June 29, 1998

Mr. C. K. McCoy Vice President Southern Nuclear Operating Company, Inc. Post Office Box 1295 Birmingham, Alabama 35201	DISTRIBUTION Docket File PUBLIC PDII-2 RF JZwolinski OGC	THarris(TLH3) (e-mail SE only) LRPlisco,RII GHill(4) T-5 C3 WBeckner, O-11 F23 ACRS T-2 E26 PSkinner, RII

SUBJECT: ISSUANCE OF AMENDMENTS - VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (TAC NOS. MA0152 and MA0153)

Dear Mr. McCoy:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 102 to Facility Operating License NPF-68 and Amendment No. 80 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated September 4, 1997 as supplemented by letters dated November 20, 1997, and May 19 and June 12, 1998.

The amendment changes the common VEGP TS to allow an increase in the Unit 1 spent fuel storage capacity from 288 to 1476 fuel assemblies.

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

Sincerely,

ORIGINAL SIGNED BY:
David Jaffe, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

DF01

Enclosures:
1. Amendment No. 102 to NPF-68
2. Amendment No. 80 to NPF-81
3. Safety Evaluation

Tech Editor review of SE 6/12/98

cc w/encl: See next page
DOCUMENT NAME: A:\LAMA0152.WP

OFFICE	PDII-2/PM	PDII-2/LA	OGC	PDII-2/D
NAME	D.JAFFE	L.BERRY	APH	H.BERKOW
DATE	6/10/98	6/10/98	6/16/98	6/26/98
COPY	YES NO	YES NO	YES NO	YES NO

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

June 29, 1998

Mr. C. K. McCoy
Vice President
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

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cc w/encl: See next page

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P. O. Box 1295
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 102
License No. NPF-68

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

A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Facility Operating License No. NPF-68 filed by the Georgia Power Company and Southern Nuclear Operating Company, Inc. (Southern Nuclear), acting for themselves, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated September 4, 1997, as supplemented by letters dated November 20, 1997, May 19 and June 12, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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 hereby amended by page 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-68 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 102 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   In addition, paragraph 2.C.(10) to Facility Operating License No. NPF-68 is hereby amended to read as follows:

   (10) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 102 , are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented on a schedule consistent with the receipt and storage of new fuel in the fall of 1998, for the spring 1999 refueling outage of Vogtle Unit 1.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix D Changes
2. Technical Specification Changes

Date of Issuance: June 29, 1998

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 80
License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Facility Operating License No. NPF-81 filed by the Georgia Power Company and Southern Nuclear Operating Company, Inc. (Southern Nuclear), acting for themselves, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the licensees), dated September 4, 1997, as supplemented by letters dated November 20, 1997, May 19 and June 12, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-81 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 80 , and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   In addition, paragraph 2.C.(3) to Facility Operating License No. NPF-81 is hereby amended to read as follows:

   (3) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 80 , are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented on a schedule consistent with the receipt and storage of new fuel in the fall of 1998 for the spring 1999 refueling outage of Unit 1.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix D Changes
2. Technical Specification Changes

Date of Issuance: June 29, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 102

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 80

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace Appendix D of Facility Operating License Nos. NPF-68 and NPF-81 with the enclosed revised Appendix D.

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

Remove	Insert
viii	viii
ix	ix
3.7-40	3.7-40
3.7-41	3.3-41
3.7-42	3.7-42
4.0-2	4.0-2
4.0-3	4.0-3
4.0-3a	4.0-3a
4.0-3b	4.0-3b
4.0-4	4.0-4
4.0-7	4.0-7
4.0-9	4.0-9
4.0-10	4.0-10
B 3.7-92	B 3.7-92
B 3.7-93	B 3.7-93
B 3.7-94	B 3-7-94
B 3.7-95	B 3.7-95
B 3.7-97	B 3.7-97
B 3.7-98	B 3.7-98
B 3.7-99	B 3.7-99

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-68

Amendment Number	Additional Condition	Implementation Date
100	The licensee shall implement a procedure that will prohibit entry into an extended Emergency Diesel Generator Allowed Outage Time (14 days), for scheduled maintenance purposes, if severe weather conditions are expected, as described in the licensee's application dated January 22, 1998, as supplemented by letter dated March 18, 1998, and evaluated in the staff's Safety Evaluation dated May 20, 1998.	Prior to implementation of Amendment No. 100
102	The spent fuel pool heat loads will be managed by administrative controls. These controls will be placed in applicable procedures as described in the licensee's letters dated September 4, 1997, May 19 and June 12, 1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998.	Before transferring irradiated fuel into the Unit 1 spent fuel pool
102	The UFSAR will be updated to include the heat load that will ensure the temperature limit of 170°F will not be exceeded, as well as the requirement to perform a heat load evaluation before transferring irradiated fuel to either pool, as described in the licensee's letters dated September 4, 1997, May 19 and June 12, 1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998.	To be included in the next appropriate UFSAR update following the installation of the Unit 1 spent fuel racks
102	A temporary gantry crane, with a hoist rated for 20 tons, will be erected on the existing fuel handling bridge rails to move the racks within the spent fuel pool area, as described in the licensee's letters dated September 4, 1997, May 19 and June 12, 1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998.	Before commencing reracking operations

Amendment Number	Additional Condition	Implementation Date
102	The licensee will implement all applicable crane, load path and height, rigging and load testing guidelines of NUREG-0612 and ANSI Standard B30.2, as described in the licensee's letters dated September 4, 1997, May 19 and June 12, 1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998.	Before and during reracking operations, as appropriate

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-81

Amendment Number	Additional Condition	Implementation Date
78	The licensee shall implement a procedure that will prohibit entry into an extended Emergency Diesel Generator Allowed Outage Time (14 days), for scheduled maintenance purposes, if severe weather conditions are expected, as described in the licensee's application dated January 22, 1998, as supplemented by letter dated March 18, 1998, and evaluated in the staff's Safety Evaluation dated May 20, 1998.	Prior to implementation of Amendment No. 78
80	The UFSAR will be updated to include the heat load that will ensure the temperature limit of 170°F will not be exceeded, as well as the requirement to perform a heat load evaluation before transferring irradiated fuel to either pool, as described in the licensee's letters dated September 4, 1997, May 19 and June 12, 1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998.	To be included in the next appropriate UFSAR update following the installation of the Unit 1 spent fuel racks

LIST OF TABLES

LIST OF FIGURES

3.7 PLANT SYSTEMS

3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

LCO 3.7.18 The combination of initial enrichment burnup and configuration of fuel assemblies stored in the fuel storage pool shall be within the Acceptable Burnup Domain of Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or in accordance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage 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 storage location.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.18.1	Verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly is in accordance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).	Prior to storing the fuel assembly in the fuel storage pool location.

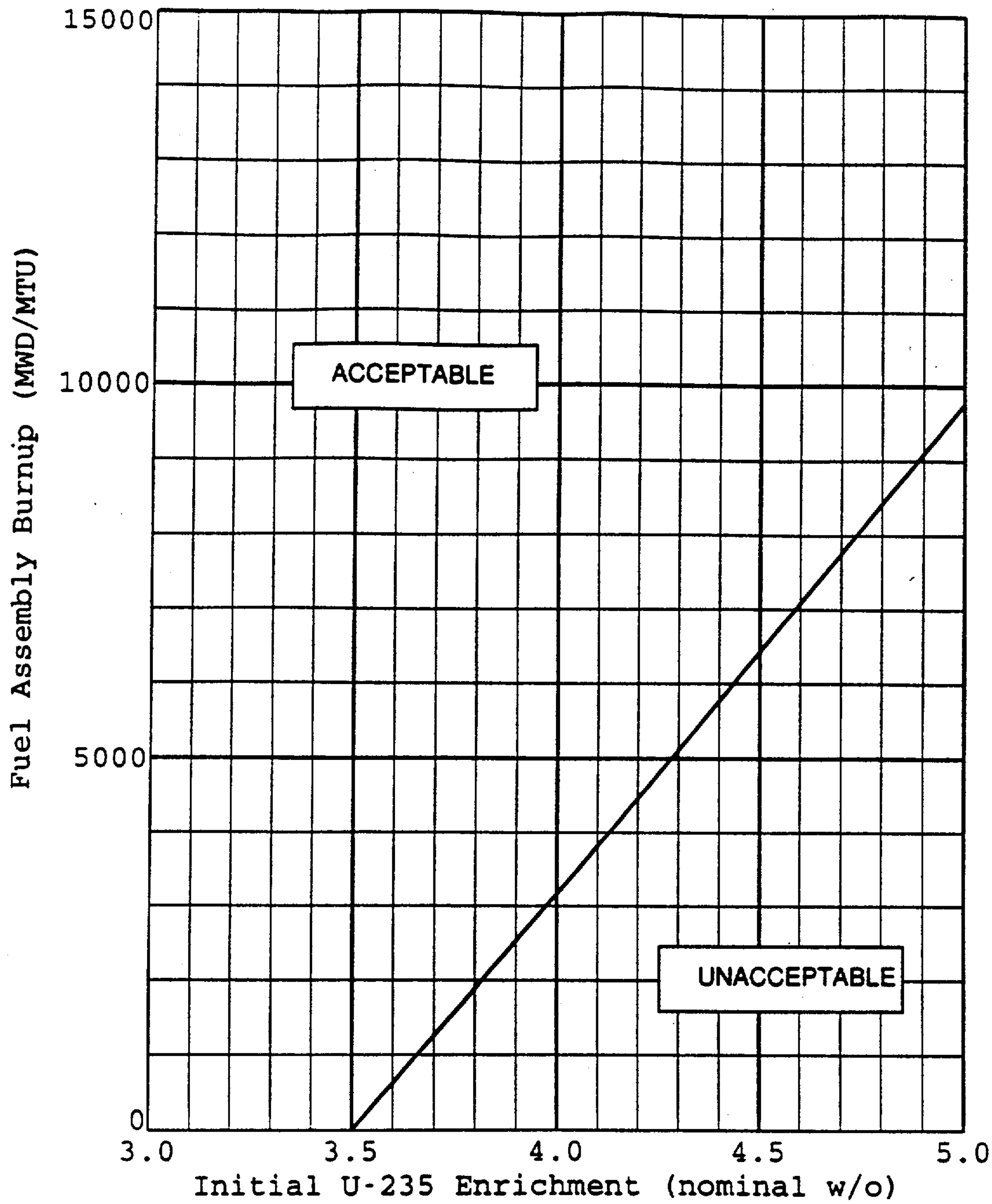


Figure 3.7.18-1 Vogtle Unit 1 Burnup Credit Requirements for All Cell Storage

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

(Unit 1) 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. $K_{eff} < 1.0$ when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. $K_{eff} \leq 0.95$ when fully flooded with water borated to 600 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-1 or having a maximum reference fuel assembly $K\infty$ less than or equal to 1.431 at 68°F may be allowed unrestricted storage in the Unit 1 fuel storage pool.

e. New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 1 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-4.

