August 23, 1994

Docket No. 50-395

Mr. John L. Skolds
Senior Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
Post Office Box 88
Jenkinsville, South Carolina 29065

Dear Mr. Skolds:

SUBJECT: ISSUANCE OF AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-12 REGARDING INCREASED LIMIT FOR FUEL ENRICHMENT - VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 (TAC NO. M88452)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1 (VCSNS). The amendment consists of changes to the Technical Specifications in response to your application dated December 13, 1993, as supplemented by letters dated February 2, 1994, and March 11, 1994.

The amendment changes the Technical Specifications to allow for the use and subsequent storage of fuel with an enrichment not to exceed a nominal 5.0 weight percent (w/o) U-235.

A copy of the related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Bi-weekly 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

Enclosures:
1. Amendment No. 116 to NPF-12
2. Safety Evaluation

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

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Office of Nuclear Reactor Regulation

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2. Safety Evaluation

cc w/enclosures:
See next page

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AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-12 - SUMMER, UNIT NO. 1

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

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. NPF-12

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

   A. The application for amendment by South Carolina Electric & Gas Company (the licensee), dated December 13, 1993, as supplemented by letter dated March 11, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-12 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 116 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Byron L. Siegel 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: August 23, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 116

TO FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are indicated by marginal lines.

Remove Pages	Insert Pages
3/4 9-14	3/4 9-14
3/4 9-15	3/4 9-15
3/4 9-16	3/4 9-16
5-6	5-6
5-7	5-7
5-7a	5-7a

REFUELING OPERATIONS

3/4.9.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 The combination of initial enrichment and cumulative burnup for spent fuel assemblies stored in Regions 2 and 3 shall be within the acceptable domain of Figure 3.9-1 for Region 2 and Figure 3.9-2 for Region 3.

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

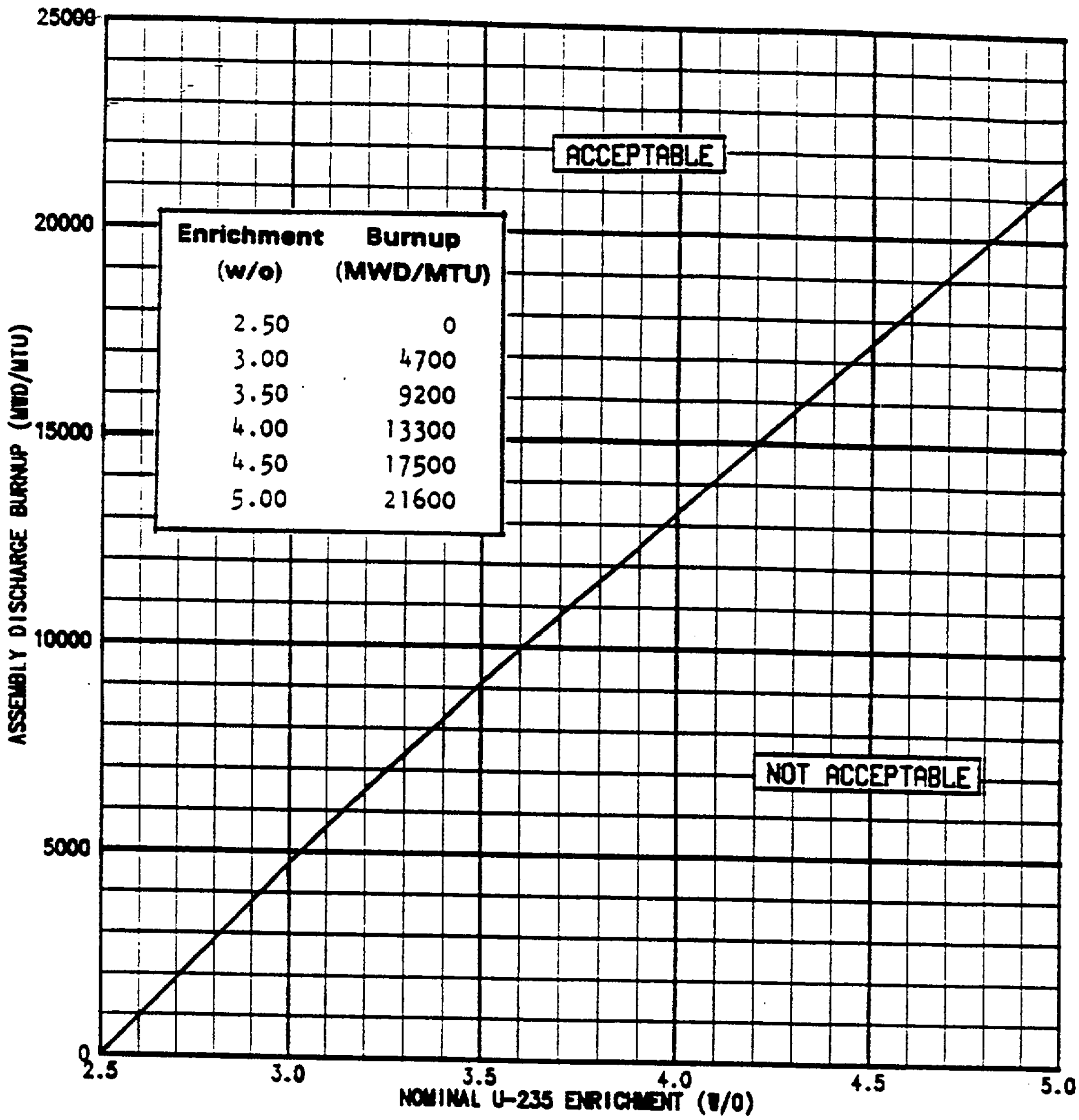
ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable

