerack with less than 10 percent increase in the SFP storage capacity will have an insignificant impact on the total SFP heat load and SFP water temperature.

Furthermore, the licensee stated that plant refueling procedures require that fuel assemblies must be held in the reactor for a minimum period of time (165 hours) after shutdown prior to being discharged to the SFP, and that CR-3 administrative controls assure that a fully capable backup cooling system will be available within an appropriate time frame. While the full core offload is in the pool, the following systems are available to assure that the pool temperature remains below 160°F :

1. One train of DHR (immediately after shutdown)

2. Two trains of SFP cooling (165 hours after reactor shutdown)

3. One spent fuel cooling pump aligned to both heat exchangers (271 hours after reactor shutdown).

Also, CR-3 has a monitor system for monitoring the SFP levels, temperatures and pump status. The monitor system alarms in the control room when the SFP water temperature reaches the temperature setpoint of 140°F, SFP reaches the high (159'-0") or low (156"-6") level setpoint, or if the SFP cooling pump trips. Annunciator response procedures list the probable causes and appropriate corrective actions to be taken when an alarm is received. This will provide additional measures to prevent the SFP water temperature of 160°F from being exceeded.

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[1] CR-3 is a pre-SRP (Standard Review Plan) plant. The structural design of the SFP is based upon a steady-water temperature of 160°F. During a normal (planned) refueling outage, an entire core (full core) is offloaded to allow for maintenance and inspection activities to be performed within the reactor vessel.

[2] Both trains are dedicated to SFP cooling. The second DHR train will be available if required.

3.2.1 Summary

Based on our review of the licensee's rationale and evaluation, we find that the proposed increase of less than 10% of fuel assembly storage capacity in the SFPs at the CR-3 plant does not change the design and operational aspects of the SFPCS and will have an insignificant impact on the calculated peak SFP water temperature. Therefore, we conclude, with respect to the thermal-hydraulic considerations, that the licensee's proposal to rerack the SFP to allow an increase in the spent fuel storage capacity from 1357 to 1474 fuel assemblies is acceptable.

3.3 Materials Compatibility

3.3.1 Structural Materials

The new storage rack arrays proposed for use in the SFP are manufactured by Westinghouse Electric Company, LLC. These freestanding, self-supporting racks are designed to stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. They are made primarily from ASME Type 304 austenitic stainless steel. The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-304L for base plates, support plates, cells and leveling pad, ASME SA564-630 (17-4PH) for the screw of the leveling pad assembly, and ASME Type 308L for weld material.

These materials used in the Westinghouse racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the SFP environment. Therefore, they are acceptable for use in this application.

3.3.2 Neutron Absorber Material

The Westinghouse racks employ Boral as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the SFP environments where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed with spot welding to vent the corrosion gases. The neutron absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

3.3.3 Summary

Based on its evaluation, the staff finds the materials utilized in the fabrication of the spent fuel racks manufactured by Westinghouse Electric Company, LLC are compatible with the SFP environment at CR- 3. The type of degradation exhibited by the racks does not affect their

neutron absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

3.4 Structural Evaluation

3.4.1 Storage Racks

FPC has proposed to install eight new racks in the SFP. The storage racks are seismic Category I equipment and are required to remain functional during and after an SSE. FPC, with its contractor, Westinghouse Electric Company, performed structural analyses of the racks in support of the requested license amendment.

The computer program ANSYS was used in performing the dynamic analysis of the proposed racks to demonstrate their structural adequacy under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are freestanding and self-supporting equipment, and are not anchored or attached to the floor or walls of the SFP. A nonlinear dynamic model consisting of fluid, beam, spring, gap and friction elements, as defined in the program, was used to simulate the three-dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculates nodal forces and displacements as well as detailed stress field in the rack elements from the calculated nodal forces.

Two dynamic model analyses were performed: the 3-D single-rack (SR) analysis and the 3-D whole pool multirack (MR) analysis. In these 3-D model analyses, each rack was considered fully loaded, 50 percent loaded, 25 percent loaded and almost empty, with two different coefficients of friction between the rack and the pool floor (μ=0.2 and 0.8) to identify the worst-case response for rack movement and for rack member stresses and strains.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) was generated from the design response spectra defined in the final safety analysis report (FSAR) (Reference 4). FPC demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density function compatible with the design response spectra as discussed in SRP Section 3.7.1.

Tables 3.3 through 3.6 of the licensee's submittal show the SR and MR analysis results. The results of the SR analysis show that the maximum displacement at top of the rack is about 0.1 inch, thus assuring that there are no rack-to-wall or rack-to-rack impacts under the service, upset and faulted loading conditions (Level A, B and D Service Limits). The analysis results show that the uplift rack movement is very small (less than 0.1 inch) indicating that there are large safety margins against an overturning of the racks. The analysis results demonstrate that the structural integrity and stability of the racks and fuel assemblies are maintained. In addition, the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The results show that all induced stresses under the service, upset and faulted loading conditions (Level A,

B and D Service Limits) are smaller than the corresponding allowable stresses specified in the ASME Code, thus indicating the structural adequacy of the rack design.

In the 3-D MR analysis, ten freestanding racks were considered to investigate the fluid-structure interaction effects between the racks and the pool walls as well as those among the racks. The results of the MR analysis indicate that all calculated stresses are lower than the corresponding allowable stresses of the ASME Code. In addition, the results show that there are no rack-to-wall or rack-to-rack impacts as a result of an SSE; thus, assuring that the structural integrity and stability of the racks are maintained.

FPC also calculated the rack weld stresses at the connections under the dynamic loading conditions. FPC indicated that the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, thus, indicating that the weld connection design of the rack is adequate.

Based on (1) FPC's comprehensive parametric study (e.g., varying coefficients of friction and fuel-loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when compared to the corresponding allowable stresses provided in the ASME Boiler and Pressure Vessel Code, (3) the assurance that there is no rack-to-wall and rack-to-rack impacts, and (4) FPC's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.4.2 Spent Fuel Storage Pool

FPC analyzed the SFP to demonstrate the adequacy of the structures with all storage locations occupied by fuel assemblies. The fully-loaded structures were subjected to the load combinations specified in the CR-3 FSAR (Reference 4).

In their supplemental letter dated May 3, 2000 (Reference 5), the licensee indicated that the induced stresses due to the racks in the SFP are smaller than the ACI 349 corresponding allowable stresses. In view of FPC's stress calculations, the staff concludes that FPC's structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading and SSE loading conditions. Thus, the SFP design is acceptable.

3.4.3 Fuel Handling Accident

Three fuel assembly drop scenarios were considered by FPC: (1) drop of a fuel assembly onto the top of a rack with the assembly in a vertical position, (2) drop of a fuel assembly onto the top of a rack with the assembly in an inclined position, and (3) drop of a fuel assembly through an empty rack cell to the bottom of the rack.

The analysis results of accident case (3) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads, therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of accident drop cases (1) and (2) show that damage will be restricted to a

depth of 6.0 inches below the top of the rack, which is less than the acceptance criterion in the FSAR. The staff reviewed FPC's analysis results and concurs with their findings. This is acceptable based on FPC's structural integrity conclusions supported by the parametric studies.

3.4.4 Summary

Based on the review and evaluation of FPC's submittals, the staff concludes that FPC's structural analysis and design of the spent fuel rack modules and the SFP structure is adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR and applicable provisions of the SRP, and are therefore acceptable.

3.5 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the installation of the replacement spent fuel rack modules in SFP B at CR-3 with respect to occupational radiation exposure. As previously discussed, for this modification the licensee plans to replace the eight fuel racks in SFP B with eight higher-capacity racks using Boral as the neutron absorbing material. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed with a personnel dose of approximately 3 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) principles. Workers performing the SFP reracking operation will be given pre-job briefings to ensure that they are aware of their job responsibilities and precautions associated with the job. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained ALARA. Personnel will wear protective clothing and respiratory protective equipment, if necessary. The licensee will issue thermoluminescent dosimeters (TLDs) and self-reading, dose-rate alarming, dosimeters to all personnel. Additional personnel monitoring equipment (such as extremity TLDs or multiple TLDs) will be issued as required.

No dive operations are currently planned for the actual rack replacement. During the proposed reracking operation, the licensee will replace the eight fuel rack modules in SFP B with eight higher-capacity racks using Boral as the neutron absorbing material (fuel storage capacity will increase by 117 storage locations). These new, higher-capacity fuel racks will provide adequate storage capacity, including full core reserve, until the expiration of the CR-3 current operating license in the year 2016. Two aspects of the new racks that could affect the dose rates outside the pool from the stored fuel are a slight decrease in the distance between the racks and the pool walls and a slight decrease in the distance between fuel assemblies. Based on the combined effects of these two changes, the licensee estimates an increase in dose rates in areas adjacent to the pool by a factor of two or less (i.e., from about 2 mR/hr to 4 mR/hr) during full core offload conditions. During normal operating conditions, the expected dose rates in areas adjacent to the SFP walls will be well below these levels. The licensee will monitor these areas routinely and will post any dose rate increases resulting from increased fuel storage.

After temporarily relocating the fuel in existing Racks 1, 2, 3, and 4 in the south end of SFP B to other locations in the pool, the licensee will remove these racks from the pool one at a time. In order to minimize personnel exposure during the rack replacement, the racks will be hydrolazed underwater prior to removing them from the pool. Next, each rack will be lowered into the decontamination pit adjacent to Pool B for rinsing and then wrapped in plastic sheeting to minimize airborne contamination. The first four new racks will then be lowered into SFP B and the fuel from existing Racks 5, 6, 7, and 8 will then be transferred to the four new racks. Racks 5, 6, 7, and 8 will then be removed and decontaminated in the same manner as were the first four racks. The eight old racks will be stored inside the Auxiliary Building until they are transported to an off-site facility for final disposal. The licensee does not expect significant increases in the dose rate in the areas of the Auxiliary Building during or after the rack replacement. However, the licensee will take corrective actions to reduce the dose rate if such increases do occur. Routine surveys in these areas of the Auxiliary Building are performed as part of the licensee's normal radiological protection function in accordance with existing administrative controls.

Since the licensee will be moving fuel assemblies and removing contaminated spent fuel racks, the possibility exists for the generation of small amounts of crud and debris in SFP B. The licensee will utilize an underwater vacuum to remove debris from the floor of SFP B as needed. The licensee has an existing procedure that specifies the radiological controls for use of the underwater vacuum that minimizes or eliminates the potential for radiological incidents, and prevents uncontrolled or undetected high-radiation levels, loose-surface contamination, and airborne radioactivity.

On the basis of our review of the CR-3 license amendment request, the staff concludes that the proposed increase in spent fuel storage capacity at CR-3 can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds that the projected dose for the project of approximately 3 person-rem is in the range of doses for similar modifications at other plants and is therefore acceptable.