Interfaces between storage configurations in the Unit 1 fuel storage pool shall be in compliance with Figure 4.3.1-6. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-1, or which have a maximum reference fuel assembly $K\infty$ less than or equal to 1.431 at 68°F. "B" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

f. A nominal 10.25 inch center to center pitch in the Unit 1 high density fuel storage racks.

(Unit 2) 4.3.1.2 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. $K_{eff} < 1.0$ when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. $K_{eff} \leq 0.95$ when fully flooded with water borated to 500 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2 may be allowed unrestricted storage in the Unit 2 fuel storage pool.

e. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-2 may be stored in the Unit 2 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-4.

New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 2 fuel storage pool in a 2-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-4.

New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-3 may be stored in the Unit 2 fuel

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

storage pool as "low enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-5. New or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or having a maximum reference fuel assembly K_∞ less than or equal to 1.410 at 68°F may be stored in the Unit 2 fuel storage pool as "high enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-5.

Interfaces between storage configurations in the Unit 2 fuel storage pool shall be in compliance with Figures 4.3.1-6, 4.3.1-7, 4.3.1-8, and 4.3.1-9. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2. "B" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-2. "C" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235. "L" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-3. "H" assemblies are new or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or having a maximum reference fuel assembly K_∞ less than or equal to 1.410 at 68°F.

f. A nominal 10.58-inch center to center pitch in the north-south direction and a nominal 10.4-inch center to center pitch in the east-west direction in the Unit 2 high density fuel storage racks.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

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

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

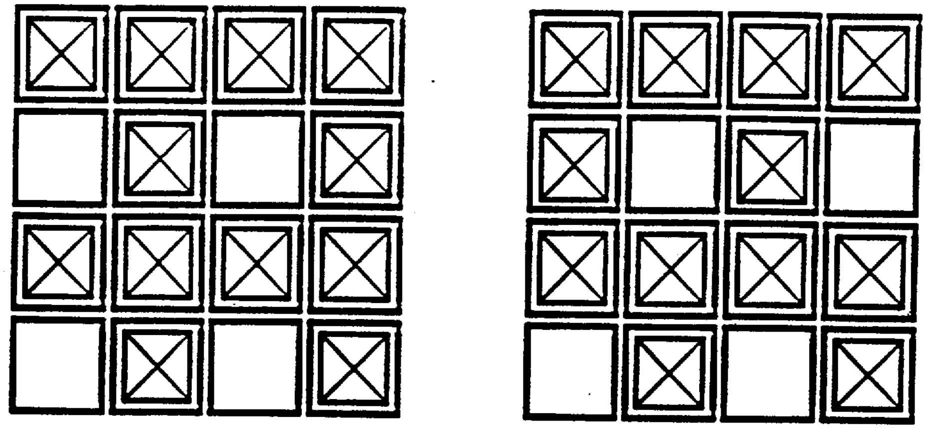
The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 194 foot-1 1/2 inch.

4.3.3 Capacity

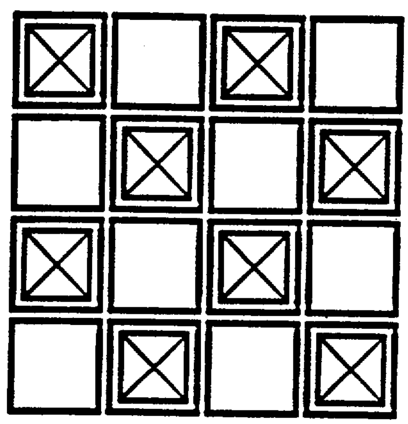
The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1476 fuel assemblies in the Unit 1 storage pool and no more than 2098 fuel assemblies in the Unit 2 storage pool.

(This figure has been deleted.)

Figure 4.3.1-1 Vogtle Unit 1 Burnup Credit Requirements for 3-out-of-4 Storage



3-out-of-4 Checkerboard Storage (Units 1 and 2)



2-out-of-4 Checkerboard Storage (Unit 2)

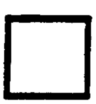
 Empty Storage Cell  Fuel Assembly in Storage Cell

Figure 4.3.1-4 Vogtle Units 1 and 2 Empty Cell Checkerboard Storage Configurations

Interface

A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
Empty	B	Empty	A	A	A
B	B	B	A	A	A
Empty	B	Empty	A	A	A

Note:
A = All Cell Enrichment
B = 3-Out-Of-4 Enrichment
Empty = Empty Cell

Boundary Between All Cell Storage and 3-out-of-4 Storage (Units 1 and 2)

Interface

A	A	A	A	A	A
A	A	A	A	A	A
A	A	A	A	A	A
Empty	B	Empty	A	A	A
C	Empty	B	A	A	A
Empty	C	Empty	A	A	A

Note:
A = All Cell Enrichment
B = 3-Out-Of-4 Enrichment
C = 2-Out-Of-4 Enrichment
Empty = Empty Cell

Boundary Between All Cell Storage and 2-out-of-4 Storage (Unit 2)

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assembly with an empty cell.

Figure 4.3.1-6 Vogtle Units 1 and 2 Interface Requirements (All Cell to Checkerboard Storage)

Interface

B	Empty	B	Empty	B	Empty
B	B	B	B	B	B
B	Empty	B	Empty	B	Empty
Empty	C	Empty	B	B	B
C	Empty	C	Empty	B	Empty
Empty	C	Empty	B	B	B

Note:
B = 3-Out-Of-4 Enrichment
C = 2-Out-Of-4 Enrichment
Empty = Empty Cell

Boundary Between 2-out-of-4 Storage and 3-out-of-4 Storage

Interface

Empty	B	Empty	B	B	B
B	B	B	B	Empty	B
Empty	B	Empty	B	B	B
C	Empty	C	Empty	B	Empty
Empty	C	Empty	B	B	B
C	Empty	C	Empty	B	Empty

Note:
B = 3-Out-Of-4 Enrichment
C = 2-Out-Of-4 Enrichment
Empty = Empty Cell

Boundary Between 2-out-of-4 Storage and 3-out-of-4 Storage

Note:
1. A row of empty cells can be used at the interface to separate the configurations.
2. It is acceptable to replace an assembly with an empty cell.

Figure 4.3.1-7 Vogtle Unit 2 Interface Requirements (Checkerboard Storage Interface)

B 3.7 PLANT SYSTEMS

B 3.7.17 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

Fuel assemblies are stored in high density racks. The Unit 1 spent fuel storage racks contain storage locations for 1476 fuel assemblies, and the Unit 2 spent fuel storage racks contain storage locations for 2098 fuel assemblies. The Unit 1 racks use boral as a neutron absorber in a flux trap design. The Unit 2 racks contain Boraflex, however, no credit is taken for Boraflex. Westinghouse 17x17 fuel assemblies with initial enrichments of up to and including 5.0 weight percent U-235 can be stored in any location in the Unit 1 or Unit 2 fuel storage pool provided the fuel burnup-enrichment combinations are within the limits that are specified in Figures 3.7.18-1 (Unit 1) or 3.7.18-2 (Unit 2) of the Technical Specifications. Fuel assemblies that do not meet the burnup-enrichment combination of Figures 3.7.18-1 or 3.7.18-2 may be stored in the storage pools of Units 1 or 2 in accordance with checkerboard storage configurations described in Figures 4.3.1-2 through 4.3.1-9. The acceptable fuel assembly storage configurations are based on the Westinghouse Spent Fuel Rack Criticality Methodology, described in WCAP-14416-NP-A, Rev. 1, (Reference 4). This methodology includes computer code benchmarking, spent fuel rack criticality calculations methodology, reactivity equivalencing methodology, accident methodology, and soluble boron credit methodology.

The Westinghouse Spent Fuel Rack Criticality Methodology ensures that the multiplication factor, K_{eff}, of the fuel and spent fuel storage racks is less than or equal to 0.95 as recommended by ANSI 57.2-1983 (Reference 3) and NRC guidance (References 1, 2 and 6). The codes, methods, and techniques contained in the methodology are used to satisfy this criterion on K_{eff}.

The methodology of the NITAWL-II, XSDRNPM-S, and KENO-Va codes is used to establish the bias and bias uncertainty. PHOENIX-P, a nuclear design code used primarily for core reactor physics calculations is used to simulate spent fuel storage rack geometries.

(continued)

BASES

BACKGROUND (continued)

Reference 4 describes how credit for fuel storage pool soluble boron is used under normal storage configuration conditions. The storage configuration is defined using K_{eff} calculations to ensure that the K_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain K_{eff} less than or equal to 0.95. The Unit 1 pool requires 600 ppm and the Unit 2 pool requires 500 ppm to maintain K_{eff} less than or equal to 0.95 for all allowed combinations of storage configurations, enrichments, and burnups. The analyses assumed 19.9% of the boron atoms have atomic weight 10 (B-10). The effects of B-10 depletion on the boron concentration for maintaining $K_{eff} \leq 0.95$ are negligible. The treatment of reactivity equivalencing uncertainties, as well as the calculation of postulated accidents crediting soluble boron is described in WCAP-14416-NP-A, Rev. 1.