SURVEILLANCE REQUIREMENTS

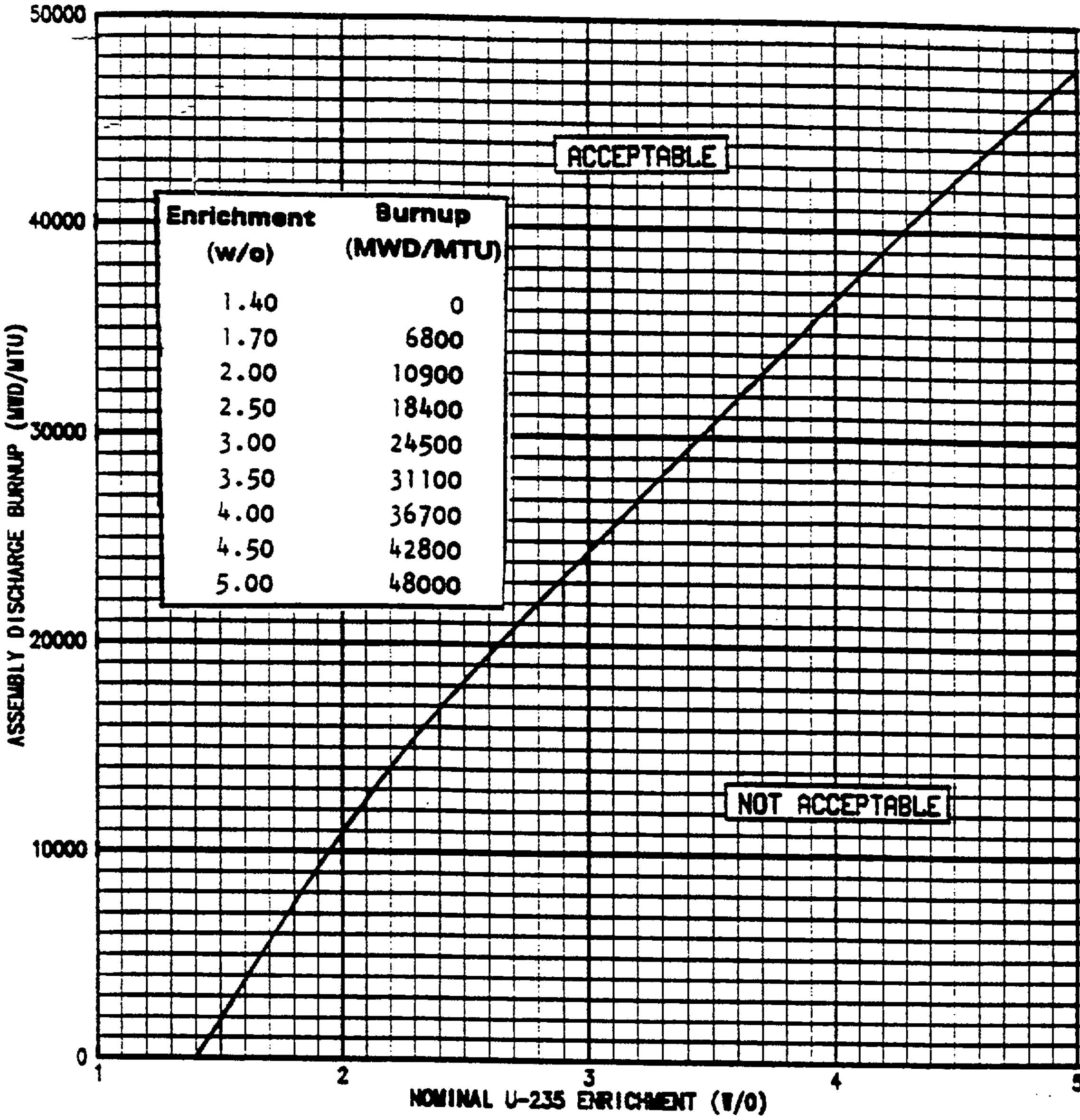
4.9.12 The burnup of each spent fuel assembly stored in Regions 2 and 3 shall be ascertained by careful analysis of its burnup history prior to storage in Region 2 or 3. A complete record of such analysis shall be kept for the time period that the spent fuel assembly remains in Region 2 or 3 of the spent fuel pool.