3.5.1 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP Purification System. No significant increase in the change-out frequency of demineralizer resin and filter media is expected due to the storage of additional spent fuel in the SFP. In order to maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP to remove any radioactive crud, sediment, and other debris before the new fuel rack modules are installed. The reracking of SFP B will result in an estimated additional 30 cubic feet of solid radioactive waste materials (i.e., resins, filters, debris). The additional 30 cubic feet of solid waste materials are less than 1 percent of the annual solid waste volume generated at CR-3 in 1998 or 1999 (based on the volumes of solid waste shipped off-site reported in the CR-3 Radioactive Effluent Release Reports for 1998 and 1999). The 30 cubic feet are less than 1 percent of the annual solid waste stream projected to be generated in 2001, the year the rack replacement is to be done. Hence, the storage of additional spent fuel in the SFP will not result in a significant change in the generation of solid radwaste at CR-3.

On the basis of our review of the CR-3 license amendment request, the staff concludes that the proposed increase in spent fuel storage capacity at CR-3 can be performed in a manner that will ensure that the generation of additional solid radioactive wastes is minimized. The staff therefore finds the proposed increase in spent fuel storage capacity at CR-3 to be acceptable.

3.6 Accident Dose Evaluation

Section 14.2.2.3 of the CR-3 FSAR describes the design basis fuel handling accident (FHA). The analysis applies both to an FHA in the reactor building and an FHA in the SFP and assumes the failure of one entire spent fuel assembly (208 spent fuel rods) 72 hours after reactor shutdown, with no credit for holdup or filters in either the reactor building or the fuel handling building. FSAR Table 14-34 gives the offsite dose consequences for the design basis fuel handling accident. The licensee's results meet the acceptance criteria for offsite dose given in SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents" of well within (25 percent of) 10 CFR Part 100 guidelines, which are 75 rem thyroid and 6.25 rem for the whole body dose.

The licensee states that the high-density storage racks only increase the storage capacity and do not change the frequency or method for handling fuel assemblies. The licensee also states that the consequences of an FHA are not increased by the installation and use of high-density racks. The staff agrees that the scenario for the postulated fuel handling accident in the fuel handling building does not change due to the use of high-density storage racks in the SFP. No change is being made to the handling of the spent fuel or the types of fuel stored in the CR-3 SFPs nor to the number of fuel assemblies being moved at any one time. Therefore, the inputs and assumptions for the dose consequences analysis do not change, and the current fuel-handling accident dose analysis in the CR-3 FSAR remains bounding.

3.6.1 Summary

The current CR-3 design basis fuel handling accident dose analysis, as described in the plant's FSAR, remains bounding during and after the installation of high-density spent fuel racks in the CR-3 SFP. The staff finds the licensee's calculated radiological consequences of a fuel-handling accident in the SFP, as shown in FSAR Table 14-34, are well within 10 CFR Part 100 guidelines. Therefore, the staff finds the proposed installation of high-density spent fuel racks at CR-3 to be acceptable with regard to potential offsite radiological consequences of a hypothetical fuel handling accident in the SFP.

3.7 Heavy Loads Handling Evaluation

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either: (1) the potential for a load drop is extremely small, or (2) the potential hazards of load drops do not exceed acceptable limits. The NUREG provides guidelines that are implemented in two phases. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads by providing criteria for establishing safe-load

paths, procedures for load handling operations, training of crane operators, and design, testing, inspection, and maintenance of cranes and lifting devices and analyses of the impact of heavy load drops.

Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased reliability of the load handling system, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drop and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety.

In a letter dated July 13, 1984, the NRC forwarded a safety evaluation (SE) to FPC which approved the licensee's measures for controlling and handling heavy loads. In addition, the licensee's response to NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated May 8, 1996, reaffirmed its commitment to apply the guidelines in NUREG-0612.

The proposed license amendment request to change CR-3 SFP storage capacity from 1357 to 1474 fuel assemblies will involve the replacement of eight spent fuel storage racks with new high-density racks. With regard to the control and handling of the SFP storage racks, considerations are given to the design and operation of the hoisting system, safe-load paths, procedures, training of the crane operator and rack installation crew, and analyses of postulated heavy load-drop accidents over the SFP, and over safety-related equipment.

3.7.1 Hoisting System

The 72-ton auxiliary building overhead crane will be used to lift the racks from the auxiliary building loading bay to a location on the operating floor adjacent to SFP B. There, a 20-ton hoist will be attached to the auxiliary building overhead crane, and a special lifting rig will be attached to the hoist. The 20-ton hoist will be suspended from the auxiliary building overhead crane and used in conjunction with the special lifting rig to lift and move the racks into the SFP.

In the SE dated July 13, 1984, we approved the auxiliary building overhead crane and found that the general load handling system and policy and procedures at CR-3 are consistent with the intent of the guidelines in NUREG-0612. However, the 20-ton hoist and the special lifting rig were obtained specifically to lift the new rack modules, therefore, they are not addressed in the SE.

As stated in the July 13, 1984, SE, the auxiliary building overhead crane was designed and fabricated in 1969 to the industry standard in effect at that time, the "Specification for Electric Overhead Traveling Cranes." On the basis of a detailed comparison of the crane with the requirements of Crane Manufacturers Association of America Specification No. 70, "Specifications for Electric Overhead Traveling Cranes," and ANSI/ASME B30.2-1976, "Safety

Standards for Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)," the auxiliary building crane was determined to, either, fully satisfy or meet the intent of these standards. It was also determined that the inspection, testing, and maintenance of this crane met the intent of ANSI B30.2-1976. In addition, the licensee stated in their June 29, 2000, Request for Additional Information (RAI) response that this crane was inspected and load tested in accordance with ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials."

The licensee states that the spent fuel storage racks will be lifted using the 20-ton hoist and a remotely engaged spent fuel rack lifting rig that is specially designed to lift the spent fuel racks. The hoist complies with ASME/ANSI B30.16, "Overhead Hoists (Underhung)," which requires the maximum design stress be no more than 20 percent of ultimate material strength for load-bearing parts. The heaviest load to be lifted during rack replacement weighs less than 20,000 lbs., consisting of a rack plus the lifting rig. Thus, the hoist provides double the normal stress design factor, including dynamic load factor, for handling heavy loads, as required by NUREG-0612. The hoist is load tested and inspected in accordance with the requirements for special lifting and handling equipment as described in ANSI N14.6.

The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612, Sections 5.1.6(1), 5.1.6(3a), and the requirements in ANSI N14.6 (1978). Accordingly, the lifting rig has twice the normal stress design safety factor, including dynamic load factor, for handling the critical loads. The lifting rig also is redundantly designed with four independently loaded lift rods that engage the rack at four lift points. The lifting rig is load tested to 3 times the heaviest load to be lifted (plus a dynamic load factor) and inspected in accordance with the requirements for special lifting and handling equipment as described in ANSI N14.6.

We believe that the capacity of the 72-ton auxiliary building overhead crane coupled with the capacity of the 20-ton hoist and the special lifting rig will support the weight of the racks and the added rigging loads without risk of a load drop. In addition, the design, inspection and testing of the crane and lifting device will help to assure the reliability of the system to enhance safe handling of the racks and further reduce any potential risk of an accidental rack drop during rack installation.

3.7.2 Load Path

The proposed replacement racks will be lifted to the auxiliary building operating level and moved adjacent to SFP B using the auxiliary building overhead crane. At the operating level, the racks will be moved along the previously NRC-approved safe-load path. The racks will then be lowered into the pool to a minimum height of 6 inches above the pool liner for movement into their permanent location. The 20-ton hoist will be attached to the auxiliary building overhead crane in order to lower the new racks into the pool and position them in their location. The licensee states that the racks and lifting rig will not be carried over any fuel or near fuel in the SFP. The racks and lifting rig will also travel the same safe-load path as for the fuel shipping cask. Since the weight of the cask is greater than the weight of the rack, this is an acceptable load path. Fuel will be shuffled into racks that are not in the safe-load path. Movement of the racks along the SFP floor shall not exceed 6 inches above the floor. During movement across the SFP operating deck, the lift height of the racks will be minimized. When a rack is near

SFP B, such that the rack could fall into the pool if dropped, the height of the rack will be maintained less than 6 inches above the operating deck, which is consistent with the load-drop analysis.

The new installed fuel storage racks will not significantly change the method of handling loads during normal plant operations because the same equipment (i.e., the fuel handling system), methods, and procedures as those used prior to the rerack will be used following the installation of the new replacement spent fuel storage racks.

We find that, based on the above discussion, safe-load paths for movement of the spent fuel storage racks do not involve any movement of the racks over spent fuel and are conducted under administrative controls implemented by the licensee. The rack replacement process will use the previously approved safe-load path for the fuel shipping cask, which weighs more than a storage rack, and therefore the safe-load path for movement of the racks is acceptable.

3.7.3 Analysis of Heavy Load-Drop Accidents

The licensee analyzed postulated load-drop accidents of a spent fuel assembly onto the new fuel storage racks and the drop of a spent fuel storage rack onto the SFP floor. Three orientations for a fuel assembly drop were considered, using a bounding impact weight of 2750 lbs. (includes the weight of the heaviest fuel assembly plus the handling tool) from a maximum lift height of 24 inches above the racks: a vertical drop on top of the racks ("shallow-drop"), drop of an assembly on top of a rack with the assembly in an inclined position, and a vertical drop to the base plate of the racks ("deep drop"). The fuel assembly drop on top of the spent fuel storage racks resulted in deformation of the racks to a maximum depth of 6.0 inches below the top of the rack, but no damage to the fuel. The fuel assembly drop onto the base plate of the racks resulted in no puncture damage to the SFP liner.

A vertical drop of the heaviest rack (one of the current SFP B racks) that weighs 17,715 lbs. from 6 inches above the SFP operating deck to the pool floor was evaluated. The combined weight of this rack and the lifting rig is less than 20,000 lbs., which is the weight used for the analysis. The results indicated that the SFP liner would be punctured and SFP water would leak. The SFP concrete floor slab would be indented to a depth of less than 6 inches, however, no structural damage to the concrete would occur. The licensee stated that any leakage through the SFP would be collected by the leak chase system. As stated in the licensee's response to the RAI dated June 29, 2000, valves in the leak chase system will be maintained closed during the rerack operation, therefore, the closed valves will isolate any SFP leakage and minimize any loss of inventory from the SFP. The drop of a rack onto another rack containing fuel was not evaluated because the licensee will preclude the racks from being moved over racks with fuel.

Furthermore, SFP makeup can be obtained from a number of sources to supplement any leakage from the SFP. As noted in FSAR Section 9.3.2.8, "Makeup Capability," makeup can be provided from the Decay Heat System, the Demineralized Water Supply System and temporary fire hoses.

Therefore, because the structural integrity of the concrete slab in the SFP remains unimpaired after a rack drop, and because the licensee has the capability to isolate an SFP leak and provide makeup to supplement any SFP inventory loss, the staff agrees with the licensee's conclusion that neither catastrophic damage of the SFP structure nor rapid loss of pool water would occur to cause uncovering of fuel stored in SFP B.

3.7.4 Load-Handling Procedures

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee stated that they will implement measures using plant programs and procedures to minimize the potential for load-drop accidents during the reracking operations. The licensee stated that the potential is further minimized by compliance with recognized standards for heavy loads equipment such as ANSI N14.6, and NUREG-0612.