This methodology was used to evaluate the storage of fuel with initial enrichments up to and including 5.0 weight percent U-235 in the Vogtle fuel storage pools. The resulting enrichment, and burnup limits for the Unit 1 and Unit 2 pools, respectively, are shown in Figures 3.7.18-1 and 3.7.18-2. Checkerboard storage configurations are defined to allow storage of fuel that is not within the acceptable burnup domain of Figures 3.7.18-1 and 3.7.18-2. These storage requirements are shown in Figures 4.3.1-2 through 4.3.1-9. A boron concentration of 2000 ppm assures that no credible dilution event will result in a K_{eff} of > 0.95.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in an increase in K_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly between rack modules, or between rack modules and the pool wall.

From a criticality standpoint, a dropped assembly accident occurs when a fuel assembly in its most reactive condition is dropped onto the storage racks. The rack structure from a criticality standpoint is not excessively deformed. Previous accident analysis with unborated water showed that the dropped assembly which comes to rest horizontally on top of the rack has sufficient water separating it from the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

active fuel height of stored assemblies to preclude neutronic interaction. For the borated water condition, the interaction is even less since the water contains boron, an additional thermal neutron absorber.

However, three accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. The first postulated accident would be a change in pool temperature to outside the range of temperatures assumed in the criticality analyses (50°F to 185°F). The second accident would be dropping a fuel assembly into an already loaded cell. The third would be the misloading of a fuel assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied.

An increase in the temperature of the water passing through the stored fuel assemblies causes a decrease in water density which results in an addition of negative reactivity for flux trap design racks such as the Unit 1 racks. However, since Boraflex is not considered to be present for the Unit 2 racks and the fuel storage pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. The reactivity effects of a temperature range from 32°F to 240°F were evaluated. The increase in reactivity due to the increase in temperature is bounded by the misload accident, for the Unit 2 racks. The increase in reactivity due to the decrease in temperature below 50°F is bounded by the misplacement of a fuel assembly between the rack and pool walls for the Unit 1 racks.

For the accident of dropping a fuel assembly into an already loaded cell, the upward axial leakage of that cell will be reduced, however, the overall effect on the rack reactivity will be insignificant. This is because the total axial leakage in both the upward and downward directions for the entire fuel array is worth about 0.003 Δk. Thus, minimizing the upward-only leakage of just a single cell will not cause any significant increase in reactivity. Furthermore, the neutronic coupling between the dropped assembly and the already loaded assembly will be low due to several inches of assembly nozzle structure which would separate the active fuel regions. Therefore, this accident would be bounded by the misload accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel assembly misloading accident involves placement of a fuel assembly in a location for which it does not meet the requirements for enrichment or burnup, including the placement of an assembly in a location that is required to be left empty. The result of the misloading is to add positive reactivity, increasing K_{eff} toward 0.95. A fourth accident was evaluated for the Unit 1 fuel storage racks containing boral. The fourth accident was the misplacement of a fuel assembly between the rack and pool wall. This was the limiting accident for the Unit 1 racks. The

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BASES

APPLICABLE SAFETY ANALYSES (continued)

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BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)

maximum required additional boron to compensate for this event is 1250 ppm for Unit 2, and 800 ppm for Unit 1 which is well below the limit of 2000 ppm.

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

LCO

The fuel storage pool boron concentration is required to be $\geq$ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in reference 5. The amount of soluble boron required to offset each of the above postulated accidents was evaluated for all of the proposed storage configurations. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than or equal to 0.95 should pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 1250 ppm for Unit 2 and 800 ppm for Unit 1. The specified minimum boron concentration of 2000 ppm assures that the concentration will remain above these values. In addition, the boron concentration is consistent with the boron dilution evaluation that demonstrated that any credible dilution event could be terminated prior to reaching the boron concentration for a K_{eff} of > 0.95. These values are 600 ppm for Unit 1 and 500 ppm for Unit 2.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

BASES

BACKGROUND

The Unit 1 spent fuel storage racks contain storage locations for 1476 fuel assemblies, and the Unit 2 spent fuel storage racks contain storage locations for 2098 fuel assemblies.

Westinghouse 17X17 fuel assemblies with an enrichment of up to and including 5.0 weight percent U-235 can be stored in the acceptable storage configurations that are specified in Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), and 4.3.1-2 through 4.3.1-9. The acceptable fuel assembly storage locations are based on the Westinghouse Spent Fuel Rack Criticality Methodology, described in WCAP-14416-NP-A, Rev. 1 (reference 1). Additional background discussion can be found in B 3.7.17.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 3.50 w/o^{235}U may be stored in all storage cell locations of the Unit 1 pool. Fuel assemblies with initial nominal enrichment greater than 3.50 w/o^{235}U must satisfy a minimum burnup requirement as shown in Figure 3.7.18-1. Fuel assemblies having a K_∞ of 1.431 at cold reactor core conditions may also be stored in all cells of the Unit 1 fuel storage racks.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 5.0 w/o^{235}U may be stored in a 3-out-of-4 checkerboard arrangement with empty cells in the Unit 1 pool. There are no minimum burnup requirements for this configuration.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater that 5.0 w/o^{235}U may be stored in a 2-out-of-4 checkerboard arrangement with empty cells in the Unit 2 pool. There are no minimum burnup requirements for this configuration.

(continued)

BASES

BACKGROUND (continued)

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 1.77 w/o ^{235}U may be stored in all storage cell locations of the Unit 2 pool. Fuel assemblies with initial nominal enrichment greater than 1.77 w/o ^{235}U must satisfy a minimum burnup requirement as shown in Figure 3.7.18-2.

(continued)

BASES

BACKGROUND
(continued)

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BASES

BACKGROUND (continued)

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 2.40 w/o ^{235}U may be stored in a 3-out-of-4 checkerboard arrangement with empty cells in the Unit 2 pool. Fuel assemblies with initial nominal enrichment greater than 2.40 w/o ^{235}U must satisfy a minimum burnup requirement as shown in Figure 4.3.1-2.

Westinghouse 17x17 fuel assemblies may be stored in the Unit 2 pool in a 3x3 array. The center assembly must have an initial enrichment no greater than 3.20 w/o ^{235}U. Alternatively, the center of the 3x3 array may be loaded with any assembly which meets a maximum infinite multiplication factor (K_∞) value of 1.410 at 68°F. One method of achieving this value of K_∞ is by the use of IFBAs. The surrounding fuel assemblies must have an initial nominal enrichment no greater than 1.48 w/o ^{235}U or satisfy a minimum burnup requirement for higher initial enrichments as shown in Figure 4.3.1-3.

APPLICABLE SAFETY ANALYSIS

Most fuel storage pool accident conditions will not result in an increase in K_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack and the drop of a fuel assembly between rack modules or between rack modules and the pool wall. However, accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. A discussion of these accidents is contained in B 3.7.17.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the fuel storage pool ensure the K_{eff} of the fuel storage pool will always remain < 0.95, assuming the pool to be flooded with borated water.

The combination of initial enrichment and burnup are specified in Figures 3.7.18-1 and 3.7.18-2 for all cell storage in the Unit 1 and Unit 2 pools, respectively. Other acceptable enrichment burnup and checkerboard combinations are described in Figures 4.3.1-2 through 4.3.1-9.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the fuel storage pool is not in accordance with the acceptable combination of initial enrichment, burnup, and storage configurations, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is within the acceptable burnup domain of Figures 3.7.18-1 (Unit 1) or 3.7.18-2 (Unit 2). For fuel assemblies in the unacceptable range of Figures 3.7.18-1 and 3.7.18-2, performance of this SR will also ensure compliance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

Fuel assembly movement will be in accordance with preapproved plans that are consistent with the specified fuel enrichment, burnup, and storage configurations. These plans are administratively verified prior to fuel movement. Each assembly is verified by visual inspection to be in accordance with the preapproved plan prior to storage in the fuel storage pool. Storage commences following unlatching of the fuel assembly in the fuel storage pool.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 102 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 80 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC., ET AL.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter application dated September 4, 1997, as supplemented by letters dated November 20, 1997, and May 19 and June 12, 1998, Southern Nuclear Operating Company, Inc., et al. (the licensee) proposed license amendments to change the Technical Specifications (TS) for Vogtle Electric Generating Plant (VEGP), Units 1 and 2. The proposed amendments would change the VEGP Units 1 and 2 TS to allow an increase in the Unit 1 spent fuel storage capacity from 288 to 1476 fuel assemblies.

The supplements dated May 19 and June 12, 1998, provided clarifying information that did not change the scope of the September 4, 1997, application and the initially proposed determination of no significant hazards consideration.

2.0 BACKGROUND

The VEGP Units 1 and 2 spent fuel pools (SFPs) are located in a common area that can accept spent fuel from either VEGP Unit 1 or Unit 2. At the present time, VEGP has a fuel storage capacity of 288 assemblies in the Unit 1 SFP and 2098 fuel assemblies in the Unit 2 SFP. The licensee has obtained high-density spent fuel racks, previously utilized at the Maine Yankee Atomic Power Plant (MYAPP). These spent fuel racks were approved for use at MYAPP in an NRC staff letter and accompanying Safety Evaluation (SE) dated June 16, 1982. The spent fuel racks were observed to perform well in service at MYAPP and did not exhibit swelling or any other type of degradation. The only modification performed on the MYAPP spent fuel racks, following issuance of the June 16, 1982, NRC staff SE, was the addition of vent and drain holes in pockets that contain the neutron-absorbing Boral material.

