December 15, 1995

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Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear
Licensing
15760 W. Power Line Street
Crystal River, Florida 34428-6708

Dear Mr. Beard:

SUBJECT: CRYSTAL RIVER NUCLEAR GENERATING PLANT UNIT 3 - ISSUANCE OF AMENDMENT RE: FUEL ENRICHMENT INCREASE (TAC NO. M91536)

The Commission has issued the enclosed Amendment No. 151 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The amendment consists of changes to the Technical Specifications (TS) in response to your application dated January 26, 1995 as supplemented March 9, and May 24, 1995, to increase the allowable U-235 enrichment of fuel to be stored in the new and spent fuel storage facilities.

Please be advised that you should perform evaluations of reload core designs using any enrichment on a cycle by cycle basis as part of the reload safety evaluation process. To ensure that reactor operation is acceptable each reload design should be evaluated to confirm that the cycle core design adheres to the limits of the accident analyses and the TS.

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

Sincerely,
Original signed by:
George F. Wunder, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 151 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: G:\CRYSTAL\CR91536.AMD

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Mr. Percy M. Beard, Jr.
Florida Power Corporation

Crystal River Unit No. 3
Generating Plant

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AMENDMENT NO. 151 TO FACILITY OPERATING LICENSE NO. DPR-72
CRYSTAL RIVER UNIT 3

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
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CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 151
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 licensee) dated January 26, 1995, as supplemented March 9 and May 24, 1995, 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.

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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, asrevised through Amendment No. 151, 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Bart C. Buckley for

David B. Matthews, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 15, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 151

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. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
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
4.0-1	4.0-1
4.0-2	4.0-2
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

3.7.15 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, Figure 3.7.15-2, Figure 3.7.15-3, or stored in accordance with the FSAR.

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, Figure 3.7.15-2, Figure 3.7.15-3, or in accordance with the FSAR.	Prior to storing the fuel assembly in Storage Pool A or Storage Pool B.