The licensee's conclusion that, based on the load-drop analyses, the integrity of the fuel and the SFP would be maintained if a fuel assembly or a spent fuel storage rack is dropped is acceptable to the staff. The design and capability of the crane and lifting devices in conjunction with the administrative procedures and controls that are focused on, but not limited to, the areas noted above would enable the licensee to help maintain safety during the rerack operation.

3.7.5. Summary

Based on the preceding discussions, we find that the aforementioned considerations for the movement of heavy loads to support the proposed increase in the SFP storage capacity at CR-3 are acceptable. The licensee's use of the cranes, the special lifting rig for the spent fuel racks, and the administrative controls and procedures enhances the licensee's capability to reduce the potential for load drops and are in accordance with NUREG-0612 and ANSI N14.6. The design, testing and inspection of the cranes, the hoist, lifting rig and other lifting devices will assure that the hoisting system is highly reliable and enable the licensee to safely handle the racks and other heavy loads during the rack installation process. The licensee's postulated accident analyses of the spent fuel storage rack indicate that the SFP liner could be breached. However, due to the capabilities of the leak chase and SFP makeup systems, the licensee is capable of maintaining the SFP and its contents within the acceptable consequence limits set forth in NUREG-0612.

3.8 PROPOSED TECHNICAL SPECIFICATION REVISIONS

The licensee proposed changes to the following Technical Specifications (TS) in support of the rack replacement project in SFP B and the change in new fuel storage configuration in SFP A.

In Section 3.7.14, "Spent Fuel Pool Boron Concentration", references to Pool B, Region 2, would be deleted, since all of the racks in SFP B would be of the same design after rack replacement.

In Section 3.7.15, "Spent Fuel Assembly Storage", Figure 3.7.15-3 would be deleted, since this figure specified acceptable storage areas for the existing SFP B Region 2. References to

Figure 3.7.15-3 would be deleted, as well as deleting reference to storage in accordance with the Final Safety Analysis Report, since all acceptable storage configurations would be specified in the TS. Figure 3.7.15-1 would be revised by adding a note that specifies that storage of fuel in a checkerboard pattern with empty cells is acceptable in any region of SFP A. Figure 3.7.15-2 would be revised to indicate the acceptable fuel assembly storage locations in SFP B.

Section 4.3.1, "Criticality", would be changed to reflect the dimensions of the replacement racks, and to delete reference to Pool B, Region 2.

Section 4.3.3, "Capacity", would be changed to reflect the increased combined storage capacity of 1474 fuel assemblies.

The Bases for Sections 3.7.14 and 3.7.15 would be changed to reflect the installation of the new racks, changes to new fuel storage locations, and to update associated references.

The staff has reviewed the proposed changes to the TS. Based on the above evaluations, the staff finds the proposed changes to TS 3.7.14 "Spent Fuel Pool Boron Concentration," 3.7.15 "Spent Fuel Assembly Storage," 4.3.1 "Criticality," and 4.3.3 "Capacity," reflect the results of the criticality analysis. In addition, the changes are appropriate to allow use of the new high-density racks in SFP B and the associated increased spent fuel storage capacity, and, therefore, are acceptable. The staff also finds that the associated Bases changes are consistent with the approved TS changes.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission (NRC), 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 September 12, 2000 (65 FR 55059). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Florida Power Corporation, Crystal River Unit 3, License Amendment Request #239, Revision 0, Attachment E, "Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5 percent Enriched Mark B-11 Fuel in Checkerboard Arrangement with Water Holes," HI-992285, Holtec International, August 1999.

2. Florida Power Corporation, Crystal River Unit 3, License Amendment Request #239, Revision 0, Attachment F, "Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3," HI-992128, Holtec International.

3. Letter, B. K. Grimes (NRC) to All Power Reactor Licensees (with Enclosure), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

4. Crystal River Unit 3, Final Safety Analysis Report.

5. Letter, Florida Power Corporation to U.S. NRC, "Response to NRC Request for Additional Information regarding Enhanced Spent Fuel Storage - License Amendment Request #239 (TAC No. MA6754)," dated May 3, 2000.

Principal Contributors: L. Kopp, C. Lauron, C. Hinson, M. Hart, L. Wiens, D. Shum, Y. Kim, S. Schneider, B. Thomas

Date: September 13, 2000

Mr. John Paul Cowan
Florida Power Corporation

CRYSTAL RIVER UNIT NO. 3

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

September 13, 2000

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SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING SPENT FUEL POOL STORAGE CAPACITY INCREASE (TAC NO. MA6754)

Dear Mr. Cowan:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 193 to Facility Operating License No. DPR-72 for Crystal River Unit 3. This amendment is in response to a Florida Power Corporation (FPC) request dated September 16, 1999, as supplemented by letters dated May 3 and June 29, 2000. FPC proposed changes to the CR-3 Improved Technical Specifications to allow an increase in the spent fuel pool (SFP) storage capacity by replacing fuel racks in the B SFP with new high-density fuel racks.

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

Sincerely,

L. Wiens, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures: 1. Amendment No. 193 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION,
CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 193
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated September 16, 1999, as supplemented on May 3 and June 29, 2000, 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. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 193 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance to be fully implemented no later than September 1, 2001. While the spent fuel pool reracking modification is in progress, both the improved technical specifications issued through Amendment No.192, and those improved technical specifications being amended by Amendment No. 193 will be applicable.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Project Licensing Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the
Technical Specifications

Date of Issuance: September 13, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 193

TO FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

Remove Page	Insert Page
3.7-28	3.7-28
3.7-30	3.7-30
3.7-31	3.7-31
3.7-32	3.7.32
3.7-33	3.7-33
3.7-33A	--------
4.0-2	4.0-2
4.0-3	4.0-3
B 3.7-69	B 3.7-69
B 3.7-71	B 3.7-71
B 3.7-72	B 3.7-72
B 3.7-73	B 3.7-73
B 3.7-74	B 3.7-74
B 3.7-75	B 3.7-75
B 3.7-76	B 3.7-76

3.7 PLANT SYSTEMS

3.7.14 Spent Fuel Pool Boron Concentration

LCO. 3.7.14 The spent fuel pool boron concentration shall be ≥1925 ppm.

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel pool boron concentration not within limit.	----------NOTE---------- LCO 3.0.3 is not applicable. ----------	
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately
	AND	
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately
	OR	
	A.2.2 Verify by administrative means a Storage Pool A and Storage Pool B spent fuel pool verification has been performed since the last movement of fuel assemblies in the spent fuel pool.	Immediately

3.7 PLANT SYSTEMS

3.7.15 Spent Fuel Assembly Storage

LCO 3.7.15 The combination of initial enrichment and burnup of each spent fuel assembly stored in Storage Pool A and Storage Pool B, shall be within the acceptable region of Figure 3.7.15-1 or Figure 3.7.15-2.

APPLICABILITY: Whenever any fuel assembly is stored in Storage Pool A or Storage Pool B of the spent fuel pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ---------NOTE-------- LCO 3.0.3 is not applicable. ---------- Initiate action to move the noncomplying fuel assembly to an acceptable configuration.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.15-1 or Figure 3.7.15-2.	Prior to storing the fuel assembly in Storage Pool A or Storage Pool B.

MINIMUM BURNUP REQUIRED FOR
"A" POOL STORAGE

NOTE: One-out-of-two checkerboard loading with empty cells allowed for any combination of enrichment and burnup.