3.0 EVALUATION

The NRC staff's evaluation of the proposed use of the MYAPP spent fuel racks in the VEGP Unit 1 SFP appears in the sections that follow.

3.1 Criticality Analysis

The MYAPP spent fuel storage racks were analyzed for use in the VEGP Unit 1 SFP using the Westinghouse methodology described in "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," Westinghouse Electric Corporation, WCAP-14416-NP-A," Rev. 1, November 1996. The WCAP-14416-NP-A methodology was reviewed and approved by the NRC in an October 25, 1996, letter from T. Collins, NRC, to T. Greene, Westinghouse Owners Group. This methodology takes partial credit for soluble boron in the fuel storage pool criticality analyses and requires conformance with the following NRC acceptance criteria for preventing criticality outside the reactor:

(1) The effective neutron multiplication factor (k_{eff}) shall be less than 1.0 if fully flooded with unborated water, which includes an allowance for uncertainties at a 95 percent probability, 95 percent confidence (95/95) level as described in WCAP-14416-NP-A; and

(2) The effective neutron multiplication factor (k_{eff}) shall be less than or equal to 0.95 if fully flooded with borated water, which includes an allowance for uncertainties at a 95/95 level as described in WCAP-14416-NP-A.

The reactivity effects of VEGP fuel storage in the MYAPP spent fuel racks were analyzed with the three-dimensional Monte Carlo code, KENO-Va, with neutron cross-sections generated with the NITAWL-II and XSDRNPM-S codes. Since the KENO-Va code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with the two-dimensional transport theory code, PHOENIX-P. The analytical methods and models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the MYAPP spent fuel storage racks are designed and have been found to adequately reproduce the critical values. These experimental data are sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions, which include close proximity storage and strong neutron absorbers. The NRC staff concludes that the analytical methods used are acceptable and capable of predicting the reactivity of the MYAPP spent fuel storage racks with a high degree of confidence.

The existing VEGP spent fuel storage racks, which utilize Boraflex as a neutron poison, have previously been qualified for storage of various Westinghouse 17 x 17 fuel assembly types with maximum nominal enrichments up to 5.0 weight percent (w/o) U-235 (enrichment tolerance of ±0.05 w/o U-235). Because of the Boraflex deterioration that has been observed in many spent fuel pools, this criticality analysis for the VEGP spent fuel storage racks neglected the presence of Boraflex to allow storage of all 17 x 17 fuel assemblies with nominal enrichments up to 5.0 w/o U-235 using credit for checker boarding, burnup, burnable absorbers, and soluble boron.

The MYAPP spent fuel storage racks, which utilize Boral neutron-absorbing panels, have been analyzed to allow storage of Westinghouse 17 x 17 fuel assemblies with nominal enrichments up to 5.0 w/o U-235. The analysis takes credit for the presence of Boral absorber panels on all four sides of each spent fuel rack cell.

The moderator was assumed to be pure water at a temperature of 68 °F and a density of 1.0 gm/cc, and the array was assumed to be infinite in lateral extent. Uncertainties due to tolerances in fuel enrichment and density, storage cell inner diameter, storage cell pitch, stainless steel thickness, Boral thickness and width, Boral wrapper thickness and width, assembly position, calculational uncertainty, and methodology bias uncertainty were accounted for. These uncertainties were appropriately determined at the 95/95 probability/confidence level. A methodology bias (determined from benchmark calculations), as well as a reactivity bias to account for the effect of the normal range of SFP water temperatures (50 °F to 185 °F), were included. These biases and uncertainties meet the previously stated NRC requirements and are, therefore, acceptable.

For the MYAPP spent fuel racks, an enrichment of 3.5 w/o U-235 was found to be adequate to maintain k_{eff} less than 1.0 with all cells filled with Westinghouse 17 x 17 fuel assemblies and no soluble boron in the pool water (all-cell configuration). This resulted in a nominal k_{eff} of 0.96852. The 95/95 k_{eff} was then determined by adding the temperature and methodology biases and the statistical sum of independent tolerances and uncertainties to the nominal k_{eff} values, as described in WCAP-14416-NA-P. This resulted in a 95/95 k_{eff} of 0.99985. Since this value is less than 1.0 and was determined at a 95/95 probability/confidence level, it meets the NRC criterion for precluding criticality with no credit for soluble boron and is acceptable.

Soluble boron credit is used to provide a safety margin by maintaining k_{eff} less than or equal to 0.95, including 95/95 uncertainties. The soluble boron credit calculations assumed that the all-cell storage configuration was moderated by water borated to 350 parts per million (ppm). As previously described, the individual tolerances and uncertainties, and the temperature and methodology biases, were added to the calculated nominal k_{eff} to obtain a 95/95 value. The resulting 95/95 k_{eff} was 0.94470. Since k_{eff} is less than 0.95 with uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion for precluding criticality is satisfied with 350 ppm of boron. This value is well below the minimum SFP boron concentration value of 2000 ppm required by VEGP TS 3.7.17, "Fuel Storage Pool Boron Concentration," and is, therefore, acceptable.

The concept of reactivity equivalencing due to fuel burnup was used to achieve the storage of fuel assemblies with enrichments higher than 3.50 w/o U-235 for the all-cell storage configuration. The NRC has previously accepted the use of reactivity equivalencing predicated upon the reactivity decrease associated with fuel depletion. To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$ for storage of fuel assemblies with enrichments up to 5.0 w/o U-235, a series of reactivity calculations was performed to produce a set of enrichment versus fuel assembly discharge burnup ordered pairs, which all yield an equivalent k_{eff} when stored in the MYAPP spent fuel storage racks. These are shown in proposed VEGP TS Figure 3.7.18-1, "Vogtle Unit 1 Burnup Credit for All Cell Storage," for VEGP Unit 1, and represents combinations of fuel enrichment and discharge burnup that yield the same rack k_{eff} as the rack loaded with fresh 3.50 w/o fuel. Uncertainties associated with burnup credit include a reactivity uncertainty of 0.01 Δk at 30,000 MWD/MTU applied linearly to the burnup credit requirement to account for calculational and depletion uncertainties and 5 percent on the calculated burnup to account for burnup measurement uncertainty. The NRC staff concludes that these uncertainties conservatively reflect the uncertainties associated with burnup calculations and are acceptable. The amount of additional soluble boron, in excess of the value required above, that is needed to account for these uncertainties is 200 ppm. This results in a total soluble boron credit requirement for the all-cell configuration of 550 ppm. This value is well below the minimum SFP boron concentration value of 2000 ppm required by VEGP TS 3.7.17 and is, therefore, acceptable.

Burnup reactivity equivalencing, as previously described, was also used to determine the allowed storage of fuel assemblies with enrichments higher than 2.45 w/o (VEGP Unit 1) and 2.40 w/o (VEGP Unit 2) but no greater than 5.0 w/o U-235 in the 3-out-of-4 configuration. The amount of soluble boron needed to account for the additional uncertainties associated with burnup credit in both units was 150 ppm. This is additional boron in excess of the 200 ppm required above, resulting in a total soluble boron requirement of 350 ppm. This is well below the minimum spent fuel pool boron concentration value of 2000 ppm required by TS 3.7.17 and is, therefore, acceptable.

Storage of assemblies with enrichments higher than 3.50 w/o U-235 in the all-cell configuration in the MYAPP spent fuel storage racks was determined by crediting the reactivity decrease associated with the addition of integral fuel burnable absorbers (IFBAs). IFBAs consist of neutron-absorbing material applied as a thin ZrB_2 coating on the outside of the UO_2 pellet. The fuel assembly is modeled at its most reactive point in life. This includes any time in life when the IFBA has depleted and the fuel assembly becomes more reactive. As with burnup credit, for IFBA credit reactivity equivalencing, a series of reactivity calculations is performed to produce a set of IFBA rod number versus initial enrichment ordered pairs that all yield the equivalent k_{eff} when the fuel is stored in the all-cell configuration analyzed for the MYAPP spent fuel racks in VEGP Unit 1. Uncertainties associated with IFBA credit include a 5 percent manufacturing tolerance and a 10 percent calculational uncertainty on the B-10 loading of the IFBA rods. The staff finds these uncertainties adequately conservative and acceptable. The amount of additional soluble boron needed to account for these uncertainties is 250 ppm. Therefore, the total soluble boron credit required for the all-cell configuration in the MYAPP spent fuel racks is 600 ppm. However, this is well below the minimum SFP boron concentration value of 2000 ppm required by VEGP TS 3.7.17 and is, therefore, acceptable.

As an alternative method for determining the acceptability of fuel assembly storage based on IFBA loading, the infinite neutron multiplication factor (k_∞), was used as a reference reactivity value. When k_∞ is used as a reference reactivity point, the need to specify an acceptable enrichment versus the number of IFBA rods correlation is eliminated. Fuel assemblies with a reference k_∞ of 1.431 in the VEGP Unit 1 core geometry at 68 °F have been shown to result in a maximum $k_{eff} \leq 0.95$ when stored in the MYAPP spent fuel storage racks; therefore, all fuel assemblies placed in the MYAPP spent fuel racks in an all-cell configuration must have an initial nominal enrichment less than or equal to 3.50 w/o U-235, or must satisfy a minimum IFBA requirement for higher initial enrichments to maintain the reference fuel assembly k_∞ less than or equal to 1.431 at 68 °F in the VEGP core geometry.

The VEGP Unit 1 SFP was also analyzed assuming a 3-out-of-4 checkerboard storage configuration containing three initially enriched 5.0 w/o U-235 assemblies and an empty cell (or a stored non-fuel-bearing component). This resulted in a 95/95 k_{eff} of 0.99745 with no credit for soluble boron. This value meets the NRC criterion of k_{eff} less than 1.0 with no credit for soluble boron. The same configuration was then analyzed to obtain the required 5 percent subcritical margin assuming 450 ppm of soluble boron. The resulting 95/95 k_{eff} was 0.94104. Since this k_{eff} value is less than 0.95, including soluble boron credit and uncertainties at a 95/95 probability/confidence level, the NRC acceptance criterion is met for the 3-out-of-4-cells storage configuration.