MINIMUM BURNUP REQUIRED FOR
"A" POOL STORAGE

Minimum Burnup vs Initial Nominal Enrichment

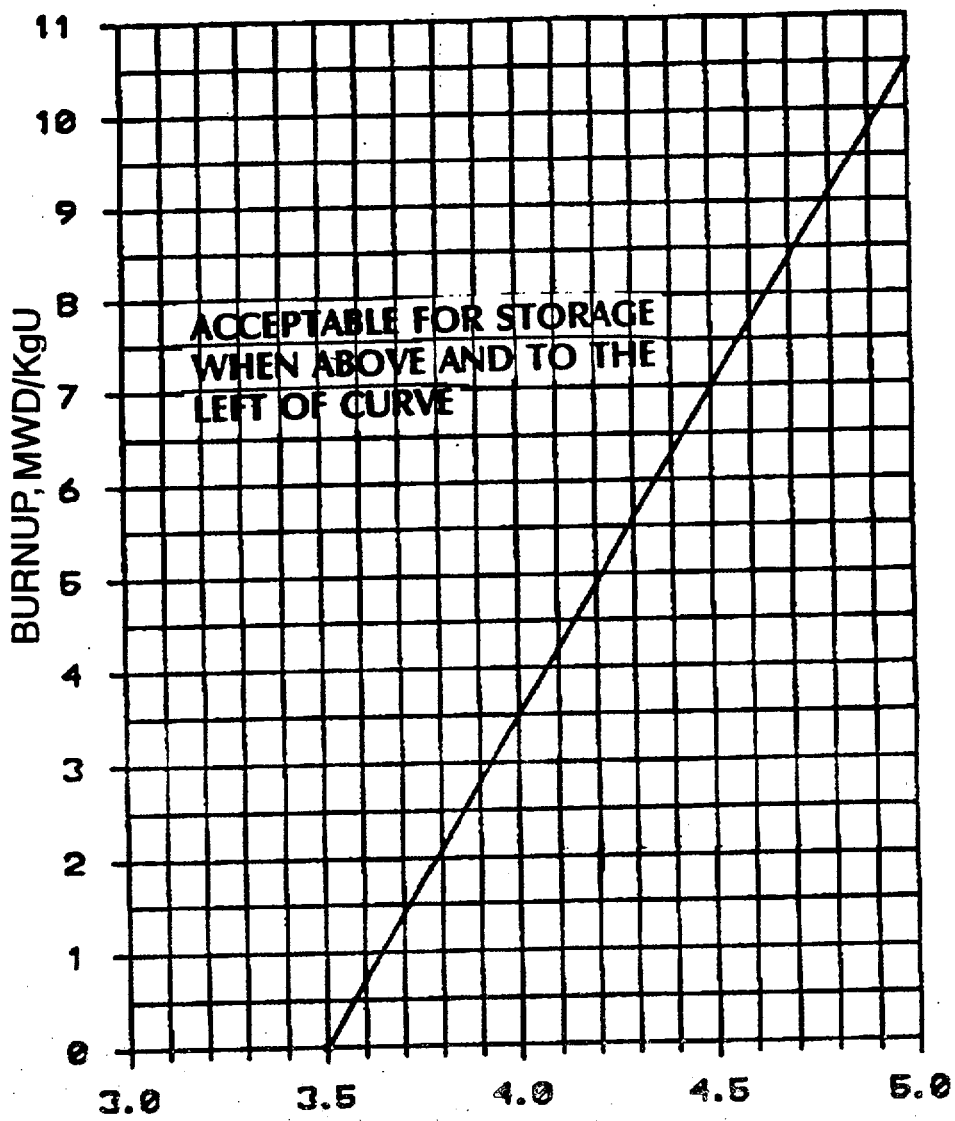

Figure 3.7.15-1 (page 1 of 3)
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool A

MINIMUM BURNUP REQUIRED FOR
REGION 1 OF "B" POOL

Minimum Burnup vs Initial Nominal Enrichment

Burned Fuel in checkerboard Configuration with 5.0 wt% Fresh Fuel

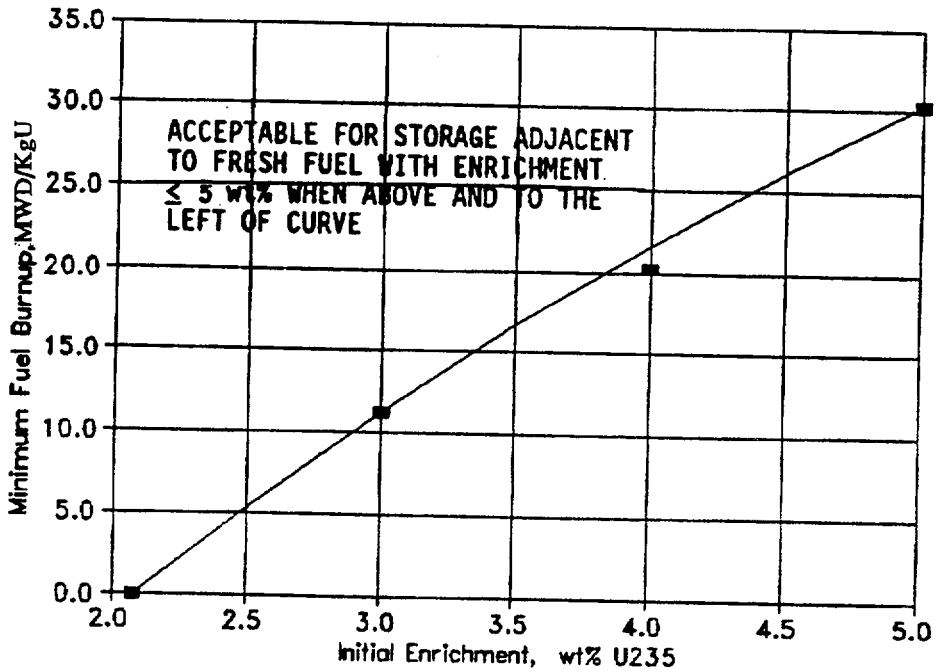

Figure 3.7.15-2 (page 2 of 3)
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool B, Region 1

