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 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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