Although most accidents will not result in a reactivity increase, four accidents can be postulated for each storage configuration that would increase reactivity beyond the analyzed conditions. The first would be a change in the spent fuel pool water temperature outside the normal operating range. The second accident would be a misload of an assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied. The third would be a drop of an assembly onto an already loaded cell. The fourth accident would be a misload between the spent fuel storage rack module and the spent fuel pool wall. Calculations have shown that the misloaded assembly accident between the rack module and pool wall in the 3-out-of-4 checkerboard results in the highest reactivity increase. The reactivity increase requires an additional 350 ppm of soluble boron above the 450 ppm normal condition requirement for the 3-out-of-4 configuration to maintain $k_{eff} \leq 0.95$. However, for such events, the double contingency principle can be applied. This states that the assumption of two unlikely, independent, concurrent events is not required to ensure protection against a criticality accident. Therefore, the minimum amount of soluble boron required by VEGP TS 3.7.17 (2000 ppm) is more than sufficient to cover any accident, and the presence of the additional soluble boron above the concentration required for normal conditions can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

In order to prevent an undesirable increase in reactivity, the boundary between the two different storage configurations in Unit 1 was analyzed. The boundary can be either separated by a vacant row of cells or the interface must be configured so that the first row of cells after the boundary in the 3-out-of-4 storage region uses alternating empty cells and cells containing assemblies at the 3-out-of-4 configuration enrichment of up to 5.0 w/o U-235. The interface requirements are shown in proposed VEGP TS Figure 4.3.1-6, "Vogtle Units 1 and 2 Interface Requirements (All Cell to Checkerboard Storage)."

On the basis of the preceding review, the staff finds that the criticality aspects of the proposed VEGP license amendment request are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The analysis assumed credit for soluble boron, as allowed by WCAP-14416-NP-A. The required amount of soluble boron for each analyzed storage configuration is given in the following Table 1 summary.

The following storage configurations and U-235 enrichment limits for Westinghouse 17 x 17 fuel assemblies were determined to be acceptable for VEGP Unit 1:

- Assemblies with initial nominal enrichments no greater than 3.50 w/o U-235 can be stored in any cell location. Fuel assemblies with initial nominal enrichments greater than this and up to 5.0 w/o U-235 must satisfy a minimum burnup requirement as shown in proposed TS Figure 3.7.18-1, "Vogtle Unit 1 Burnup Credit Requirements for All Cell Storage," or must have a maximum reference fuel assembly k_{∞} less than or equal to 1.431 at 68 °F.

- Assemblies with initial nominal enrichments no greater than 5.0 w/o U-235 can be stored in a 3-out-of-4 checkerboard arrangement with empty cells (or with cells containing non-fuel-bearing components). A 3-out-of-4 checkerboard means that no more than three fuel assemblies can occupy any 2 x 2 matrix of storage cells.

TABLE 1

Summary of Soluble Boron Credit Requirements for Vogtle Unit 1

Storage Configuration	Soluble Boron Required for $k_{eff} \leq 0.95$ (ppm)	Soluble Boron Required for Reactivity Equivalencing (ppm)	Total Soluble Boron Credit Required Without Accidents (ppm)
All Cells	350	250	600
3-out-of-4 Checkerboard	450	0	450

3.2 Hoisting and Control of Heavy Loads

The licensee plans to use the overhead bridge crane in the fuel handling building to move the MYAPP spent fuel racks to the SFP, and to install a new temporary gantry crane for moving the racks within the fuel handling building. An offset fuel handling tool is to be installed to reach spent fuel assemblies and move them to storage locations that are adjacent to the pool walls.

3.2.1 Hoisting System

As indicated by the licensee, the load handling operations will be conducted in accordance with the guidelines in Section 5 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980, as it relates to safe load paths, procedures, crane operator training, inspection and maintenance, and testing.

The hoisting system consists of the overhead bridge crane in the fuel handling building, the temporary gantry crane to be installed, and lifting devices. The lifting devices consist of standard lift rigs and spreader bars that will be interposed between the racks and the crane hook during lifts.

The overhead bridge crane in the fuel handling building is rated at 125 tons on the main hoist and 15 tons on the auxiliary hoist. The maximum weight of the spent fuel racks to be removed is 33,000 pounds and 25,075 pounds for the spent fuel racks to be installed. Because of the safe load path, some areas of the SFP are inaccessible using the fuel handling building's overhead bridge crane. Therefore, the licensee will erect a temporary 20-ton gantry crane on the existing fuel handling bridge rails to move racks within the SFP area. The loading capacity of both the overhead bridge crane and the temporary gantry crane will enable the licensee to handle the storage racks during the rerack operations. Because all of the irradiated fuel will be stored in the Unit 2 SFP and the rerack is to be performed in the Unit 1 SFP, crane and load will not travel over irradiated fuel. Also both cranes will have mechanical stops to restrict crane travel over new or spent fuel. The licensee has committed to further review the load paths to ensure adequate protection of safe-shutdown equipment. To protect safe-shutdown equipment, the licensee will either limit the maximum travel height, upgrade the hoisting system, or use redundant rigging in accordance with the guidelines of NUREG-0612.

The licensee also commits to load test the crane and the lifting devices in accordance with the guidance of NUREG-0612 and ANSI Standard B30.2, "Overhead and Gantry Cranes," 1976. Both the temporary gantry crane and the lifting devices are to be load tested at 125 percent of the maximum service load (rated load) before they are used. However, instead of load testing the crane by transporting the trolley for the full length of the bridge and runway as required by ANSI Standard B30.2, the licensee noted that a rack will be hoisted 6 inches above the floor and held for approximately 10 minutes before starting the rerack operation. This will enable the licensee to avoid transporting the test load over irradiated fuel to do the load test and satisfy the requirements of ANSI Standard B30.2.

3.2.2 Postulated Load Drop Accidents

As indicated by the licensee, the spent fuel racks will not be lifted over or close to spent fuel because the Unit 1 spent fuel assemblies will be moved to the Unit 2 SFP. In addition, load paths will be maintained at a maximum distance from the Unit 2 spent fuel. Load paths will be established so that the travel and lift heights over safety-related equipment are minimized in accordance with guidelines of NUREG-0612. This will help to reduce the impact if a heavy load is dropped.

The licensee presented a refueling accident analysis performed in accordance with NRC's "Office Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978. In that analysis, the licensee evaluated the consequences of an accidental drop of a spent fuel assembly from the highest lift point during fuel handling operations. The licensee found that no significant safety impacts would result from a dropped fuel assembly. Releases of radioactive material resulting from load drops could not occur due to the absence of irradiated fuel during rack installation in the Unit 1 SFP. Fuel damage could not result in any increase in the subcriticality margin where K_{eff} is less than 0.95 because there is no irradiated fuel in the SFP. Damage to the SFP would not result in water leakage that could uncover the fuel; and the potential for damaging safe-shutdown systems is highly unlikely. This enables the licensee to satisfy the guidelines in Section 5.1 of NUREG-0612.

The licensee did not evaluate the drop of a rack during installation because irradiated fuel assemblies would not be present in the Unit 1 SFP or within the load path. The licensee stated that there is no potential for the load handling accident to result in consequences that exceed the guidelines presented in Section 5.1 of NUREG-0612. As noted, the licensee will prevent accidental load drops by applying defense-in-depth measures described in NUREG-0612. These measures include plans to train load handling system operators; steps to assure that the lifting devices and rigs are in accordance with the provisions of ANSI N14.6-1978, "Special Lifting Devices for Shipping Containers Weighing 10,000 lbs (4500 kg) or More," 1978; plans to conduct inspections and load testings in accordance with ANSI Standard B30.2; and plans to implement procedures to address the handling of specific heavy loads during the entire rerack operation.

The licensee committed to verify the operability of all cranes and lifting devices before starting the reracking operation. Both the crane/hoists system in the fuel handling building and the temporary gantry crane would be verified for compliance with design and testing requirements of CMAA Specification No. 70, "Crane Manufactures Association of America Inc., Specification No. 70—Specification for Electric Overhead Traveling Cranes," 1975, and ANSI Standard B30.2. In addition to the testing, the licensee stated that it will develop various load handling procedures to assure compliance with NUREG-0612. The licensee's method of verifying that the hoisting system is functional, coupled with its procedures to minimize operator errors, rigging failures, and inadequate inspections, is acceptable to the staff.

3.2.3 Conclusions Concerning Hoisting and Control of Heavy Loads

On the basis of the preceding discussion, the NRC staff finds that the licensee's methods of handling heavy loads during the rerack operation, including the licensee's commitment to verify the operability of the crane and lifting systems in accordance with the requirements for design and operation before performing the rerack, and the administrative procedures to improve the handling and control of heavy loads are in accordance with the guidelines of NUREG-0612. These changes enable the licensee to perform its rerack operation in a safe manner.

The licensee's evaluation of the consequences of postulated load drops of spent fuel storage racks and spent fuel assemblies satisfies the guidelines in Section 5.1 of NUREG-0612. The licensee has committed to use procedures and redundant rigging to prevent load drops that could severely impact safe-shutdown equipment. On the basis of the preceding discussion, the staff concludes that the temporary gantry crane and the upgraded lifting devices, testing of the hoisting system, operator training, and procedures for inspection and rack removal and installation will reduce the probability of a load drop in the SFP to an acceptable level. Therefore, the proposed changes to the capacity of the SFP are acceptable from the standpoint of hoisting and control of heavy loads.