MINIMUM BURNUP REQUIRED FOR
REGION 2 OF "B" POOL

Minimum Burnup vs Initial Nominal Enrichment

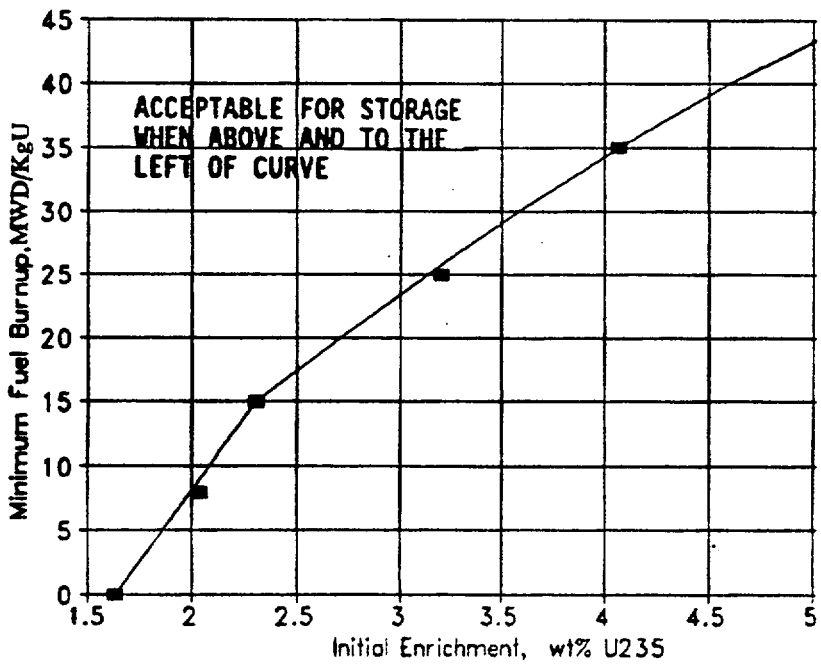

Figure 3.7.15-3 (page 3 of 3)
Burnup versus Enrichment Curve for
Spent Fuel Storage Pool B, Region 2

4.0 DESIGN FEATURES

4.1 Site

The 4,738 acre site is characterized by a 4,400 foot minimum exclusion radius centered on the Reactor Building; isolation from nearby population centers; sound foundation for structures; an abundant supply of cooling water; an ample supply of emergency power; and favorable conditions of hydrology, geology, seismology, and meteorology.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each fuel assembly shall consist of a matrix of Zircaloy-4 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material, with a maximum enrichment of 5.0 weight percent U-235. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. Each fuel rod shall have a nominal active fuel length of 144 inches and shall contain a maximum total weight of 2253 grams uranium.

4.2.2 CONTROL RODS

The reactor core shall contain 60 safety and regulating (including extended life CONTROL RODS) and 8 AXIAL POWER SHAPING (APSR) rods. Except for the extended life CONTROL RODS, the CONTROL RODS shall contain a nominal 134 inches of absorber material. The extended life CONTROL RODS shall contain a nominal 139 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium, and 5 percent cadmium. Except for extended life CONTROL RODS, all CONTROL RODS shall be clad with stainless steel tubing. The extended life CONTROL RODS shall be clad with Inconel. The APSRs shall contain a nominal 63 inches of absorber material at their lower ends. The absorber material for the APSRs shall be 100 % Inconel.

(continued)

4.0 DESIGN FEATURES (continued)

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 10.6 inch center to center distance between fuel assemblies placed in Region 1 of the B pool;

d. A nominal 9.17 inch center to center distance between fuel assemblies placed in Region 2 of the B pool; and

e. 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. $k_{eff} \leq 0.95$ if 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)

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.

Spent fuel pool B also contains high density racks separated into 2 regions. The racks in Region 1 have a 10.60 inch center-to-center spacing capable of storing 174 assemblies. The high density racks in Region 2 have 9.17 inch center-to-center distance capable of storing 641 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.

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 (Ref. 2).

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 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 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 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 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 required fuel assembly burnup as a function of initial enrichment.)