Enrichment (w/o)	Burnup (MWD/MTU)
2.50	0
3.00	4700
3.50	9200
4.00	13300
4.50	17500
5.00	21600

Note: The use of linear interpolation between the minimum burnups reported above is acceptable.

FIGURE 3.9-1 MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2



Enrichment (w/o)	Burnup (MWD/MTU)
1.40	0
1.70	6800
2.00	10900
2.50	18400
3.00	24500
3.50	31100
4.00	36700
4.50	42800
5.00	48000

Note: The use of linear interpolation between the minimum burnups reported above is acceptable.

FIGURE 3.9-2 MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 3

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies. Each fuel assembly shall consist of 264 Zircaloy-4 or ZIRLO(TM) clad fuel rods with an initial composition of uranium dioxide with a maximum nominal enrichment of 5.0 weight percent U-235 as fuel material. Limited substitutions of Zircaloy-4, ZIRLO(TM) and/or stainless steel filler rods for fuel rods, if justified by a cycle specific reload analysis using an NRC-approved methodology, may be used. Fuel assembly configurations shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or cycle-specific reload analyses to comply with all fuel safety design bases. Reload fuel shall contain sufficient integral fuel burnable absorbers such that the requirements of Specifications 5.6.1.1a.2 and 5.6.1.2 b are met. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 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.

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 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9407 ± 100 cubic feet at a nominal T_{avg} of 586.8°F.

5.5 METEOROLOGICAL TOWER LOCATION

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

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks consist of 1276 individual cells, each of which accommodates a single assembly. The cells are grouped into 3 regions. The spent fuel storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases. This is ensured by maintaining the following for each region:

a. REGION 1 - designated for storage of fresh fuel assemblies and freshly discharged fuel assemblies.

   1. A nominal 10.4025 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A maximum nominal enrichment of 5.0 weight percent U-235 with sufficient integral fuel burnable absorbers such that the maximum reference fuel assembly K_{∞} is less than or equal to 1.460 at 68°F.

b. REGION 2 - designated for storage of discharged fuel assemblies.