3.3 Material Compatibility Considerations

The June 16, 1982, NRC staff SE, concerning the MYAPP spent fuel racks, addressed the compatibility of the spent fuel rack materials with the SFP environment. The sections from the June 16, 1982, NRC staff SE that relate to this issue are as stated:

2.5.1 Materials Description

We have reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) in the pool water. The proposed new spent fuel storage racks are fabricated of Type 304 stainless steel with the exception of the adjusting bolts of the rack feet. These bolts are made from Type 17-4 PH stainless steel. The 17-4 PH stainless steel [will] be heat-treated at 1100°F.

The spent fuel storage pool contains high purity water with approximately 2,000 ppm boron as boric acid present in it. Tight controls are placed on impurities in this water, such as chlorides and fluorides to minimize stress corrosion cracking (SCC).

The new high density fuel rack modules are composed of poison canisters and a bottom grid. The poison canisters consist of two concentric stainless steel tubes with Boral neutron poisonous material in the annulus. Boral is Boron carbide in an aluminum matrix core, clad with 1100 series aluminum.

2.5.2 Chemical Compatibility

Leakage of water into the weld sealed Boral cavity due to weld failure is unlikely, since welds are made in accordance with ASME Code [American Society of Mechanical Engineers Boiler and Pressure Vessel Code] procedures and both inspected and leak checked. Without the presence of water in the cavity, hydrogen gas resulting from the corrosion of aluminum will not be present. Even if some gas should form, the rack design utilizes the inner wall core as the structural member, so that only the outer skin would bow from gas buildup, thereby preventing the fuel bundle, which is inside the canister, from being wedged and causing any mislocation of the Boral. If isolated cases of leakage should occur, any swelling of the cans would not represent a safety hazard.

Upon exposure of the Boral plates (B_4C/AL matrix) to the spent fuel pool water, galvanic coupling between the aluminum-Boral liner, aluminum binder and the stainless steel shroud could occur. Deterioration of the Boral plates would be limited to edge attack by general corrosion and pitting corrosion of the aluminum liner and binder in the general area of the leak. The B_4C neutron adsorption particles are inert to the pool water and would become embedded in corrosion products preventing loss of the B_4C particles. Thus, this small amount of deterioration would have no effect on neutron shielding, attenuation properties or criticality considerations [Fuel Storage Racks Corrosion Test Program, Boral-

Stainless Steel Xn-NS-TP-009, Exxon Nuclear Company, Inc., October 1978, Richland Washington].

Boral neutron poison material encapsulated in stainless steel in a borated water coolant environment has been previously reviewed and accepted by us for similar designs in the Salem Nuclear Station and the Zion Nuclear Station. These plants have ongoing material surveillance programs which will timewise, lead the operation of the Maine Yankee Spent Fuel Pool. In the unlikely event that any adverse service experience is noted in these surveillance programs there would be sufficient time to initiate corrective action for Maine Yankee. In addition, the performance of other materials of which the spent fuel pool is constructed have been proven by experience and tests [Reactor Handbook, Volume 1 - Materials, Interscience Publishers, 1960] to be stable and to operate satisfactorily at both temperatures and radiation levels in excess of those anticipated in the Maine Yankee Spent Fuel Pool. Based on the above, we conclude that a materials surveillance program is not necessary in the case of the Maine Yankee Spent Fuel Pool.

The pool liner, rack lattice structure, adjusting bolts and fuel storage canisters are stainless steel, which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.00 X 10^{-5} inch in 100 years [A. B. Johnson, Jr., "Behavior of Spent Nuclear Fuel in Water Pool Storage" BNWL 2256, September 1977], which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, adjusting bolts and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials.

2.5.3 Conclusions

We conclude that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the remaining life of the plant [C. Czajkowski, J. R. Weeks, et. al., "Corrosion of Structural and Poison Material in Spent Fuel Storage Pools". Paper 163, Corrosion/81, April 6, 1981.]. Components in the spent fuel storage pool are constructed of alloys which have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. We therefore conclude that the environmental compatibility and stability of the materials used in the spent fuel storage pool is adequate based on test data and actual service experience in operating reactors. We find that the selection of appropriate materials of construction by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 62, preventing criticality by maintaining structural integrity of components and is therefore acceptable.

As stated previously, the only modification performed on the MYAPP spent fuel racks, following issuance of the June 16, 1982, NRC staff SE, was the addition of vent and drain holes in pockets that contain the neutron-absorbing Boral material. Long-term contact between Boral and borated water is not expected to cause any problems (e.g., swelling or excessive corrosion) as demonstrated by the acceptable performance that the racks demonstrated at MYAPP.

The VEGP SFP environment is very similar to the SFP environment at MYAPP in that they both employ the same materials at similar temperatures and radiation levels. Both VEGP and MYAPP SFPs are lined with stainless steel and the pool water is of high quality and borated for criticality control. In addition, both MYAPP and VEGP store Zircaloy-clad fuel in the spent fuel racks. From the preceding, the NRC staff concludes that use of the MYAPP spent fuel racks in the VEGP Unit 1 SFP will be acceptable from a materials compatibility standpoint, as was concluded for the use of these racks in the MYAPP SFP.

3.4 Spent Fuel Cooling System

Each VEGP SFP has an SFP cooling and purification system (SFPCPS). Each SFPCPS contains two subsystems; the SFP cooling system, and the SFP purification system. The SFPCPS is designed to remove the decay heat generated by stored spent fuel assemblies and to clarify and purify the water in the SFP. The primary safety function of the SFPCPS is to adequately transport this heat load to the component cooling water (CCW) system and thereby maintain the bulk pool temperature within its specified limit. The system consists of two independent cooling trains. Each train is seismically qualified and safety-related, and contains one heat exchanger and one pump. Heat is removed from the SFP heat exchangers by the CCW system. A purification loop, which includes a demineralizer and a filter, removes fission products and other contaminants that may be introduced if leaking fuel assemblies are transferred to the SFP. A portion of either train of SFP water may be diverted through the demineralizer and filter at a rate of 100 gallons per minute (gpm) to maintain pool clarity and purity.

3.4.1 Decay Heat Load Limit

The decay heat load limit for the refueling cases discussed in the VEGP Updated Final Safety Analysis Report (UFSAR), Section 9.1.3, is applicable to both Units 1 and 2 SFPs. In the current UFSAR, a projected fuel discharge scheme was used to determine the loading of the Unit 2 pool, which has a capacity of 2098 assemblies. On the basis of this scheme, it was determined that with the pool essentially at capacity, the addition of a full-core offload would not result in the bulk pool temperature exceeding the licensing basis temperature of 171.1 °F. Since fuel management (i.e., numbers of assemblies and burnups) may vary from cycle to cycle, the licensee determined that it would not be feasible to assume a single discharge scheme. Instead, in the proposed amendments, the licensee manages the total heat load in order to control the SFP temperature within a specified limit.

Accordingly, the licensee performed the rerack heat load analysis in accordance with NRC Branch Technical Position (ASB) 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling," Revision 2, in order to determine the maximum total heat load to control the bulk SFP temperature to within a limit of 170 °F for nonaccident conditions. This temperature was chosen since it is within VEGP's previously licensed value of 171.1 °F. Since the SFP cooling systems for Units 1 and 2 SFPs are identical, the licensee plans to apply the same temperature limit and corresponding heat load to both pools and to manage the pool heat loads by means of administrative controls in plant procedures.

The following conservatisms are included in the licensee's decay heat load limit calculation:

(1) SFPCPS heat exchanger thermal performance is based on the design maximum fouling level to minimize the heat rejection capability of the SFPCPS.

(2) In calculating the SFP evaporation heat losses, the SFP building is assumed to have the maximum ambient air temperature of 104 °F and 100 percent relative humidity to minimize the credit for evaporative heat loss.

3.4.2 Maximum Normal Refueling Case

The normal practice at VEGP is to unload the entire core for each refueling outage, which is referred to as the maximum normal refueling case in VEGP UFSAR Section 9.1.3. In the VEGP UFSAR heat load analysis, a full-core offload at 120 hours after shutdown is assumed with a heat load limit of 54.1 x 10E6 BTU/hr. The bulk SFP temperature peaks at 171.1 °F, then the temperature falls to below 150 °F approximately 400 hours after discharging the entire core to the spent fuel pool. Since a refueling outage takes place about once every 18 months, this is equivalent to approximately 3 percent of the total cycle time. The licensee stated that usually more than half of the fuel that was offloaded will be reloaded into the core well before 400 hours. Therefore, the actual time for the temperature to be above 150 °F would be less than 400 hours. In addition, the concrete walls and floor are several feet thick with a temperature gradient across them. Only a few inches of concrete would experience temperatures above 150 °F for short periods. For long-term durations between refuelings, the bulk pool water temperature and the concrete temperature would remain below 150 °F.

For the maximum normal refueling case, the rerack heat load analysis revealed that a steady-state heat load of 51.87 x 10E6 BTU/hr would maintain the SFP bulk SFP temperature at 170 °F with a single train of cooling in operation. Operating at or below this heat load will ensure that the SFP bulk spent fuel pool temperature would not exceed its limit of 170 °F for nonaccident conditions. In the current UFSAR, the temperature of 171.1 °F and the corresponding heat load of 54.1 x 10E6 BTU/hr are similar to the values in the rerack analyses. Therefore, the licensee concluded that the time referenced above for estimating the maximum time the temperature would remain above 150 °F is bounding for the rerack analyses. The licensee stated that the pool temperatures for long-term operation would remain as described in the VEGP UFSAR, which is below the 150 °F requirement in the American Concrete Institute (ACI) Standard 349, "Code Requirements for Nuclear Safety-Related Concrete Structures," 1985.

Before offloading a core to the SFP, the licensee will evaluate the impact of the offload on the total SFP heat load through plant procedures to ensure that the total heat load from earlier offloads and the recent core discharge do not exceed 51.87 x 10E6 BTU/hr. The evaluations will include the necessary in-core delay times, prior to offloading spent fuel to the SFP, to ensure that this limit will be met.