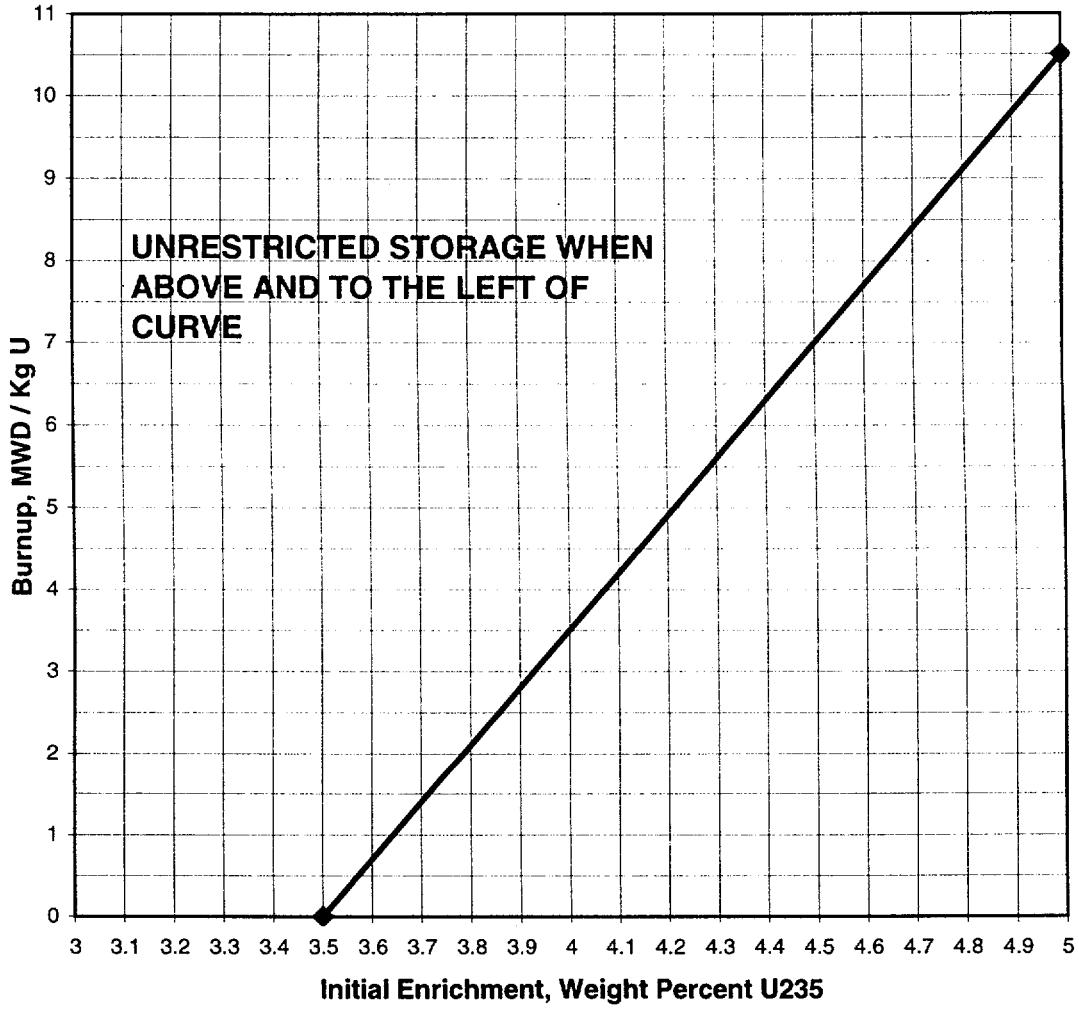

Figure 3.7.15-1
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool A

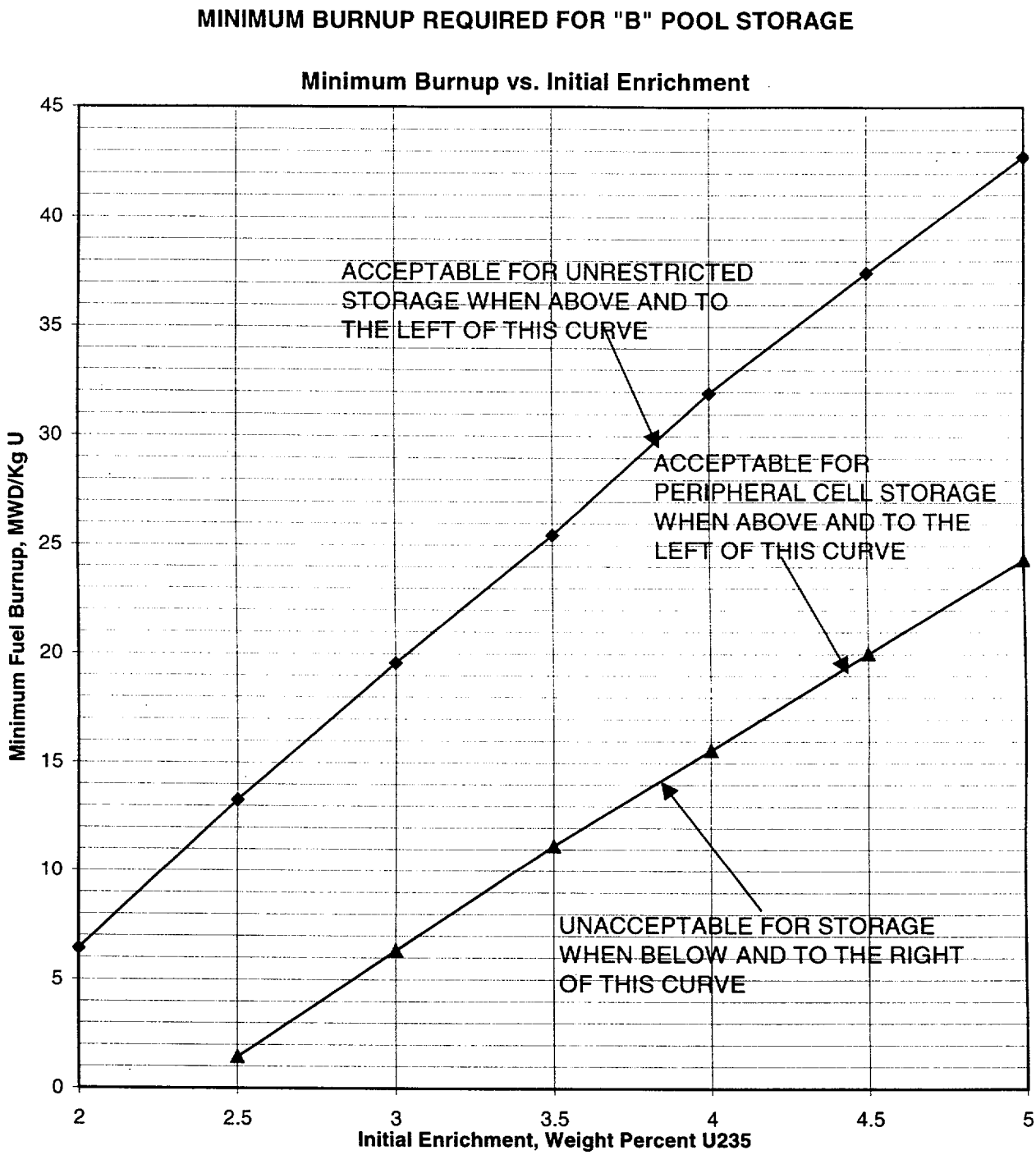

Figure 3.7.15-2
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool B

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.6 of the FSAR;

c. A nominal 9.11 inch center to center distance between fuel assemblies placed in the B pool;

d. A nominal 10.5 inch center to center distance between fuel assemblies placed in the A pool.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. $k_{eff} \leq 0.95$ is fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.6 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 9.6 of the FSAR; and

d. A nominal 21.125 inch center to center distance between fuel assemblies placed in the storage racks.

(continued)

4.0 DESIGN FEATURES

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 138 feet 4 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1474 fuel assemblies and six failed fuel containers.

B 3.7 PLANT SYSTEMS

B 3.7.14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the Bases for LCO 3.7.15, "Spent Fuel Assembly Storage," fuel assemblies are stored in the high-density region of the spent fuel pool storage racks in accordance with criteria based on initial weight-percent enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ 2000 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations (criticality analysis) are conservatively developed without taking credit for the boron in the pool water.

APPLICABLE SAFETY ANALYSIS

The acceptance criteria for the fuel storage pool criticality analyses is that a k_{eff} of ≤ 0.95 must be maintained for all postulated events. The storage racks are capable of maintaining this k_{eff} with unborated pool water at a temperature yielding the highest reactivity (assuming the storage restrictions of LCO 3.7.15 are met). Most abnormal storage locations will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the mis-loading of an assembly with a burnup and enrichment combination outside the acceptable area in Figure 3.7.15-1 and 3.7.15-2, or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. For such events, credit is taken for the presence of boron in the pool water since the NRC does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in k_{eff}, caused by the boron more than offsets the reactivity addition caused by credible accidents.

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The required concentration of dissolved boron in the fuel storage pool of ≥ 1925 ppm preserves the assumption used in the analyses of the potential accident scenarios described above. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the fuel storage pool.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

Operating experience has shown significant differences between boron measured near the top of the pool and that measured elsewhere. As long as this SR is met, the analyzed events are fully bounded. The 7 day Frequency is acceptable because no major replenishment of pool water is expected to take place over this period of time.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 With Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI-931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

This document describes the Bases for the Spent Fuel Assembly Storage which imposes storage requirements upon irradiated and unirradiated fuel assemblies stored in the fuel storage pools containing high density racks. The storage areas, which are part of the Spent Fuel System, governed by this Specification are:

a. Fuel storage pool "A" and
b. Fuel storage pool "B".

In general, the function of the storage racks is to support and protect new and spent fuel from the time it is placed in the storage area until it is shipped offsite.

Spent fuel is stored underwater in either fuel storage pool A or B. Only fuel pool A has the capability to store failed fuel in containers. Spent fuel pool A features high density poison storage racks with a 10 1/2 inch center-to-center distance capable of storing 542 assemblies. Fuel pool A is capable of storing fuel with enrichments up to 5.0 weight percent U-235 (Ref. 1) without exceeding the criticality criteria of Reference 3 providing the fuel has sufficient burnup. New fuel will be placed into pool A only.

Spent fuel pool B also contains high density racks having a 9.11 inch center-to-center distance capable of storing 932 assemblies. Fuel pool B is capable of storing fuel with enrichments up to 5.0 weight percent U-235 (Ref. 2) without exceeding the criticality criteria of Reference 3, providing the fuel has sufficient burnup and required storage configuration. New fuel will not be placed into pool B.

It should be noted that the maximum enrichment limits are actually nominal values. The tolerance of fuel supplied by DOE is ± 0.013 weight percent. Thus, it is possible to have fuel with an initial enrichment slightly in excess of the stated limit. This is accounted for in the criticality analysis and is therefore acceptable.

(continued)

BASES

BACKGROUND (continued)

Both of the spent fuel pools are constructed of reinforced concrete and lined with stainless steel plate. They are located in the fuel handling area of the auxiliary building.

New fuel storage requirements are addressed in Section 4.0, "Design Features".

APPLICABLE SAFETY ANALYSES

The function of the spent fuel storage racks are to support and protect spent fuel assemblies from the time they are placed in the pool until they are shipped offsite. The spent fuel assembly storage LCO was derived from the need to establish limiting conditions on fuel storage to assure sufficient safety margin exists to prevent inadvertent criticality. The spent fuel assemblies are stored entirely underwater in a configuration that has been shown to result in a reactivity of less than or equal to 0.95 under worse case conditions (Ref. 1 and 2). The spent fuel assembly enrichment requirements in this LCO are required to ensure inadvertent criticality does not occur in the spent fuel pool.

Inadvertent criticality within the fuel storage area could result in offsite radiation doses exceeding 10 CFR 100 limits.

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

Limits on the new and irradiated fuel assembly storage in high density racks were established to ensure the assumptions of the criticality safety analysis of the spent fuel pools is maintained.

Limits on initial fuel enrichment and burnup for both new and for spent fuel stored in pool A have been established. Two limits are defined:

1. Initial fuel enrichment must be less than or equal to 5.0 weight percent U-235, and

(continued)

BASES

LCO (continued)

2. For new, low irradiation, and spent fuel with initial enrichment less than or equal to 5.0 weight percent and greater than or equal to 3.5 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-1. Figure 3.7.15-1 presents two areas of required fuel assembly burnup as a function of initial enrichment. For fuel with enrichment-burnup combinations in the area above the curve, there are no restrictions on where the fuel can be stored. For fuel with enrichment-burnup combinations below the curve, the fuel must be stored in a one-out-of-two checkerboard configuration with water cells that contain no fuel. The acceptability of storing this fuel in the checkerboard configuration is documented in Reference 6.

Fuel enrichment limits are based on avoiding inadvertent criticality in the spent fuel pool. The CR-3 spent fuel storage system was initially designed to a maximum enrichment of 3.5 weight percent. Enrichments of up to 5.0 weight percent are permissible for storage in spent fuel pool A as long as the fuel burnup is sufficient to limit the worst case reactivity in the storage pool to less than or equal to 0.95. Fuel burnup reduces the reactivity of the fuel due to the accumulation of fission product poisons. Reference 1 documents that the required burnup varies linearly as a function of enrichment with 10500 megawatt days per metric ton uranium (Mwd/mtU) required for fuel with 5.0 weight percent enrichment and 0 burnup required for 3.5 weight percent enriched fuel.

Similar types of restrictions have been established for Pool B.

1. Initial fuel enrichment must be ≤ 5.0 weight percent U-235, and

2. For fuel with initial enrichment ≤ 5.0 weight percent and ≥ 2.0 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-2.

(continued)

BASES

LCO (continued)

Fuel with burnup-enrichment combinations in the area above the upper curve has no restrictions on where it can be stored. Fuel with burnup-enrichment combinations in the area between the lower and upper curves must be stored in the peripheral cells of the pool. The peripheral cells are those that are adjacent to the walls of the spent fuel pool. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in Pool B, but must be stored in Pool A.

The LCO allows compensatory loading techniques, specified in the FSAR and applicable fuel handling procedures, as an alternative to storing fuel assemblies in accordance with Figures 3.7.15-1 and 3.7.15-2. This is acceptable since these loading patterns assure the same degree of subcriticality within the pool.

APPLICABILITY

In general, limiting fuel enrichment of stored fuel prevents inadvertent criticality in the storage pools. Inadvertent criticality is dependent on whether fuel is stored in the pools and is completely independent of plant MODE.