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 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 $\leq$ 5.0 weight percent U-235,

2. For Region 1, fuel with initial enrichment $\leq$ 5.0 weight percent and $\geq$ 2.08 weight percent fuel burnup must be within the limits specified in Figure 3.7.15-2 and arranged in a required checkerboard configuration with new fuel or burned fuel of $\leq$ 5.0 weight percent, and

3. For spent fuel with initial enrichment $\leq$ 5.0 weight percent and $\geq$ 1.63 weight percent in Region 2, fuel burnup must be within the limits specified in Figure 3.7.15-3. (Figure 3.7.15-3 presents required fuel assembly burnup as a function of initial enrichment.)

(continued)

BASES

LCO (continued)

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, 3.7.15-2 and 3.7.15-3. 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, Figure 3.7.15-2, Figure 3.7.15-3, or the FSAR, 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.

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.

(continued)

BASES (continued)

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, Figure 3.7.15-2, and Figure 3.7.15-3 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 analysis (Ref. 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. Crystal River Unit 3 Spent Fuel Storage Pool B Criticality Analysis, W.A. Wittkopf, L.A. Hassler, B&W Fuel Company, BAW-2209P, October 1993.

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, Revision 11.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TECHNICAL SPECIFICATION CHANGE REQUEST FOR

FUEL ENRICHMENT INCREASE

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

1.0 INTRODUCTION

By letter dated January 26, 1995, as supplemented March 9, 1995, Florida Power Corporation (FPC or the licensee) requested changes to the Crystal River Unit 3 Technical Specifications (TS) 3.7.15, 4.2.1 and 4.3 to allow an increased limit for fuel enrichment. The proposed changes would allow for the storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (w/o) U-235 in the new (fresh) and spent fuel storage racks. In response to the staff's request, by letter dated May 24, 1995, the licensee provided additional information which did not change the staff's proposed no significant hazards consideration.

The staff's evaluation of the criticality aspects of the proposed changes follows.

2.0 EVALUATION

New Fuel Storage Vault

The new fuel storage vault provides a 6 x 11 cell array of storage locations arranged on a 21.128-inch lattice spacing. The storage vault is intended for the receipt and storage of fresh fuel under dry (air) conditions. However, to assure the criticality safety under normal and accident conditions and to conform to the requirements of General Design Criterion (GDC) 62 for the prevention of criticality in fuel storage and handling, two separate criteria must be satisfied as defined in NRC Standard Review Plan (SRP), Section 9.1.1. These criteria state that the maximum reactivity of the fully loaded fuel racks shall not exceed a k_{eff} of 0.95 if fully flooded with unborated water or a k_{eff} of 0.98 assuming the optimum hypothetical low density moderation (e.g., fog or foam). The maximum calculated reactivity must include a margin for uncertainties in reactivity calculations and in manufacturing tolerances such that the true k_{eff} will not exceed the calculated maximum value at a 95% probability, 95% confidence level (95/95).

The analysis of the reactivity effects of fuel storage in the new fuel storage vault was performed for FPC by Holtec International with the three-dimensional multi-group Monte Carlo code, KENO-5a, using neutron cross sections generated

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by the NITAWL code package from the 27 energy group SCALE data library. This code is widely used for the analysis of fuel rack reactivity and has been benchmarked against results from numerous critical experiments which simulate the Crystal River storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, fuel rod size, and fuel assembly spacing. To minimize the statistical uncertainty of the KENO-5a reactivity calculations, a minimum of 500,000 neutron histories were typically accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO-5a reactivity calculations. Based on the above, the staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Crystal River new fuel storage racks with a high degree of confidence.

The reference design fuel assembly used in the calculations was the standard Babcock & Wilcox 15 x 15 array of fuel rods, with 17 rods replaced by 16 control rod guide tubes and one instrument thimble. The reactivity calculations included a calculational bias and uncertainty derived from the benchmark calculations, as well as uncertainties associated with manufacturing tolerances due to lattice spacing, fuel enrichment, and fuel density. The manufacturing uncertainties were based on the reactivity difference between nominal and maximum tolerance values and, therefore, meet the 95/95 probability/confidence level requirement.