   1. A nominal 10.4025 x 10.1875 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A maximum nominal enrichment of 2.5 weight percent U-235 with no burnup and up to 5.0 weight percent U-235 with a minimum burnup of up to 21,600 MWD/MTU, as specified in Figure 3.9-1.

c. REGION 3 - designated for storage of discharged fuel assemblies.

   1. A nominal 10.116 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A maximum nominal enrichment of 1.4 weight percent U-235 with no burnup and up to 5.0 weight percent U-235 with a minimum burnup of up to 48,000 MWD/MTU, as specified in Figure 3.9-2.

5.6.1.2 The new fuel storage racks consist of 60 individual cells, each of which accommodates a single assembly. The new fuel pit storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 for low density optimum moderation conditions, including conservative allowances for uncertainties and biases. This is ensured by maintaining:

a. A nominal 21 inch center-to-center distance between new fuel assemblies placed in the storage rack.

b. A maximum nominal enrichment of 5.0 weight percent U-235 with sufficient integral fuel burnable absorbers such that the maximum reference fuel assembly K_{∞} is less than or equal to 1.460 at 68°F.

DRAINAGE

5.6.2 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 460'3".

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1276 fuel assemblies, 242 in Region 1, 99 in Region 2, and 935 in Region 3.

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 limits of Table 5.7-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By letter dated December 13, 1993, as supplemented by letters dated February 2, 1994, and March 11, 1994, South Carolina Electric & Gas Company (SCE&G or the licensee) requested changes to the Virgil C. Summer Nuclear Station, Unit No. 1, (VCSNS or Summer) Technical Specifications (TS) to allow the use and subsequent storage of fuel initially enriched to 5 weight percent (w/o) Uranium 235 (U-235). The March 11, 1994, letter provided clarifying information that did not change the initial determination of no significant hazards consideration as published in the FEDERAL REGISTER.

2.0 EVALUATION

The analysis of the reactivity effects of fuel storage in the spent fuel storage racks was performed with the three-dimensional multi-group Monte Carlo computer code, KENO Va, using neutron cross sections generated by the AMPX code package from the 227 energy group ENDF/B-V data library. Since the KENO Va code package does not have depletion capability, burnup analyses were performed with the two-dimensional transport theory code, PHOENIX, using a 25 energy group nuclear data library based on a modified version of the British WIMS cross section library. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the VCSNS fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber thickness. The intercomparison between two independent methods of analysis (KENO-5a and PHOENIX) also provides an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a reactivity calculations, a minimum of 60,000 neutron histories were accumulated in each calculation. Experience has shown that this number of histories is quite sufficient to assure convergence of KENO Va reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the VCSNS storage racks with a high degree of confidence.

The spent fuel storage racks in Region 1 were reevaluated for 4.0 w/o U-235 enriched fuel based on the as-built boron-10 (B-10) loading in the Boraflex panels supplied by the Boraflex material vendor. The calculations were made for pure water moderator at 20° C with a density of 1.0 gm/cc. For the nominal storage cell design in Region 1, uncertainties due to tolerances in fuel enrichment and density, fuel pellet dishing, storage cell I.D., cell lattice spacing, stainless steel thickness, Boraflex width, thickness, and length, and B-10 loading were accounted for as well as eccentric fuel positioning. These uncertainties were appropriately determined at the 95/95 probability/confidence level. In addition, calculational and methodology biases and uncertainties due to benchmarking, B-10 self shielding, and pool water temperature ranges were included as well as consideration of Boraflex gaps and shrinkage. The calculations assume that 75% of the Boraflex panels experience non-uniform shrinkage (random gaps) and the remaining 25% of the panels experience uniform shrinkage (pull-back) from the bottom. Based on the results of recent blackness testing performed in the Region 1 racks, the staff concurs that these assumptions, in conjunction with the assumption of a 4% width and length shrinkage, bound the current measured data and future development of additional shrinkage and gaps. The final Region 1 design, when fully loaded with fuel enriched to 4.0 w/o U-235, resulted in a k_{eff} of 0.9485 when combined with all known uncertainties. This meets the staff's criterion of k_{eff} no greater than 0.95 including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