Therefore, this LCO is applicable whenever any fuel assembly is stored in high density fuel storage locations.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating LCO 3.0.3 does not apply. Since the design basis accident of concern in this Specification is an inadvertent criticality, and since the possibility or consequences of this event are independent of plant MODE, there is no reason to shutdown the plant if the LCO or Required Actions cannot be met.

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1 or Figure 3.7.15-2, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance. The Immediate Completion Time underscores the necessity of restoring spent fuel pool irradiated fuel loading to within the initial assumptions of the criticality analysis.

(continued)

BASES

ACTIONS

A.1 (continued)

The ACTIONS do not specify a time limit for completing movement of the affected fuel assemblies to their correct location. This is not meant to allow an unnecessary delay in resolution, but is a reflection of the fact that the complexity of the corrective actions is unknown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Verification by administrative means that initial enrichment and burnup of fuel assemblies in accordance with Figure 3.7.15-1 and Figure 3.7.15-2 is required prior to storage of spent fuel in storage pool A or pool B (as applicable). This surveillance ensures that fuel enrichment limits, as specified in the criticality safety analyses (Ref. 1 and 2), are not exceeded. The surveillance Frequency (prior to storage in high density region of the fuel storage pool) is appropriate since the initial fuel enrichment and burnup cannot change after removal from the core.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 with Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI 931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

3. NUREG 0800, Standard Review Plan, Section 9.1.1 and 9.1.2, Rev. 2, July 1981.

4. 10 CFR 100.

5. CR-3 FSAR, Section 9.6.

6. Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5% Enriched Mark B-11 Fuel in Checkerboard Arrangement With Water Holes, S. E. Turner, Holtec Report HI-992285, August 1999.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NO. 50-302

1.0 INTRODUCTION

By application dated September 16, 1999, as supplemented by letters dated May 3 and June 29, 2000, the licensee, Florida Power Corporation (FPC), requested an amendment to its facility Operating License No. DPR-72, for Crystal River Unit 3 (CR-3), in accordance with Title 10, *Code of Federal Regulations* (10 CFR) Section 50.90. The requested changes would allow the removal of the existing fuel storage racks from spent fuel pool (SFP) B and the installation of the new high-density racks in SFP B only. References to Region 1 and Region 2 storage racks in SFP B would be deleted since all SFP B racks would be of the same design after the rack replacement. In addition, the allowable fuel storage combination in SFP A would be changed to allow a checkerboard pattern to optimize overall combined fuel storage.

The May 3 and June 29, 2000, supplements did not affect the original no significant hazards determination, or expand the scope of the request as noticed in the FEDERAL REGISTER on December 8, 1999 (64 FR 68702).

2.0 BACKGROUND

The current licensed CR-3 combined SFP storage capacity is 1357 fuel assemblies. Based on expected spent fuel discharges, the licensee projects that full-core-discharge capability would be exceeded in the year 2013. The CR-3 operating license has an expiration date of December 3, 2016. Thus, the current combined capacity of SFP A and B is not adequate to allow all spent fuel discharged from the reactor to be stored onsite for the remainder of the CR-3 operating license.

The licensee has proposed to replace the racks in SFP B with new high-density racks which will provide an additional 117 storage locations. As a result, the combined SFP storage capacity would increase from the current 1357 to 1474 fuel assemblies. The increased storage capacity would provide adequate spent fuel storage capacity for the remainder of the CR-3 operating license.

The replacement racks use Boral as a neutron absorber material instead of the existing Boraflex. Boraflex degrades under long-term exposure to gamma radiation, resulting in water chemistry and water clarity problems. Replacement of the racks with racks using Boral would

be beneficial in maintaining proper water chemistry, as well as reducing the amount of radioactive waste due to increased water cleanup requirements.

3.0 EVALUATION

3.1 Criticality Considerations

3.1.1 Criteria

CR-3 has two SFPs designated as the "A" and "B" pools, which are physically joined together through a transfer canal. The A SFP has high-density storage rack modules which do not utilize Boraflex. The B SFP has eight high-density racks which are constructed with Boraflex. Fuel storage is divided into two regions within the B pool. Region 1 was designed to accommodate new (fresh) fuel assemblies or fuel which has not experienced sufficient burnup to be stored in Region 2. Region 2 was designed to accommodate less reactive irradiated fuel, determined by burnup calculations. The Region 1 racks have a double layer of Boraflex panels within each cell with a 1-inch water gap between each cell. The Region 2 racks have only a single layer of Boraflex.

Boraflex is known to degrade under the influence of gamma radiation and chemical reaction with free radicals in the pool water. Over the first few years of use, the Boraflex is expected to shrink, typically creating gaps distributed randomly in the axial direction. As the gamma dose increases, the Boraflex panels are expected to slowly begin to deteriorate, losing the neutron absorbing component B_4C. Therefore, the proposed reracking request would replace the existing Boraflex racks in Region 1 and Region 2 of pool B with a new set of racks all of the same design containing Boral.

General Design Criterion (GDC) 62 of Appendix A to 10 CFR Part 50 requires the prevention of criticality in fuel storage and handling. For the normal spent fuel storage configuration, the pool water contains approximately 2000 parts per million (ppm) of soluble boron. This is usually sufficient to maintain the stored spent fuel assemblies at least 25 percent subcritical. However, to conform to the requirements of GDC 62 and to assure the criticality safety under all conditions, the criterion stated in U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) 9.1.2 must also be satisfied. This criterion states that the maximum reactivity of the storage racks shall not exceed a k_{eff} of 0.95 if fully flooded with unborated water. The maximum calculated reactivity includes a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed 0.95 at the 95/95 probability/confidence level.

3.1.2 Spent Fuel Pool A

The storage rack design in pool A is composed of a B_4C neutron absorber sandwiched between two 0.60-inch thick stainless steel boxes of 8.9375-inch inside dimension. The cells are arranged on a 10.50-inch lattice spacing with a 1.173-inch water gap between the storage cells so that there are actually two B_4C plates between each stored fuel assembly. The plates have a thickness of 0.075 inches and a nominal boron-10 (B-10) loading of 0.015 gm/cm^2.

Pool A is currently licensed for storage of irradiated fuel assemblies which conform to the initial enrichment versus burnup combinations specified in Technical Specification (TS) Figure 3.7.15-1. The figure illustrates that fuel with initial enrichments less than or equal to 3.5 weight percent (w/o) U-235 may be stored in the pool A storage racks independent of burnup. Fuel assemblies with initial enrichments between 3.5 and 5.0 w/o U-235 must meet the initial enrichment versus burnup requirements shown in Figure 3.7.15-1. A new criticality analysis for pool A is presented in Holtec International report HI-992285 (Reference 1) and was performed to evaluate the acceptability of storing fresh (unirradiated) fuel enriched to 5.0 w/o U-235 in a 1-out-of-2 checkerboard configuration.

The reactivity calculations for a 1-out-of-2 checkerboard configuration of fuel in the spent fuel racks in pool A were performed by Holtec International with the NITAWL-KENO-5a Monte Carlo code using the 238-group SCALE cross section library and the Nordheim integral treatment for U-238 resonance shielding effects. This code has been acceptably benchmarked against numerous critical experiments which simulate the CR-3 storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, fuel rod size, fuel assembly spacing, and absorber reactivity worth. The Babcock & Wilcox 15 x 15 Mark B-11 fuel assembly at a maximum enrichment of 5.0 w/o U-235 was used as the reference design in the calculations. Previous calculations for CR-3 by Holtec, and approved by the NRC, have shown that the Mark B-11 fuel is the most reactive in pool A. The calculational bias and uncertainty derived from the benchmarks and the uncertainties due to manufacturing tolerances used in the previous approved Holtec analyses for CR-3 were included, as well as an assumed 15% loss in B-10 of the absorber panel to encompass possible degradation and other uncertainties including particle self-shielding. The uncertainties meet the 95/95 probability/confidence level requirement and are, therefore, acceptable.

Based on the above, the staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the storage racks in pool A with a high degree of confidence.

The resulting 95/95 k_{eff} was 0.8466, well below the regulatory limit of 0.95. Therefore, fresh (unirradiated) CR-3 fuel assemblies enriched to 5.0 w/o U-235 can be safely stored in the pool A storage racks in a 1-out-of-2 checkerboard arrangement (fresh fuel alternating with cells containing only water) and maintain at least a 5% subcriticality margin.

3.1.3 Spent Fuel Pool B

The proposed pool B storage racks would replace the current two-region racks with a new set of racks all of the same design. The racks' modules are composed of individual storage cells made of austenitic stainless steel and utilize a neutron absorbing material, Boral, which is attached to each cell wall. The cells are located on a lattice spacing of 9.11 inches. The Boral absorber panels have a thickness of 0.075 inches and a nominal B-10 loading of 0.0216 gm/cm^2. The Boral panels are 7.5 inches in width.

The analysis of the reactivity effects of fuel storage in the proposed CR-3 pool B spent fuel racks was also performed by Holtec International and the results are presented in HI-992128 (Reference 2). As for pool A, the analysis was performed primarily with the three-dimensional

NITAWL-KENO5a Monte Carlo code package. Verification calculations were made with the MCNP4A Monte Carlo code. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, CASMO4. The methods used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the CR-3 racks are designed and have been found to adequately reproduce the critical values. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include close proximity storage and strong neutron absorbers. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the CR-3 pool B storage racks with a high degree of confidence.

For the nominal storage cell design, the racks were assumed to contain the most reactive fuel authorized to be stored without any control rods or burnable poison. These are the Babcock & Wilcox 15x15 Mark B-10F and Mark B-11 fuel. The moderator was assumed to be pure water at a temperature within the design basis range corresponding to the highest reactivity. No credit was taken for radial neutron leakage or for neutron absorption in minor structural members such as spacer grids. Uncertainties due to tolerances in U-235 enrichment and density, boron loading, Boral panel width, cell box inner diameter or lattice pitch, and stainless steel thickness were accounted for as well as a method bias and uncertainty. These uncertainties were appropriately determined at least at the 95/95 probability/confidence level. In addition, an allowance of 5 percent of the reactivity decrement from beginning of life to the burnup of interest was included for uncertainty in depletion calculations for those cases where burnup credit is used. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

Pool B was analyzed for fuel of various initial enrichment versus burnup combinations up to a maximum enrichment of 5.0 w/o U-235. Acceptable storage is shown in TS Figure 3.7.15-2. The upper curve represents the minimum burnup requirements for unrestricted storage in pool B. Fuel assemblies meeting these initial enrichment versus minimum burnup requirements will maintain k_{eff} less than or equal to 0.95 when stored anywhere in pool B. Acceptable storage of fuel of lower burnup in peripheral storage cells was also analyzed. This is shown in the lower curve of TS Figure 3.7.15-2. The peripheral cells are those that are adjacent to the walls of the SFP. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in pool B, but must be stored in pool A. Most of the fuel assemblies currently in storage may be safely stored in the unrestricted area of the storage racks, although a few (six at the present time) must be administratively restricted to peripheral cells.