The maximum k_{eff} for fuel assemblies of 5.0 w/o U-235 under fully flooded conditions was calculated to be 0.9479. For the hypothetical low-density optimum moderation condition, the maximum calculated k_{eff} was 0.9783 at a moderator density of approximately 7.5% of full density. These calculations were based on maintaining two rows (row 4 and 8) of storage locations empty of fuel. Fuel assemblies are prevented from storage in these two rows by bolted stainless steel cover plates. The results conform to the acceptance criteria of SRP 9.1.1 and are, therefore, acceptable.

Spent Fuel Pool A

The storage rack design in pool A is composed of a B_4C neutron absorber sandwiched between two 0.060-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 stored fuel assemblies. The plates have a thickness of 0.075 inches and a nominal boron-10 (B-10) loading of 0.015 gm/cm^2. Although the pool water normally contains approximately 2000 ppm of boron, to conform to the requirements of GDC 62 and to assure the criticality safety under all conditions, the criterion stated in SRP 9.1.2 must be satisfied. This criterion states that the maximum reactivity of the fully loaded racks shall not exceed a k_{eff} of 0.95 if fully flooded with unborated water. The maximum calculated reactivity must include 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.

The reactivity calculations for the spent fuel racks in pool A were performed by Holtec International with both the NITAWL-KENO-5a code and the CASMO-3 code. CASMO-3, which was used for burnup calculations, is a two-dimensional

transport theory code which has been acceptably benchmarked against numerous critical experiments which simulate the Crystal River 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. As in the new storage vault analyses previously mentioned, a minimum of 500,000 neutron histories were accumulated in each KENO-5a calculation. 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 standard Babcock & Wilcox 15 x 15 fuel assembly was used as the reference design in the calculations. In addition to the calculational bias and uncertainty derived from the benchmarks and the uncertainties due to manufacturing tolerances, an uncertainty due to the burnup calculations was included. The uncertainties meet the 95/95 probability/confidence level requirement and are, therefore, acceptable.

To enable the storage of fuel assemblies with nominal enrichments greater than 3.5 w/o U-235, the concept of burnup credit reactivity equivalencing was used. This is predicated upon the reactivity decrease associated with fuel depletion and has been previously accepted by the staff for spent fuel storage analysis in pressurized water reactors (PWRs). For burnup credit, a series of reactivity calculations are performed to generate a set of initial enrichment versus fuel assembly discharge burnup ordered pairs which all yield an equivalent k_{eff} less than 0.95 when stored in the spent fuel storage racks. The results show that a fresh 3.5 w/o enriched fuel assembly yields the same rack reactivity (k_{eff} of 0.9435) for the pool A storage racks as an initially enriched 5.0 w/o assembly depleted to 10.5 MWD/KgU. The minimum burnup requirements for storage in Pool A are given in TS Figure 3.7.15-1.

The accidental misloading of a fresh 5.0 w/o enriched assembly into a storage cell in Pool A resulted in a maximum k_{eff} of 0.946, including uncertainties, with all other cells filled with fuel of the maximum permissible reactivity and with no credit for soluble boron in the pool water. This meets the NRC criticality criterion of k_{eff} no greater than 0.95 and is acceptable.

Spent Fuel Pool B

Spent fuel pool B consists of two regions. Region 1 (174 locations) consists of high density fuel assembly spacing obtained by utilizing Boraflex as a neutron absorber on each cell wall and is reserved for core off-loading. The nominal cell pitch is 10.60 inches with two Boraflex plates between stored assemblies. Region 2 (641 locations) also consists of high density fuel assembly spacing using Boraflex and provides normal storage for spent fuel assemblies. The cell pitch for the Region 2 racks is 9.17 inches with only one Boraflex plate between assemblies.

The reactivity calculations for the racks in pool B were performed for FPL by the B&W Fuel Company (BWFC) using CASMO-3 and KENO-4 (300,500 neutron histories). Validation of these methods was accomplished by comparison with critical experiment data for assemblies, spacings, and interspersed absorbers similar to those in the Crystal River storage racks. The staff concludes that the benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to the Crystal River racks.