To enable the storage of fuel assemblies with nominal enrichments greater than 4.0 w/o U-235, the concept of reactivity equivalencing was used. In this technique, which has been previously approved by the NRC, credit is taken for the reactivity decrease due to the integral fuel burnable absorber (IFBA) material coated on the outside of the UO_2 pellet. The fuel assembly depletion calculations performed show that the maximum reactivity for rack geometry occurs at 0 burnup. Based on these calculations, the reactivity of the fuel rack array when filled with fuel assemblies enriched to 5.0 w/o U-235 with each containing 80 IFBA rods was found to be equivalent to the reactivity of the rack when filled with fuel assemblies enriched to 4.0 w/o and containing no IFBAs.

Since the worth of individual IFBA rods can change depending on position within the assemblies due to local variations in thermal neutron flux, the licensee has included a conservative reactivity margin to assure that the IFBA requirement remains valid at intermediate enrichments where standard IFBA patterns may not be available. In addition, to account for calculational uncertainties, the IFBA requirements also include a conservatism of approximately 10% on the total number of IFBA rods at the 5.0 w/o enrichment limit (i.e., about 8 extra IFBA rods for a 5.0 w/o fuel assembly). The staff concludes that sufficient conservatism has been incorporated to bound the calculational assumption that the IFBA requirements were based on the standard IFBA patterns used by Westinghouse.

As an alternative method for determining the acceptability of fuel storage in Region 1, the infinite multiplication factor, k_∞, is used as a reference reactivity point. The PHOENIX code was used for the fuel assembly k_∞ calculations based on a unit assembly configuration in the VCSNS core geometry moderated by pure water at a temperature of 68° F with a density of 1.0 gm/cc. A 1% reactivity bias was included to account for calculational uncertainties.

Calculations for a fresh 4.0 w/o Westinghouse 17x17 OFA fuel assembly, which yields equivalent or bounding reactivity results relative to the other Westinghouse 17x17 fuel types, in VCSNS core geometry resulted in a reference k_∞ of 1.460. Since the fuel rack reactivity of a fresh 4.0 w/o assembly is less than 0.95 and has been shown to be equivalent to a 5.0 w/o assembly with the standard number of IFBA rods, an assembly of maximum nominal enrichment of 5.0 w/o U-235 with a maximum reference k_∞ less than or equal to 1.460 at 68° F can be safely stored in the Region 1 racks.

The Region 2 spent fuel storage racks were reanalyzed for storage of Westinghouse 17x17 fuel assemblies with nominal enrichments up to 2.5 w/o U-235. The same initial assumptions, biases, and uncertainties, as used for the Region 1 analyses, were included. Since the blackness testing performed in the Region 2 racks did not indicate any gaps in any of the Boraflex panels inspected, no gaps were assumed in the analysis. However, a 4% total width and length shrinkage was assumed in every Boraflex panel with the placement of the entire 4% of length shrinkage at the bottom of every panel. Calculations performed for storage racks similar to VCSNS have indicated that positioning all of the Boraflex shrinkage at the bottom results in the most conservative k_{eff}. The maximum k_{eff} for Region 2 is 0.9442, within the NRC acceptance criterion of 0.95.

To enable the storage of fuel assemblies initially enriched to greater than 2.5 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. For burnup credit, a series of reactivity calculations are performed to generate a set of initial enrichment-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. This is shown in Figure 3.9-1 in which a fresh 2.5 w/o enriched fuel assembly yields the same rack reactivity as an initially enriched 5.0 w/o assembly depleted to 21,600 MWD/MTU. This curve includes a reactivity uncertainty of 0.0072 due to depletion calculations.

Region 3 has been analyzed for the storage of Westinghouse 17x17 fuel assemblies with nominal enrichments up to 1.4 w/o U-235. The same initial assumptions, biases and uncertainties used in the Region 2 analyses were also used for Region 3 except for the Boraflex related uncertainties. The Region 3 racks do not contain Boraflex. The maximum k_{eff} for Region 3 is 0.9441, within the NRC acceptance criterion of 0.95.