It is possible to postulate events which could lead to an increase in reactivity. However, since the fuel is normally about 25 percent subcritical when stored with at least 1925 ppm of soluble boron in the pool water, as required by TS 3.7.14, (the actual boron concentration is expected to be at least 2000 ppm at all times), a complete loss of soluble boron represents the largest possible increase in reactivity achievable by a single accident (approximately 20 percent Δk). Any other events which could lead to an increase in reactivity, such as fuel assembly misplacement or pool water temperature changes, would result in significantly lower reactivity increases (i.e., on the order of only a few percent Δk). The complete loss of soluble boron is considered to be a highly unlikely event for the following reasons. The racks and pool structure

are designed to Seismic Class I requirements and therefore, pool liner integrity would be assured even in the event of a Safe Shutdown Earthquake (SSE). In addition, a high-water-level alarm at 6 inches above normal pool level would alert the operators to an increasing volume in the pool. If the alarm does not function, or the operators failed to notice the increase, water would overflow the pool and drain to the auxiliary building sump. Frequent operation of the sump pump would be readily noticed well before any substantial reduction in pool soluble boron concentration occurs. A low-level alarm at 2 feet below normal pool level is also available to alert operators to a loss of pool water. Both the high-level and low-level alarms annunciate in the main control room. The pool level is also monitored and recorded every shift by the main control room staff and increasing or decreasing levels in the pool would be readily detected. The addition of pool makeup water requires manipulation of valves and the only personnel authorized to operate these valves are licensed operators. These operators are aware of the need to keep the SFPs borated and the need to control addition of unborated water. Because of the large volume of water required to completely dilute the pool, plant personnel would have a sufficiently long time period to detect the event and stop the dilution. Periodic sampling of the boron concentration every 7 days, as required by TS 3.7.14, also verifies that the level of soluble boron remains consistent with the approved design and would detect any credible dilution events on a timely basis. Therefore, this represents the bounding reactivity accident since, as recommended in the NRC guidance sent to all power reactor licensees on April 14, 1978 (Reference 3), the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The licensee has shown that a 5 percent subcriticality margin (k_{eff} less than or equal to 0.95) is maintained even with no soluble boron in the pool water, thus meeting the NRC subcriticality criterion as well as GDC 62 requirements.

3.1.4 Summary

Based on the review described above, the staff finds the criticality aspects of the proposed license amendment for CR-3 are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The acceptability of revisions to TS Sections 3.7.14 and 3.7.15 is addressed in section 3.8 below.

3.2 Spent Fuel Pool Cooling

The SFP cooling system (SFPCS) which provides cooling to both SFPs consists of two cooling trains, each equipped with one pump and one heat exchanger. Heat is removed from the SFPCS heat exchangers by the nuclear services closed cycle cooling system. In addition, the decay heat removal (DHR) system, which has a higher heat removal capacity, serves as a back-up system to the SFPCS when the reactor vessel is unloaded.

The SFPCS is designed to maintain the SFP water at or below 160°F[1] with both cooling trains operating in parallel to remove a full core discharge heat load of 29.6 x 10^6 Btu/hr. One DHR[2] train alone is capable of maintaining the SFP water temperature at or below 160°F during a planned or an unplanned full-core discharge.

The increased SFP storage capacity from 1357 fuel assemblies to 1474 fuel assemblies would result in a slight increase of SFP decay heat load for any specific fuel discharge scenario; however, the increase would be negligible. The licensee stated that the limiting SFP heat load is from the combined stored spent fuel assemblies and a full core off-load. The full core off-load accounts for approximately 90 percent of the heat load. The increase of less than 10 percent (approximately 8.6 percent) in stored fuel assemblies is comprised of fuel that has been stored in the SFPs the longest, resulting in less decay heat. Thus, the impact of the increased spent fuel storage capacity on the total heat load is less than 1 percent.

Based on the review of licensee's rational and evaluation, we find that the proposed SFP rerack with less than 10 percent increase in the SFP storage capacity will have an insignificant impact on the total SFP heat load and SFP water temperature.

Furthermore, the licensee stated that plant refueling procedures require that fuel assemblies must be held in the reactor for a minimum period of time (165 hours) after shutdown prior to being discharged to the SFP, and that CR-3 administrative controls assure that a fully capable backup cooling system will be available within an appropriate time frame. While the full core offload is in the pool, the following systems are available to assure that the pool temperature remains below 160°F :

1. One train of DHR (immediately after shutdown)

2. Two trains of SFP cooling (165 hours after reactor shutdown)

3. One spent fuel cooling pump aligned to both heat exchangers (271 hours after reactor shutdown).

Also, CR-3 has a monitor system for monitoring the SFP levels, temperatures and pump status. The monitor system alarms in the control room when the SFP water temperature reaches the temperature setpoint of 140°F, SFP reaches the high (159'-0") or low (156"-6") level setpoint, or if the SFP cooling pump trips. Annunciator response procedures list the probable causes and appropriate corrective actions to be taken when an alarm is received. This will provide additional measures to prevent the SFP water temperature of 160°F from being exceeded.

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[1] CR-3 is a pre-SRP (Standard Review Plan) plant. The structural design of the SFP is based upon a steady-water temperature of 160°F. During a normal (planned) refueling outage, an entire core (full core) is offloaded to allow for maintenance and inspection activities to be performed within the reactor vessel.

[2] Both trains are dedicated to SFP cooling. The second DHR train will be available if required.

3.2.1 Summary

Based on our review of the licensee's rationale and evaluation, we find that the proposed increase of less than 10% of fuel assembly storage capacity in the SFPs at the CR-3 plant does not change the design and operational aspects of the SFPCS and will have an insignificant impact on the calculated peak SFP water temperature. Therefore, we conclude, with respect to the thermal-hydraulic considerations, that the licensee's proposal to rerack the SFP to allow an increase in the spent fuel storage capacity from 1357 to 1474 fuel assemblies is acceptable.

3.3 Materials Compatibility

3.3.1 Structural Materials

The new storage rack arrays proposed for use in the SFP are manufactured by Westinghouse Electric Company, LLC. These freestanding, self-supporting racks are designed to stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. They are made primarily from ASME Type 304 austenitic stainless steel. The structural materials used in the fabrication of the new spent fuel racks include: ASME SA240-304L for base plates, support plates, cells and leveling pad, ASME SA564-630 (17-4PH) for the screw of the leveling pad assembly, and ASME Type 308L for weld material.

These materials used in the Westinghouse racks have a history of in-pool usage. They are compatible with the spent fuel assemblies and the SFP environment. Therefore, they are acceptable for use in this application.

3.3.2 Neutron Absorber Material

The Westinghouse racks employ Boral as the neutron absorber material. Boral is a hot-rolled cermet of aluminum and boron carbide, clad in 1100 alloy aluminum. It is chemically inert and has a long history of applications in the SFP environments where it has maintained its neutron attenuation capability under thermal loads. A strongly adhering film of impervious hydrated aluminum oxide passivates the surface of the aluminum typically within a few days of being placed in water. The corrosion layer only penetrates the surface of the aluminum cladding a few microns during passivation and causes no net loss of aluminum cladding. Hydrogen, a product of the corrosion process, may cause swelling in the rack panels resulting in deformation of the storage cells. To prevent this from occurring, the racks are designed with spot welding to vent the corrosion gases. The neutron absorbing capability of Boral is not affected by this corrosion process. Based on these characteristics, the staff finds the use of Boral in this application acceptable.

3.3.3 Summary

Based on its evaluation, the staff finds the materials utilized in the fabrication of the spent fuel racks manufactured by Westinghouse Electric Company, LLC are compatible with the SFP environment at CR- 3. The type of degradation exhibited by the racks does not affect their

neutron absorbing capability. The staff concludes, therefore, that the materials used in the new spent fuel racks are acceptable.

3.4 Structural Evaluation

3.4.1 Storage Racks

FPC has proposed to install eight new racks in the SFP. The storage racks are seismic Category I equipment and are required to remain functional during and after an SSE. FPC, with its contractor, Westinghouse Electric Company, performed structural analyses of the racks in support of the requested license amendment.

The computer program ANSYS was used in performing the dynamic analysis of the proposed racks to demonstrate their structural adequacy under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are freestanding and self-supporting equipment, and are not anchored or attached to the floor or walls of the SFP. A nonlinear dynamic model consisting of fluid, beam, spring, gap and friction elements, as defined in the program, was used to simulate the three-dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies including frictional and hydrodynamic effects. The program calculates nodal forces and displacements as well as detailed stress field in the rack elements from the calculated nodal forces.

Two dynamic model analyses were performed: the 3-D single-rack (SR) analysis and the 3-D whole pool multirack (MR) analysis. In these 3-D model analyses, each rack was considered fully loaded, 50 percent loaded, 25 percent loaded and almost empty, with two different coefficients of friction between the rack and the pool floor (μ=0.2 and 0.8) to identify the worst-case response for rack movement and for rack member stresses and strains.

The seismic analyses were performed utilizing the direct integration time-history method. One set of three artificial time histories (two horizontal and one vertical acceleration components) was generated from the design response spectra defined in the final safety analysis report (FSAR) (Reference 4). FPC demonstrated the adequacy of the single artificial time history set used for the seismic analyses by satisfying requirements of both enveloping design response spectra as well as matching a target power spectral density function compatible with the design response spectra as discussed in SRP Section 3.7.1.

Tables 3.3 through 3.6 of the licensee's submittal show the SR and MR analysis results. The results of the SR analysis show that the maximum displacement at top of the rack is about 0.1 inch, thus assuring that there are no rack-to-wall or rack-to-rack impacts under the service, upset and faulted loading conditions (Level A, B and D Service Limits). The analysis results show that the uplift rack movement is very small (less than 0.1 inch) indicating that there are large safety margins against an overturning of the racks. The analysis results demonstrate that the structural integrity and stability of the racks and fuel assemblies are maintained. In addition, the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. The results show that all induced stresses under the service, upset and faulted loading conditions (Level A,

B and D Service Limits) are smaller than the corresponding allowable stresses specified in the ASME Code, thus indicating the structural adequacy of the rack design.

In the 3-D MR analysis, ten freestanding racks were considered to investigate the fluid-structure interaction effects between the racks and the pool walls as well as those among the racks. The results of the MR analysis indicate that all calculated stresses are lower than the corresponding allowable stresses of the ASME Code. In addition, the results show that there are no rack-to-wall or rack-to-rack impacts as a result of an SSE; thus, assuring that the structural integrity and stability of the racks are maintained.

FPC also calculated the rack weld stresses at the connections under the dynamic loading conditions. FPC indicated that the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, thus, indicating that the weld connection design of the rack is adequate.

Based on (1) FPC's comprehensive parametric study (e.g., varying coefficients of friction and fuel-loading conditions of the rack), (2) the adequate factor of safety of the induced stresses in the rack when compared to the corresponding allowable stresses provided in the ASME Boiler and Pressure Vessel Code, (3) the assurance that there is no rack-to-wall and rack-to-rack impacts, and (4) FPC's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.4.2 Spent Fuel Storage Pool

FPC analyzed the SFP to demonstrate the adequacy of the structures with all storage locations occupied by fuel assemblies. The fully-loaded structures were subjected to the load combinations specified in the CR-3 FSAR (Reference 4).

