D7MB-016

October 14, 1986

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
P. O. Box 14042; M.A.C. H-3
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 92 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 10, 1985.

This amendment changes the TSs to increase the enrichment of fuel assemblies stored in Spent Fuel Pool B and the dry fuel storage rack.

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

Sincerely,

ORIGINAL SIGNED BY

Brenda Mozafari, Project Manager
PWR Project Directorate #6
Division of PWR Licensing-B

Enclosures:
1. Amendment No. 92 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
Generating Plant

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

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
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 92
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 December 10, 1985, 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. 92 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
PWR Project Directorate #6
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 14, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 92

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
5-4	5-4
5-5	5-5

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The Reactor Containment building is designed and shall be maintained for a maximum internal pressure of 55 psig and a temperature of 281°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy - 4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2253 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.0 (nominal) weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 60 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The APSRs shall contain a nominal 63 inches of absorber material at their lower ends. The absorber material for the APSRs shall be 100% Iconel.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line, which is 670°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,180 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEORLOGICAL TOWER LOCATION

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

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The dry storage racks and the spent fuel storage racks in pool "B" are designed and shall be maintained with a nominal 21-1/8 inch center-to-center distance between fuel assemblies placed in the storage racks. The high density spent fuel storage racks in pool "A" are designed and shall be maintained with a nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks. All of these rack designs ensure a keff equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The keff of $\leq$ 0.95 includes a conservative allowance of >1% $\Delta K/K$ for uncertainties. In addition, fuel stored in pool "A" shall have a U-235 loading of $\leq$ 46.14 grams of U-235 per axial centimeter of fuel assembly ($\leq$ an enrichment of 3.5 weight percent U-235). Fuel stored in the dry storage racks and pool "B" shall have a U-235 loading of $\leq$ 52.73 (nominal) grams of U-235 per axial centimeter of fuel assembly ($\leq$ an enrichment of 4.0 (nominal) weight percent U-235).

DRAINAGE

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

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1153 fuel assemblies and 6 failed fuel containers.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

INTRODUCTION

By letter dated December 10, 1985, Florida Power Corporation (FPC or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The proposed amendment would increase the permitted enrichment for fuel to be stored in both Spent Fuel Pool B and the dry fuel storage rack at CR-3. The current analyses and TSs permit a 3.5 weight percent uranium-235 enrichment for all storage areas. This amendment would increase the enrichment to 4.0 weight percent for Storage Pool B and the dry fuel storage rack only. In support of the proposed amendment, the licensee has submitted two reports (Refs. 2 and 3) prepared by the Southern Science Office of Black & Veatch (Southern Science). The NRC staff has reviewed the proposed amendment and prepared the following evaluation.

EVALUATION

a. Spent Fuel Pool B

Four independent methods of evaluation were used to provide confidence in the reference criticality calculations. These methods included the (1) CASMO-2E program, (2) AMPX-KENO computer package with the NITAWL subroutine for performing uranium-238 resonance shielding calculations using the Nordheim integral treatment, and (3) NULIF-SNEID diffusion theory method of analysis. The AMPX-KENO calculations were performed using both a 123 group cross section library and a more recently developed 27 group cross section library. The AMPX-KENO computer codes are widely used in the calculation of spent fuel pool criticality where AMPX is used to generate the neutron cross section data and KENO is used to perform the Monte Carlo criticality calculations. The CASMO-2E code is a two-dimensional, transport theory code for calculating fuel assemblies that is used by a number of organizations to perform reactor core physics calculations. The NULIF-SNEID diffusion theory method of analysis is used by Southern Science. Appendix A of Reference 2 describes the benchmarking that has been performed by Southern Science for its AMPX-KENO and CASMO-2E methods of evaluation. Based on our review of these benchmark calculations, we conclude that use of the AMPX-KENO and CASMO-2E by Southern Science is acceptable for the criticality evaluation of Spent Fuel Pool B.

The reference calculation was performed for a Babcock & Wilcox (B&W) 15x15 fuel assembly for 4.0 weight percent uranium-235 enrichment and for nominal fuel assembly composition. The nominal Spent Fuel Pool B rack lattice spacing (center-to-center distance between rack locations) of 21.125 inches was used. Pure water at a temperature of 20°C (density of 0.998 grams per cubic centimeter) was assumed. The lattice of storage locations is assumed to be infinite in all directions. All four independent calculational methods gave values of the effective multiplication factor, K_{eff}, that were in good agreement. Since CASMO-2E gave the highest value of K_{eff}, it was used by Southern Science as the reference value of K_{eff} for added conservatism. This reference value of K_{eff} is 0.9221 and includes the calculational bias derived from the benchmarking results. This evaluation of the reference case K_{eff} is acceptable since a conservative value has been obtained with appropriately benchmarked methods for a 15x15 B&W fuel assembly with 4.0 percent enriched fuel at nominal rack dimensions and spent fuel pool water conditions.