The NRC staff performed a confirmatory decay heat load calculation and verified that the proposed decay heat load limit was acceptable. Also, the NRC staff verified that the long-term bulk SFP temperature of less than 150 °F was within the limit specified in ACI Standard 349. Therefore, the staff finds that the licensee's VEGP Unit 1 rerack analysis for the maximum normal refueling case is acceptable.

3.4.3 Maximum Emergency Core Unloading Case

The maximum emergency core unloading case for VEGP Unit 2 as described in the current UFSAR Section 9.1.3 assumes that the entire core is unloaded into the SFP at 150 hours after the emergency shutdown of the reactor. It also assumes that 84 assemblies from the most recent refueling, with a decay time of 36 days, and 1821 assemblies from earlier refuelings are present in the SFP. For this case, the decay heat load limit is 58.13 x 10E6 BTU/hr to maintain the SFP temperature below 182 °F.

In the rerack analysis, the licensee did not perform a new analysis for the maximum emergency core unloading case for VEGP Unit 1. Since the VEGP Unit 2 SFP has a capacity of 2098 assemblies, the heat load and temperature analysis for the VEGP Unit 2 SFP bounds the VEGP Unit 1 SFP with a capacity of 1476 assemblies. The licensee concluded that the Units 1 and 2 SFPCPSs are identical, so the bounding analysis for the emergency core unloading case applies to both units as it does in the VEGP UFSAR. The licensee plans to revise the UFSAR to reflect that the SFP temperature will be limited to 182 °F by controlling the heat load, and that a heat load evaluation will be performed before an emergency unloading of the spent fuel assemblies.

The NRC staff finds that the Unit 1 maximum emergency core unloading case is bounded by the Unit 2 analysis with a single train of cooling and a bulk SFP temperature of less than 182 °F. This case is conservative since Standard Review Plan (SRP) Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," only requires that bulk SFP boiling should not occur with two trains of cooling. Therefore, the staff finds that the licensee's Unit 1 maximum emergency core unloading case for the proposed amendments is acceptable.

3.4.4 Effects of SFP Boiling

In the event that all forced SFP cooling becomes unavailable, the SFP water temperature will rise and eventually reach the normal bulk boiling temperature of 212 °F. The licensee determined that the minimum time to reach the boiling point is 2.90 hours by assuming that the decay heat load and the bulk SFP temperature limit are at their maximum calculated values. Since the SFPCPS has two independent trains, which are seismically qualified and safety related, the probability of a complete loss-of-cooling event coinciding with the instant that the SFP water has reached its peak value is unlikely.

The licensee calculated the boiloff rate of the VEGP Unit 1 SFP at the decay heat load limit to be 5.347 x 10E4 lb/hr (approximately 111.5 gpm). The primary source of makeup water for the SFP is the refueling water storage tank (RWST), which serves as the seismic Category 1 makeup water source that can be pumped or gravity-fed into the discharge line from SFP

Pump A. The RWST has a total capacity of 715,000 gallons that can be provided to the SFP at a rate of 200 gpm within 2.90 hours.

On the basis of its review, the staff finds that the makeup rate to the SFP exceeds the boiloff rate, and the time in which the makeup water can be provided to the SFP occurs within the minimum "time-to-boil," as recommended by Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Bases," and SRP Section 9.1.3; therefore, the staff finds that the licensee's time-to-boil analysis is acceptable.

3.4.5 Conclusion Concerning Spent Fuel Pool Cooling

On the basis of the preceding evaluation, the NRC staff's confirmatory decay heat load calculation and the licensee's fulfillment of the commitments documented in Section 5.0, herein, the NRC staff concludes that the thermal-hydraulic aspects of the proposed amendments for increasing the capacity of the Unit 1 SFP from 288 to 1476 assemblies are acceptable .

3.5 Radiological Assessment

The NRC staff has evaluated the radiological aspects of the licensee's proposed reracking of the VEGP Unit 1 SFP as described in this section.

3.5.1 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for the replacement of the VEGP Unit 1 spent fuel racks with respect to occupational radiation exposure. As stated above, for this modification the licensee plans to remove the two existing SFP rack modules and replace them with 26 rack modules that were previously licensed by the NRC for use at MYAPP. The licensee will then decontaminate the two SFP rack modules removed from VEGP and will take them from the site. On the basis of experience gained from SFP reracking operations performed at other plants, the licensee estimates that it can perform the proposed reracking for approximately 4.3 person-rem. This dose estimate is based on the licensee's detailed review of the anticipated work activities, durations, and expected dose rates associated with each of the activities associated with the SFP reracking.

In order to achieve this dose, the licensee plans to closely monitor and control work, personnel traffic, and the movement of equipment to minimize contamination and to assure that exposures are maintained as low as is reasonably achievable (ALARA). All activities will be governed by radiation work permits, and all personnel will be provided with electronic personnel dosimeters.

Each diver will be monitored, using multiple teledosimetry devices to ensure accurate recording of their doses. These teledosimetry devices will transmit diver dose and dose rate data to a computer, which will display the data on a monitor near the SFP. These data will be monitored continuously by a technician. Divers will be able to perform underwater area surveys using a remote-readout radiation-monitoring instrument capable of measuring dose rates as high as 1000 rem/hr.

The licensee will remove all spent fuel assemblies and all known sources of high radiation from the Unit 1 SFP before sending divers into the SFP. In addition, the licensee will close the weir gates connecting the Unit 1 SFP with the cask loading pit and the Unit 1 transfer canal. The licensee will perform an extensive underwater radiation survey of the Unit 1 SFP before allowing divers access to the SFP to remove the old SFP storage racks. All divers will be fitted with a tether to control their movements in the SFP. The licensee will also use cameras to monitor diving operations.

The 26 SFP rack modules that will be installed in the Unit 1 SFP were obtained from MYAPP. These racks will be unpacked in a contamination control area having high-efficiency particulate air filtered ventilation. The racks will be surveyed, checked for hot spots, and decontaminated, if necessary, before installation in the Unit 1 SFP.

The licensee will monitor and control personnel traffic and equipment movement in the SFP area to minimize contamination and generation of radioactive wastes. The licensee will use a combination of long-handled and diver-controlled tools to facilitate SFP rack module removal and installation. The use of diver-controlled tools will reduce the need for decontamination during remote-tool handling.

During reracking operations, there is the potential for an increase in radioactivity concentrations in the SFP from crud spalling from spent fuel assemblies during movement. In order to minimize the effects of spalling in the SFP, the licensee will move all spent fuel to the Unit 2 SFP and will clean the racks to be removed before removing them from the Unit 1 SFP. The licensee also plans to use an underwater vacuum to minimize any potential radiological effects of spalling and to maintain water clarity in the Unit 1 SFP.

The licensee estimates that the increased number of fuel assemblies stored in the Unit 1 SFP may result in a small increase in doses in the areas adjacent to the sides of the SFP, although any increase will not be enough to change any existing radiation zone designations. To minimize any potential dose rate increases from the increased storage of spent fuel, the licensee plans to control the placement of freshly discharged fuel so it is not placed in SFP rack positions adjacent to the sides of the SFP. Dose rates on the fuel pool level are primarily due to radionuclides in the pool water. During normal operations, dose rates in this area are generally 2.5 mrem/hr or less. The staff finds these dose rates to be acceptable and in accordance with SFP dose rates at other plants.

The licensee does not expect the concentrations of airborne radioactivity in the vicinity of the SFP to increase because of the expanded SFP storage capacity. However, there will be a monitor in the area to continuously monitor airborne radioactivity levels. In addition, the plant effluent radiation monitoring system will monitor any gaseous releases.

On the basis of its review of the licensee's proposal, the NRC staff concludes that the VEGP Unit 1 SFP rack modification can be performed in a manner that will ensure that doses to the workers will be maintained ALARA. The staff finds that the projected dose for the work (approximately 4.3 person-rem) is in the range of doses for similar SFP modifications at other plants, and it is acceptable.

3.5.2 Solid Radioactive Waste

Spent resins are generated by the SFP purification system. In order to minimize the generation of spent resins, the licensee will vacuum, inspect, and remove any debris from the floor of the SFP before installing the SFP rack modules. Since the number of fuel assemblies handled in the pools annually at VEGP will not increase with the expanded storage capacity, the licensee concludes that the additional fuel storage will not result in a change of the amount of solid radwaste generated.

The existing spent fuel racks in the VEGP Unit 1 SFP will be removed from the site by a salvage company. After usable material has been salvaged, the volume of the remainder will be reduced and disposed of at the Barnwell, South Carolina, facility. In a worst-case scenario, no salvageable material and no volume reduction, the resulting material would represent 44 percent of the expected solid waste volume associated with VEGP Units 1 and 2 for 1998; however, this volume is not significant when viewed over the 40-year, operational lifetime of the VEGP facility.

3.5.3 Design-Basis Accidents

In the VEGP UFSAR, the licensee evaluated the possible consequences of the following three hypothetical accidents involving fuel in the SFP: a fuel handling accident in the fuel handling building; a fuel handling accident in the containment with the airlock closed; and a fuel handling accident in the containment with the airlock open. The licensee evaluated these hypothetical accidents to determine the thyroid and whole-body doses at the exclusion area boundary (EAB), low-population zone (LPZ), and control room. The proposed reracking of the VEGP Unit 1 SFP did not affect any of the assumptions or data used in evaluating the dose consequences of any of these hypothetical accidents.