In their supplemental letter dated May 3, 2000 (Reference 5), the licensee indicated that the induced stresses due to the racks in the SFP are smaller than the ACI 349 corresponding allowable stresses. In view of FPC's stress calculations, the staff concludes that FPC's structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading and SSE loading conditions. Thus, the SFP design is acceptable.

3.4.3 Fuel Handling Accident

Three fuel assembly drop scenarios were considered by FPC: (1) drop of a fuel assembly onto the top of a rack with the assembly in a vertical position, (2) drop of a fuel assembly onto the top of a rack with the assembly in an inclined position, and (3) drop of a fuel assembly through an empty rack cell to the bottom of the rack.

The analysis results of accident case (3) show that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads, therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis results of accident drop cases (1) and (2) show that damage will be restricted to a

depth of 6.0 inches below the top of the rack, which is less than the acceptance criterion in the FSAR. The staff reviewed FPC's analysis results and concurs with their findings. This is acceptable based on FPC's structural integrity conclusions supported by the parametric studies.

3.4.4 Summary

Based on the review and evaluation of FPC's submittals, the staff concludes that FPC's structural analysis and design of the spent fuel rack modules and the SFP structure is adequate to withstand the effects of the applicable loads including that of the SSE. The analysis and design are in compliance with current licensing basis set forth in the FSAR and applicable provisions of the SRP, and are therefore acceptable.

3.5 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the installation of the replacement spent fuel rack modules in SFP B at CR-3 with respect to occupational radiation exposure. As previously discussed, for this modification the licensee plans to replace the eight fuel racks in SFP B with eight higher-capacity racks using Boral as the neutron absorbing material. A number of facilities have performed similar operations in the past. On the basis of the lessons learned from these operations, the licensee estimates that the proposed fuel rack installation can be performed with a personnel dose of approximately 3 person-rem.

All of the operations involved in the fuel rack installation will utilize detailed procedures prepared with full consideration of ALARA (as low as is reasonably achievable) principles. Workers performing the SFP reracking operation will be given pre-job briefings to ensure that they are aware of their job responsibilities and precautions associated with the job. The licensee will monitor and control work, personnel traffic, and equipment movement in the SFP area to minimize contamination and to assure that exposures are maintained ALARA. Personnel will wear protective clothing and respiratory protective equipment, if necessary. The licensee will issue thermoluminescent dosimeters (TLDs) and self-reading, dose-rate alarming, dosimeters to all personnel. Additional personnel monitoring equipment (such as extremity TLDs or multiple TLDs) will be issued as required.

No dive operations are currently planned for the actual rack replacement. During the proposed reracking operation, the licensee will replace the eight fuel rack modules in SFP B with eight higher-capacity racks using Boral as the neutron absorbing material (fuel storage capacity will increase by 117 storage locations). These new, higher-capacity fuel racks will provide adequate storage capacity, including full core reserve, until the expiration of the CR-3 current operating license in the year 2016. Two aspects of the new racks that could affect the dose rates outside the pool from the stored fuel are a slight decrease in the distance between the racks and the pool walls and a slight decrease in the distance between fuel assemblies. Based on the combined effects of these two changes, the licensee estimates an increase in dose rates in areas adjacent to the pool by a factor of two or less (i.e., from about 2 mR/hr to 4 mR/hr) during full core offload conditions. During normal operating conditions, the expected dose rates in areas adjacent to the SFP walls will be well below these levels. The licensee will monitor these areas routinely and will post any dose rate increases resulting from increased fuel storage.

After temporarily relocating the fuel in existing Racks 1, 2, 3, and 4 in the south end of SFP B to other locations in the pool, the licensee will remove these racks from the pool one at a time. In order to minimize personnel exposure during the rack replacement, the racks will be hydrolazed underwater prior to removing them from the pool. Next, each rack will be lowered into the decontamination pit adjacent to Pool B for rinsing and then wrapped in plastic sheeting to minimize airborne contamination. The first four new racks will then be lowered into SFP B and the fuel from existing Racks 5, 6, 7, and 8 will then be transferred to the four new racks. Racks 5, 6, 7, and 8 will then be removed and decontaminated in the same manner as were the first four racks. The eight old racks will be stored inside the Auxiliary Building until they are transported to an off-site facility for final disposal. The licensee does not expect significant increases in the dose rate in the areas of the Auxiliary Building during or after the rack replacement. However, the licensee will take corrective actions to reduce the dose rate if such increases do occur. Routine surveys in these areas of the Auxiliary Building are performed as part of the licensee's normal radiological protection function in accordance with existing administrative controls.

Since the licensee will be moving fuel assemblies and removing contaminated spent fuel racks, the possibility exists for the generation of small amounts of crud and debris in SFP B. The licensee will utilize an underwater vacuum to remove debris from the floor of SFP B as needed. The licensee has an existing procedure that specifies the radiological controls for use of the underwater vacuum that minimizes or eliminates the potential for radiological incidents, and prevents uncontrolled or undetected high-radiation levels, loose-surface contamination, and airborne radioactivity.

On the basis of our review of the CR-3 license amendment request, the staff concludes that the proposed increase in spent fuel storage capacity at CR-3 can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds that the projected dose for the project of approximately 3 person-rem is in the range of doses for similar modifications at other plants and is therefore acceptable.

3.5.1 Solid Radioactive Waste

Spent resins are generated by the processing of SFP water through the SFP Purification System. No significant increase in the change-out frequency of demineralizer resin and filter media is expected due to the storage of additional spent fuel in the SFP. In order to maintain the SFP water as clean as possible, and thereby minimize the generation of spent resins, the licensee will vacuum the floor of the SFP to remove any radioactive crud, sediment, and other debris before the new fuel rack modules are installed. The reracking of SFP B will result in an estimated additional 30 cubic feet of solid radioactive waste materials (i.e., resins, filters, debris). The additional 30 cubic feet of solid waste materials are less than 1 percent of the annual solid waste volume generated at CR-3 in 1998 or 1999 (based on the volumes of solid waste shipped off-site reported in the CR-3 Radioactive Effluent Release Reports for 1998 and 1999). The 30 cubic feet are less than 1 percent of the annual solid waste stream projected to be generated in 2001, the year the rack replacement is to be done. Hence, the storage of additional spent fuel in the SFP will not result in a significant change in the generation of solid radwaste at CR-3.

On the basis of our review of the CR-3 license amendment request, the staff concludes that the proposed increase in spent fuel storage capacity at CR-3 can be performed in a manner that will ensure that the generation of additional solid radioactive wastes is minimized. The staff therefore finds the proposed increase in spent fuel storage capacity at CR-3 to be acceptable.

3.6 Accident Dose Evaluation

Section 14.2.2.3 of the CR-3 FSAR describes the design basis fuel handling accident (FHA). The analysis applies both to an FHA in the reactor building and an FHA in the SFP and assumes the failure of one entire spent fuel assembly (208 spent fuel rods) 72 hours after reactor shutdown, with no credit for holdup or filters in either the reactor building or the fuel handling building. FSAR Table 14-34 gives the offsite dose consequences for the design basis fuel handling accident. The licensee's results meet the acceptance criteria for offsite dose given in SRP Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents" of well within (25 percent of) 10 CFR Part 100 guidelines, which are 75 rem thyroid and 6.25 rem for the whole body dose.

The licensee states that the high-density storage racks only increase the storage capacity and do not change the frequency or method for handling fuel assemblies. The licensee also states that the consequences of an FHA are not increased by the installation and use of high-density racks. The staff agrees that the scenario for the postulated fuel handling accident in the fuel handling building does not change due to the use of high-density storage racks in the SFP. No change is being made to the handling of the spent fuel or the types of fuel stored in the CR-3 SFPs nor to the number of fuel assemblies being moved at any one time. Therefore, the inputs and assumptions for the dose consequences analysis do not change, and the current fuel-handling accident dose analysis in the CR-3 FSAR remains bounding.

3.6.1 Summary

The current CR-3 design basis fuel handling accident dose analysis, as described in the plant's FSAR, remains bounding during and after the installation of high-density spent fuel racks in the CR-3 SFP. The staff finds the licensee's calculated radiological consequences of a fuel-handling accident in the SFP, as shown in FSAR Table 14-34, are well within 10 CFR Part 100 guidelines. Therefore, the staff finds the proposed installation of high-density spent fuel racks at CR-3 to be acceptable with regard to potential offsite radiological consequences of a hypothetical fuel handling accident in the SFP.

3.7 Heavy Loads Handling Evaluation

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated July 1980, provides regulatory guidelines for licensees to assure safe handling of heavy loads in areas where a load drop could impact on stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. The objectives of the guidelines are to assure that either: (1) the potential for a load drop is extremely small, or (2) the potential hazards of load drops do not exceed acceptable limits. The NUREG provides guidelines that are implemented in two phases. Phase I guidelines address measures for reducing the likelihood of dropping heavy loads by providing criteria for establishing safe-load

paths, procedures for load handling operations, training of crane operators, and design, testing, inspection, and maintenance of cranes and lifting devices and analyses of the impact of heavy load drops.

Phase II guidelines address alternatives for mitigating the consequences of heavy load drops, including using either (1) a single-failure-proof crane for increased reliability of the load handling system, or (2) electrical interlocks and mechanical stops for restricting crane travel, or (3) load drop and consequence analyses for assessing the impact of dropped loads on plant safety and operations.

Generic Letter (GL) 85-11, "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants, NUREG-0612," dated June 28, 1985, dismissed the need for licensees to implement the requirements of NUREG-0612, Phase II. However, GL 85-11 encouraged licensees to implement actions they perceive to be appropriate to provide adequate safety.

In a letter dated July 13, 1984, the NRC forwarded a safety evaluation (SE) to FPC which approved the licensee's measures for controlling and handling heavy loads. In addition, the licensee's response to NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," dated May 8, 1996, reaffirmed its commitment to apply the guidelines in NUREG-0612.

The proposed license amendment request to change CR-3 SFP storage capacity from 1357 to 1474 fuel assemblies will involve the replacement of eight spent fuel storage racks with new high-density racks. With regard to the control and handling of the SFP storage racks, considerations are given to the design and operation of the hoisting system, safe-load paths, procedures, training of the crane operator and rack installation crew, and analyses of postulated heavy load-drop accidents over the SFP, and over safety-related equipment.

3.7.1 Hoisting System

The 72-ton auxiliary building overhead crane will be used to lift the racks from the auxiliary building loading bay to a location on the operating floor adjacent to SFP B. There, a 20-ton hoist will be attached to the auxiliary building overhead crane, and a special lifting rig will be attached to the hoist. The 20-ton hoist will be suspended from the auxiliary building overhead crane and used in conjunction with the special lifting rig to lift and move the racks into the SFP.

In the SE dated July 13, 1984, we approved the auxiliary building overhead crane and found that the general load handling system and policy and procedures at CR-3 are consistent with the intent of the guidelines in NUREG-0612. However, the 20-ton hoist and the special lifting rig were obtained specifically to lift the new rack modules, therefore, they are not addressed in the SE.