The standard Babcock & Wilcox 15 x 15 fuel assembly was used in the pool B reactivity calculations. In addition to the calculational bias and uncertainty derived from the benchmarks, tolerance penalties and uncertainties due to stainless steel wall thickness, cell ID, cell pitch, off-center fuel assembly position, pool water temperature, and Boraflex shrinkage and gap formation were considered.

The results show that the maximum k_{eff} for a Region 1 checkerboard pattern with fuel enriched to a nominal 5.0 w/o U-235 alternating with empty cells (water) is 0.8285 ± 0.0015, well within the NRC acceptance criterion of k_{eff} no greater than 0.95.

Analyses were also performed for a checkerboard pattern of fresh 5.0 w/o U-235 fuel alternating with burned fuel. An additional uncertainty of 0.0088 Δk was included for the depletion calculations. The results show that the reactivity of a fresh assembly of nominal 2.08 w/o U-235 enrichment is equivalent to an assembly initially enriched to a nominal 5.0 w/o U-235 and depleted to 30.3 GWD/MTU. The resulting k_{eff} is 0.9200, thus meeting the NRC criterion of no greater than 0.95. The minimum burnup requirements for Region 1 storage in a checkerboard configuration with fresh 5.0 w/o fuel are given in TS Figure 3.7.15-2.

For Region 2, fuel with initial nominal enrichments up to 5.0 w/o U-235 and depleted to approximately 43 GWD/MTU results in the same rack reactivity as fresh fuel of nominal 1.63 w/o U-235 enrichment. In this case, k_{eff} is 0.9499, which meets the 0.95 criticality criterion. The minimum burnup requirements for Region 2 storage are given in TS Figure 3.7.15-3.

The misplacement of a fresh nominally enriched 5.0 w/o fuel assembly into a position requiring a depleted assembly results in an increase in the reactivity of the storage rack. However, for this event, credit may be taken for the 2000 ppm of soluble boron in the pool water (double contingency principle). Even if all checkerboarded burned fuel positions were to contain 5.0 w/o fresh fuel, the rack configuration would be well subcritical with 2000 ppm of soluble boron (k_{eff} = 0.855).

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the Crystal River Unit 3 new and spent fuel storage racks are acceptable and meet the requirements of GDC 62 for the prevention of criticality in fuel storage and handling. The proposed TS revisions appropriately reflect the criticality analyses and are acceptable.

Although the Crystal River TS have been modified to specify the above-mentioned fuel as acceptable for storage, evaluations of reload core designs using any enrichment will, of course, be performed on a cycle by cycle basis as part of the reload safety evaluation process. Each reload design is evaluated to confirm that the cycle core design adheres to the limits that exist in the accident analyses and TS to ensure that reactor operation is acceptable.

Spent Fuel Pool Cooling

The staff evaluated the effects of the proposed changes on the Spent Fuel Pool Cooling System. The SFP Cooling System is designed to maintain water clarity

and remove decay heat from the spent fuel pool. The staff performed calculations to ensure that the increase in decay heat as a result of the change in allowable enrichment of the stored fuel did not exceed previously accepted calculations. The staff found that the increase in the decay heat load was negligible and was bounded by the analyses performed by the staff in support of Amendment 134 to the Crystal River Unit 3 Operating License dated April 16, 1991, and is therefore acceptable.

An issue associated with spent fuel pool cooling adequacy was identified in NRC Information Notice 93-83 and its Supplement 1, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," dated October 7, 1993 and August 24, 1995, respectively, and in a 10 CFR Part 21 notification, dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the licensee under separate cover.

3.0 STATE CONSULTATION

Based upon the written notice of the proposed amendment, the Florida State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR Sections 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been published (60 FR 64183) in the Federal Register on December 14, 1995. Accordingly, the Commission has determined that the issuance of this amendment will not result in any significant environmental impact other than those evaluated in the Final Environmental Statement.

5.0 CONCLUSION

Based on the staff evaluation in Section 2.0 above, the staff concludes that the proposed Technical Specifications changes are acceptable.

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.

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