The uncertainties treated included those due to rack lattice spacing, eccentricity of fuel assembly placement in a rack location, and fuel enrichment and density variation. The effect of these uncertainties when combined statistically at the 95/95 probability/confidence level is ±0.0024 in reactivity. The uncertainty in the calculational methodology based on the benchmarking results is ±0.0018 in reactivity on a 95/95 probability/confidence level. Statistically combining these two uncertainties gives an overall uncertainty of ±0.0030 on a 95/95 probability/confidence level. Therefore, a K_{eff} of 0.925 (0.9221 + 0.0030) is conservatively estimated to be the maximum K_{eff} under the worst possible combination of calculational and mechanical uncertainties with a 95 percent probability at a 95 percent confidence level under normal conditions. This K_{eff} of 0.925 is acceptable since it meets the NRC staff's criterion of 0.95 for this quantity.

The loss of pool cooling with a subsequent increase in the spent fuel pool temperature has been analyzed. The results presented show that reactivity decreases from its nominal value with an increase in pool temperature. A calculation simulating boiling within the fuel assemblies gave a decrease in reactivity from the reactivity for the pool water at nominal temperature.

Heat generated by 4.0 percent enriched fuel in Spent Fuel Pool B will not differ significantly from that generated by 3.3 percent enriched fuel for the same core power density. Therefore, the existing spent fuel pool cooling system will be able to maintain a temperature of 129°F when the fuel assemblies from 16 successive refuelings (944 spent fuel assemblies) are stored in the pools and to maintain a spent fuel temperature of 140°F or less when a fuel core is discharged to the spent fuel pools in addition to the 944 assemblies noted above. This cooling capability was previously reviewed and approved by the NRC staff as documented in the Safety Evaluation transmitted to the licensee by letter (Reference 4) dated November 17, 1980, from Robert Reid, NRC, to J. A. Hancock, Florida Power Corporation.

A fuel assembly accidentally positioned outside the rack cannot be located any closer than 10 inches from another fuel assembly and will, therefore, have a negligible reactivity effect. An assembly dropped on top of the rack will be about 20 inches away from other fuel assemblies and will have a negligible reactivity effect. Moreover, for the accident conditions analyzed, credit may be taken for the soluble boron present in the water which would reduce K_{eff} significantly below the criterion of 0.95. We conclude that credible accident configurations will not lead to a reduction in the margin to criticality for Spent Fuel Pool B.

Irradiation of 4.0 percent enriched fuel will not appreciably increase the quantity of radioactive species in the spent fuel assemblies over that for fuel enriched to 3.3 or 3.5 percent. Therefore, the radioactive doses both within and outside the plant resulting from an accidental drop of a permitted load over Spent Fuel Pool B will not change over those previously analyzed and approved by the NRC staff.

b. Dry Fuel Storage Rack

The new configuration of the dry fuel storage rack consists of a 6x11 array of storage cells with a center-to-center spacing of 21.125 inches. This storage cell spacing is identical to that of the storage rack in Spent Fuel Pool B. Therefore, the calculation performed for the Spent Fuel Pool B rack for K_{eff} as a function of enrichment and the associated uncertainties are directly applicable to the dry fuel storage rack when flooded with pure water at a density of 1 gram per cubic centimeter. The maximum K_{eff} is 0.944 for the 15x15 B&W fuel assembly with a uranium 235 enrichment of 4.5 percent. This K_{eff} meets our criterion of 0.95 and is, therefore, acceptable.

The dry fuel storage rack was also analyzed for low-density water such as may occur for fog or mist. The analysis was performed with the KENO Monte Carlo code using the 123 group neutron cross section library. The results of preliminary analyses indicated that if all locations were filled with fuel, the criticality criterion would not be met for the 4.5 weight percent enriched fuel. A new configuration was selected such that the original 66 storage locations arranged in a 6x11 array were reduced to 54 storage locations arranged in three 3x6 arrays. Two rows of storage locations (rows 4 and 8), each containing six storage locations, would be blocked to prevent fuel from being placed within. The analysis of this new fuel storage configuration showed that the maximum K_{eff} occurred at a water density of 0.075 grams per cubic centimeter. The K_{eff} for this configuration is 0.941. Including uncertainties at least at a 95/95 probability/confidence level gives a maximum K_{eff} of 0.952 for fuel having a 4.5 weight percent uranium-235 enrichment. This K_{eff} of 0.952 is acceptable since it meets the NRC staff's criterion of 0.98 for this quantity.

Since the NRC staff's criteria for the storage of 4.5 weight percent uranium-235 enrichment fuel are met, the storage of 4.0 weight percent uranium-235 enrichment fuel in the dry storage rack is, therefore, acceptable.

On the basis of our review described above we conclude that fuel of the B&W 15x15 design having enrichment no greater than 4.0 weight percent uranium-235 may be safely stored in Spent Fuel Pool B and in the dry fuel storage rack, and the revised TSs are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 14, 1986

Principal Contributors: D. Fieno, N. Wagner, B. Mozafari

REFERENCES

1. Letter from E. C. Simpson (Florida Power Corporation) to H. R. Denton (NRC), 3F1285-05, December 10, 1985.

2. "Criticality Safety Analysis of the Crystal River Pool B Fuel Storage Rack with Fuel of 4% Enrichment," SSA-160, Southern Science Office of Black & Veatch, September 1985.

3. "Criticality Safety Analysis of the New-Fuel Storage Vault With Fuel of 4.5% Enrichment", Southern Science Office of Black & Veatch (undated).

4. Letter from Robert Reid (NRC) to J. A. Hancock (Florida Power Corporation) November 17, 1980.