As stated in the July 13, 1984, SE, the auxiliary building overhead crane was designed and fabricated in 1969 to the industry standard in effect at that time, the "Specification for Electric Overhead Traveling Cranes." On the basis of a detailed comparison of the crane with the requirements of Crane Manufacturers Association of America Specification No. 70, "Specifications for Electric Overhead Traveling Cranes," and ANSI/ASME B30.2-1976, "Safety

Standards for Overhead and Gantry Cranes (Top Running Bridge and Multiple Girder)," the auxiliary building crane was determined to, either, fully satisfy or meet the intent of these standards. It was also determined that the inspection, testing, and maintenance of this crane met the intent of ANSI B30.2-1976. In addition, the licensee stated in their June 29, 2000, Request for Additional Information (RAI) response that this crane was inspected and load tested in accordance with ANSI N14.6 (1978), "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials."

The licensee states that the spent fuel storage racks will be lifted using the 20-ton hoist and a remotely engaged spent fuel rack lifting rig that is specially designed to lift the spent fuel racks. The hoist complies with ASME/ANSI B30.16, "Overhead Hoists (Underhung)," which requires the maximum design stress be no more than 20 percent of ultimate material strength for load-bearing parts. The heaviest load to be lifted during rack replacement weighs less than 20,000 lbs., consisting of a rack plus the lifting rig. Thus, the hoist provides double the normal stress design factor, including dynamic load factor, for handling heavy loads, as required by NUREG-0612. The hoist is load tested and inspected in accordance with the requirements for special lifting and handling equipment as described in ANSI N14.6.

The lifting rig is designed and tested in accordance with the guidelines in NUREG-0612, Sections 5.1.6(1), 5.1.6(3a), and the requirements in ANSI N14.6 (1978). Accordingly, the lifting rig has twice the normal stress design safety factor, including dynamic load factor, for handling the critical loads. The lifting rig also is redundantly designed with four independently loaded lift rods that engage the rack at four lift points. The lifting rig is load tested to 3 times the heaviest load to be lifted (plus a dynamic load factor) and inspected in accordance with the requirements for special lifting and handling equipment as described in ANSI N14.6.

We believe that the capacity of the 72-ton auxiliary building overhead crane coupled with the capacity of the 20-ton hoist and the special lifting rig will support the weight of the racks and the added rigging loads without risk of a load drop. In addition, the design, inspection and testing of the crane and lifting device will help to assure the reliability of the system to enhance safe handling of the racks and further reduce any potential risk of an accidental rack drop during rack installation.

3.7.2 Load Path

The proposed replacement racks will be lifted to the auxiliary building operating level and moved adjacent to SFP B using the auxiliary building overhead crane. At the operating level, the racks will be moved along the previously NRC-approved safe-load path. The racks will then be lowered into the pool to a minimum height of 6 inches above the pool liner for movement into their permanent location. The 20-ton hoist will be attached to the auxiliary building overhead crane in order to lower the new racks into the pool and position them in their location. The licensee states that the racks and lifting rig will not be carried over any fuel or near fuel in the SFP. The racks and lifting rig will also travel the same safe-load path as for the fuel shipping cask. Since the weight of the cask is greater than the weight of the rack, this is an acceptable load path. Fuel will be shuffled into racks that are not in the safe-load path. Movement of the racks along the SFP floor shall not exceed 6 inches above the floor. During movement across the SFP operating deck, the lift height of the racks will be minimized. When a rack is near

SFP B, such that the rack could fall into the pool if dropped, the height of the rack will be maintained less than 6 inches above the operating deck, which is consistent with the load-drop analysis.

The new installed fuel storage racks will not significantly change the method of handling loads during normal plant operations because the same equipment (i.e., the fuel handling system), methods, and procedures as those used prior to the rerack will be used following the installation of the new replacement spent fuel storage racks.

We find that, based on the above discussion, safe-load paths for movement of the spent fuel storage racks do not involve any movement of the racks over spent fuel and are conducted under administrative controls implemented by the licensee. The rack replacement process will use the previously approved safe-load path for the fuel shipping cask, which weighs more than a storage rack, and therefore the safe-load path for movement of the racks is acceptable.

3.7.3 Analysis of Heavy Load-Drop Accidents

The licensee analyzed postulated load-drop accidents of a spent fuel assembly onto the new fuel storage racks and the drop of a spent fuel storage rack onto the SFP floor. Three orientations for a fuel assembly drop were considered, using a bounding impact weight of 2750 lbs. (includes the weight of the heaviest fuel assembly plus the handling tool) from a maximum lift height of 24 inches above the racks: a vertical drop on top of the racks ("shallow-drop"), drop of an assembly on top of a rack with the assembly in an inclined position, and a vertical drop to the base plate of the racks ("deep drop"). The fuel assembly drop on top of the spent fuel storage racks resulted in deformation of the racks to a maximum depth of 6.0 inches below the top of the rack, but no damage to the fuel. The fuel assembly drop onto the base plate of the racks resulted in no puncture damage to the SFP liner.

A vertical drop of the heaviest rack (one of the current SFP B racks) that weighs 17,715 lbs. from 6 inches above the SFP operating deck to the pool floor was evaluated. The combined weight of this rack and the lifting rig is less than 20,000 lbs., which is the weight used for the analysis. The results indicated that the SFP liner would be punctured and SFP water would leak. The SFP concrete floor slab would be indented to a depth of less than 6 inches, however, no structural damage to the concrete would occur. The licensee stated that any leakage through the SFP would be collected by the leak chase system. As stated in the licensee's response to the RAI dated June 29, 2000, valves in the leak chase system will be maintained closed during the rerack operation, therefore, the closed valves will isolate any SFP leakage and minimize any loss of inventory from the SFP. The drop of a rack onto another rack containing fuel was not evaluated because the licensee will preclude the racks from being moved over racks with fuel.

Furthermore, SFP makeup can be obtained from a number of sources to supplement any leakage from the SFP. As noted in FSAR Section 9.3.2.8, "Makeup Capability," makeup can be provided from the Decay Heat System, the Demineralized Water Supply System and temporary fire hoses.

Therefore, because the structural integrity of the concrete slab in the SFP remains unimpaired after a rack drop, and because the licensee has the capability to isolate an SFP leak and provide makeup to supplement any SFP inventory loss, the staff agrees with the licensee's conclusion that neither catastrophic damage of the SFP structure nor rapid loss of pool water would occur to cause uncovering of fuel stored in SFP B.

3.7.4 Load-Handling Procedures

NUREG-0612 recommends that licensees provide an adequate defense-in-depth approach to maintaining safety during the handling of heavy loads near spent fuel and cited four major causes of accidents: operator errors, rigging failures, lack of adequate inspection, and inadequate procedures. The licensee stated that they will implement measures using plant programs and procedures to minimize the potential for load-drop accidents during the reracking operations. The licensee stated that the potential is further minimized by compliance with recognized standards for heavy loads equipment such as ANSI N14.6, and NUREG-0612.

The licensee's conclusion that, based on the load-drop analyses, the integrity of the fuel and the SFP would be maintained if a fuel assembly or a spent fuel storage rack is dropped is acceptable to the staff. The design and capability of the crane and lifting devices in conjunction with the administrative procedures and controls that are focused on, but not limited to, the areas noted above would enable the licensee to help maintain safety during the rerack operation.

3.7.5. Summary

Based on the preceding discussions, we find that the aforementioned considerations for the movement of heavy loads to support the proposed increase in the SFP storage capacity at CR-3 are acceptable. The licensee's use of the cranes, the special lifting rig for the spent fuel racks, and the administrative controls and procedures enhances the licensee's capability to reduce the potential for load drops and are in accordance with NUREG-0612 and ANSI N14.6. The design, testing and inspection of the cranes, the hoist, lifting rig and other lifting devices will assure that the hoisting system is highly reliable and enable the licensee to safely handle the racks and other heavy loads during the rack installation process. The licensee's postulated accident analyses of the spent fuel storage rack indicate that the SFP liner could be breached. However, due to the capabilities of the leak chase and SFP makeup systems, the licensee is capable of maintaining the SFP and its contents within the acceptable consequence limits set forth in NUREG-0612.

3.8 PROPOSED TECHNICAL SPECIFICATION REVISIONS

The licensee proposed changes to the following Technical Specifications (TS) in support of the rack replacement project in SFP B and the change in new fuel storage configuration in SFP A.

In Section 3.7.14, "Spent Fuel Pool Boron Concentration", references to Pool B, Region 2, would be deleted, since all of the racks in SFP B would be of the same design after rack replacement.

In Section 3.7.15, "Spent Fuel Assembly Storage", Figure 3.7.15-3 would be deleted, since this figure specified acceptable storage areas for the existing SFP B Region 2. References to

Figure 3.7.15-3 would be deleted, as well as deleting reference to storage in accordance with the Final Safety Analysis Report, since all acceptable storage configurations would be specified in the TS. Figure 3.7.15-1 would be revised by adding a note that specifies that storage of fuel in a checkerboard pattern with empty cells is acceptable in any region of SFP A. Figure 3.7.15-2 would be revised to indicate the acceptable fuel assembly storage locations in SFP B.

Section 4.3.1, "Criticality", would be changed to reflect the dimensions of the replacement racks, and to delete reference to Pool B, Region 2.

Section 4.3.3, "Capacity", would be changed to reflect the increased combined storage capacity of 1474 fuel assemblies.

The Bases for Sections 3.7.14 and 3.7.15 would be changed to reflect the installation of the new racks, changes to new fuel storage locations, and to update associated references.

The staff has reviewed the proposed changes to the TS. Based on the above evaluations, the staff finds the proposed changes to TS 3.7.14 "Spent Fuel Pool Boron Concentration," 3.7.15 "Spent Fuel Assembly Storage," 4.3.1 "Criticality," and 4.3.3 "Capacity," reflect the results of the criticality analysis. In addition, the changes are appropriate to allow use of the new high-density racks in SFP B and the associated increased spent fuel storage capacity, and, therefore, are acceptable. The staff also finds that the associated Bases changes are consistent with the approved TS changes.

4.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission (NRC), 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 September 12, 2000 (65 FR 55059). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Florida Power Corporation, Crystal River Unit 3, License Amendment Request #239, Revision 0, Attachment E, "Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5 percent Enriched Mark B-11 Fuel in Checkerboard Arrangement with Water Holes," HI-992285, Holtec International, August 1999.

2. Florida Power Corporation, Crystal River Unit 3, License Amendment Request #239, Revision 0, Attachment F, "Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3," HI-992128, Holtec International.

3. Letter, B. K. Grimes (NRC) to All Power Reactor Licensees (with Enclosure), "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

4. Crystal River Unit 3, Final Safety Analysis Report.

5. Letter, Florida Power Corporation to U.S. NRC, "Response to NRC Request for Additional Information regarding Enhanced Spent Fuel Storage - License Amendment Request #239 (TAC No. MA6754)," dated May 3, 2000.

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Date: **September 13, 2000**

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