As for Region 2, burnup credit reactivity equivalencing was used to allow storage of fuel assemblies with initial enrichments greater than 1.4 w/o U-235. Figure 3.9-2 shows that fresh 1.4 w/o enriched fuel is equivalent to initially enriched 5.0 w/o fuel which has achieved a burnup of 48,000 MWD/MTU. The curve includes a reactivity uncertainty of 0.0160 due to depletion calculations.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as heatup or cooldown events or the misloading of an assembly with a burnup and enrichment combination outside of the acceptable area in Figure 3.9-1 or 3.9-2, which could lead to an increase in reactivity. For such events, credit may be taken for the presence of approximately 2000 ppm of boron in the pool water required during fuel handling operations since the staff 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. In fact, the licensee has determined that only 400 ppm of boron is necessary to mitigate the worst postulated accident in any pool region. Therefore, the staff criterion of k_{eff} no greater than 0.95 for any postulated accident is met.

The new (fresh) fuel racks have been previously analyzed for storage of Westinghouse 17x17 fuel assemblies with enrichments up to 5.0 w/o U-235 and the criticality analysis was included with this amendment request. For the fully flooded condition, k_{eff} did not exceed 0.95, including appropriate allowances for biases and uncertainties. For the low density optimum moderation condition, k_{eff} did not exceed 0.98. Therefore, the criticality analyses of the fresh fuel racks meet the applicable NRC criteria and are acceptable. No credit for IFBAs was included in these analyses. However, due to the restrictions required on spent fuel storage, the proposed TS changes require fuel assemblies with enrichments above 4.0 w/o U-235 to contain IFBAs such that the maximum reference fuel k_∞ is no greater than 1.460 in unborated water at 68° F.

The following Technical Specification changes have been proposed as a result of the requested enrichment increase. The staff finds that these changes are consistent with the above evaluation and, therefore, are acceptable.

(1) Figure 3.9-1 has been revised to place restrictions on fuel burnup as a function of initial enrichment up to 5.0 w/o U-235 and to account for the effects of Boraflex panel shrinkage and gaps in Region 2 of the spent fuel pool.

(2) Figure 3.9-2 has been revised to place restrictions on fuel burnup as a function of initial enrichment up to 5.0 w/o U-235 in Region 3 of the spent fuel pool. Region 3 does not contain Boraflex.

(3) TS 5.3.1 has been revised to permit reload fuel with a maximum enrichment of 5.0 w/o U-235 and to incorporate the recommendations described in NRC Generic Letter GL 90-02, Supplement 1. It also requires fuel to contain sufficient IFBAs in order to comply with the requirements of TS 5.6.1.1.a-2.

(4) TS 5.6.1.1 has been revised to delineate the requirements for each region of the spent fuel pool, to add the new minimum burnups as a function of initial enrichment, to permit the storage of 5.0 w/o U-235 fuel, and to add the requirements for k_∞.

(5) TS 5.6.1.2 has been revised to permit the storage of 5.0 w/o U-235 fuel in the new fuel storage racks, remove the reference to Section 4.3 of the FSAR, and to add the requirements for k_∞.

Based on the review described above, the staff finds the criticality aspects of the proposed enrichment increase to the VCSNS new and spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The staff concludes that Westinghouse 17x17 fuel from VCSNS may be safely stored in Region 1 of the spent fuel pool provided that the U-235 enrichment does not exceed 5.0 w/o and there are sufficient IFBAs such that the maximum reference fuel assembly k_∞ does not exceed 1.460 at 68° F. Any of these fuel assemblies may also be stored in Region 2 or 3 of the spent fuel pool provided it meets the burnup and enrichment limits specified in TS Figure 3.9-1 or 3.9-2, respectively.

Although the VCSNS TS have been modified to specify the above-mentioned fuel as acceptable for storage in the fresh or spent fuel racks, evaluations of reload core designs (using any enrichment) will 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.

Based on the foregoing evaluation, the staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact has been prepared and published in the Federal Register on August 15, 1994, (59 FR 41799). Accordingly, based upon the Environmental Assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 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.

Principal Contributor: L. Kopp

Date: August 23, 1994