The NRC staff reviewed the licensee's analysis and performed confirmatory calculations to check the acceptability of the licensee's doses. In performing these calculations, the staff used the assumptions in Regulatory Guide 1.25, "Assumptions Used For Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors." The NRC staff performed an assessment for the fuel handling accident with the most limiting dose consequences. For a fuel handling accident in the containment with the airlock closed, the radionuclide release to the environment will be mostly contained. For a fuel handling accident in containment with the airlock open, the release to the environment will be assumed to be released directly to the environment and, therefore, this accident will be bounded by the fuel handling accident in the fuel handling building. For a fuel handling accident in the fuel handling building, the radionuclide release is assumed to be released directly to the environment with no filtration; therefore, the NRC staff performed an assessment of a fuel handling accident occurring in the fuel handling building. For this accident, the staff assumed that the cladding of all of the fuel rods in a single fuel assembly (264 rods) plus an additional 50 rods (for a total of 314 rods) would be perforated if the fuel assembly were dropped during handling. The damaged fuel assembly is assumed to contain freshly offloaded fuel with a minimum of 100 hours of decay. The other parameters that the staff utilized in its assessment are presented in Table 2.

TABLE 2

ASSUMPTIONS USED FOR CALCULATING RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT IN THE FUEL HANDLING BUILDING

Parameters	Value
Power Level, MWt	3565
Number of Fuel Rods Damaged (Single Assembly + 50 Rods)	314
Total Number of Rods in Core	50,952
Shutdown Time, hours	100
Power Peaking Factor	1.7
Fission-Product Release Fractions (%)*	
Iodine	12
Noble Gases	30
Pool Decontamination Factors*	
Iodine	100
Noble Gases	1
Iodine Forms (%)*	
Elemental	75
Organic	25
Filter Efficiencies for Control Room (%)	99
Control Room Flow Rates (ft^3/min)	
Recirculation (emergency)	15,600
Intake (emergency/normal)	1,500/3,000
Unfiltered inleakage (emergency/normal)	15/15
Atmospheric Dispersion Factors, χ/Q (sec/m^3)	
Exclusion Area Boundary (0–2 hours)	1.8×10^{-4}
Low Population Zone (0–8 hours)	3.1×10^{-5}
Control Room (0–8 hours)	5.7×10^{-3}
Core Fission Product Inventories per TID-14844	

* Regulatory Guide 1.25

The staff's calculations confirmed that the thyroid doses at the EAB, LPZ, and control room from a fuel handling accident in the fuel handling building meet the acceptance criteria, and that the licensee's calculations are acceptable. The results of the staff's calculations are presented in Table 3. For a fuel handling accident in the fuel handling building, the staff calculated a dose of 54.5 rem to the thyroid at the EAB and 9.4 rem to the thyroid at the LPZ. The acceptance criterion at the EAB and LPZ for these accidents is contained in SRP Section 15.7.4 of NUREG-0800 and is 75 rem to the thyroid dose (25 percent of 10 CFR Part 100 guidelines of 300 rem). For the same accident, the staff calculated a dose of 5.0 rem to the thyroid of the control room operator. For this calculation, the staff assumed that the control room emergency ventilation system did not initiate until 2 minutes into the fuel handling accident (as per accident description in the VEGP UFSAR). The acceptance criterion for the control room operator dose is 30 rem to the thyroid (SRP Section 6.4 of NUREG-0800). The NRC staff, therefore, finds the proposed reracking of the VEGP Unit 1 SFP to be acceptable with respect to potential radiological consequences as a result of a hypothetical fuel handling accident.

TABLE 3

THYROID DOSES FROM FUEL HANDLING ACCIDENT
IN THE FUEL HANDLING BUILDING
AT VOGTLE, UNIT 1 (VALUES CALCULATED BY NRC STAFF)

AREA	FUEL HANDLING ACCIDENT DOSE (REM-THYROID)
EAB*	54.5
LPZ*	9.4
Control Room**	5.0

*Acceptance Criterion = 75 rem thyroid
**Acceptance Criterion = 30 rem thyroid

3.6 Structural Evaluation

The NRC staff has reviewed the use of the MYAPP spent fuel racks in the VEGP Unit 1 SFP to assure the structural integrity and functionality of the racks, the stored fuel assemblies and the SFP structure subject to the effects of the postulated loads (Appendix D to SRP Section 3.8.4) and fuel handling accidents.

3.6.1 Storage Racks

The 1476 storage cells will be contained in 26 fuel storage racks, which are seismic Category I equipment, and are required to remain functional during and after a safe-shutdown earthquake (SSE). The licensee, with assistance from its contractor, Holtec International, performed structural analyses for the spent fuel storage racks.

The licensee used a computer program, DYNARACK, for dynamic analysis to demonstrate the structural adequacy of the VEGP spent fuel rack design under the combined effects of earthquake and other applicable loading conditions. The proposed spent fuel storage racks are free-standing and self-supporting equipment (not attached to the floor of the storage pool). A nonlinear dynamic model consisting of inertial mass elements, spring elements, gap elements, and friction elements, as defined in the program, was used to simulate three-dimensional (3-D) dynamic behavior of the rack and the stored fuel assemblies, including frictional and hydrodynamic effects. The program calculated nodal forces and displacements at the nodes, and then obtained the detailed stress field in the rack elements from the calculated nodal forces.

Two model analyses were performed: the 3-D single rack (SR) model analysis and the 3-D whole-pool multi-rack (MR) model analysis. For the 3-D SR analysis, two rack geometries were considered for the calculation of stresses and displacements:

(1) 5.2 ft (W) x 7.7 ft (L) x 14.8 ft (H), and
(2) 6.8 ft (W) x 7.7 ft (L) x 14.8 ft (H) where W, L, and H are defined as width, length, and height of a rack, respectively.

Each rack was considered fully loaded, half-loaded, and almost empty, with three different coefficients of friction between the rack and the pool floor (μ=0.2, 0.5, and 0.8, respectively) to identify the worst-case response for rack movement and for rack member stresses and strains. In the 3-D MR analysis, 26 free-standing racks were considered to investigate the fluid-structure interaction effects between racks and pool walls, as well as those among the racks.

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

The licensee performed 85 3-D single-rack model analyses. The results of the analyses show that the maximum displacements of the racks at the top and the baseplate corners are about 4.81 inches and 2.67 inches, respectively, indicating that there is adequate safety margin against overturning of the racks and, thereby, the structural integrity and stability of the racks are maintained. In addition, the calculated stresses in tension, compression, bending, combined flexure and compression, and combined flexure and tension were compared with corresponding allowable stresses specified in ASME Code, Section III, Subsection NF. The comparisons show that all induced stresses under an SSE loading condition are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the rack design is adequate.

In the 3-D MR analyses, 26 fully loaded racks were considered and were subjected to the service, upset, and faulted loading conditions (Level A, B, and D Service Limits). The results of the MR analysis indicate that the calculated stresses on a rack are higher than those obtained from the single-rack analyses. However, all calculated stresses for the MR analyses are smaller than the corresponding allowable stresses of the ASME Code.

The licensee also calculated the weld stresses of the rack at the connections (e.g., baseplate-to-rack, baseplate-to-pedestal, and cell-to-cell connections) under the dynamic loading conditions. The licensee demonstrated that all the calculated weld stresses are smaller than the corresponding allowable stresses specified in the ASME Code, indicating that the weld connection design of the rack is adequate.

Based on (1) the licensee's comprehensive parametric study (e.g., varying coefficients of friction, different geometries and fuel loading conditions of the rack), (2) the adequate factor of safety of the induced stresses of the rack when they are compared to the corresponding allowable values given in the ASME Code, and (3) the licensee's overall structural integrity conclusions supported by both SR and MR analyses, the staff concludes that the rack modules will perform their safety function and maintain their structural integrity under postulated loading conditions and, therefore, are acceptable.

3.6.2 Spent Fuel Storage Pool

The SFP structure is made of reinforced-concrete and is designed as seismic Category I. The dimensions of the VEGP pool structure are approximately 34 feet wide, 50 feet long, and 40 feet deep. The internal surface of the pool structure is lined with 0.25-inch-thick stainless steel plates to ensure watertight integrity.

The pool structure was analyzed by using the finite element computer program ANSYS to demonstrate the adequacy of the pool structure under fully loaded fuel racks with all storage locations occupied by fuel assemblies. The fully loaded pool structure was subjected to the load combinations specified in the VEGP UFSAR.

The May 19, 1998, supplement shows the predicted factors of safety varying from 1.01 to 1.39 for bending moments and axial forces of the concrete walls and slab. In view of the calculated factors of safety, the staff concludes that the licensee's pool structural analysis demonstrates the adequacy and integrity of the pool structure under full fuel loading, thermal loading, and SSE loading conditions. Thus, the SFP design is acceptable.

3.6.3 Fuel Handling Accident

The licensee evaluated the following two refueling accident cases: (1) drop of a fuel assembly with its handling tool, which impacts the baseplate (deep drop scenario) and (2) drop of a fuel assembly with its handling tool, which impacts the top of a rack (shallow drop scenario).

The analysis of accident drop case 1 shows that the load transmitted to the liner through the rack structure is properly distributed through the bearing pads located near the fuel handling area; therefore, the liner would not be ruptured by the impact as a result of the fuel assembly drop through the rack structure. The analysis of accident drop case 2 shows that damage will be restricted to a depth of 13.5 inches below the top of the rack, which is above the active fuel region. The NRC staff reviewed the licensee's findings in the September 4, 1997, application and concurs with them. This is acceptable on the basis of the licensee's conclusions about structural integrity, supported by the parametric studies.

3.6.4 Conclusions Concerning Structural Evaluation

On the basis of its review and evaluation of the licensee's application, as supplemented, the NRC staff concludes that the licensee's structural analysis and design of the spent fuel rack modules and the SFP structures are adequate to withstand the effects of the applicable loads, including the effects of the SSE. The analysis and design are in compliance with the current licensing basis given in the VEGP UFSAR and applicable provisions of the SRP and, therefore, are acceptable.

4.0 TS CHANGES

The licensee has proposed changes to the VEGP Units 1 and 2 TS to reflect the proposed increase in the VEGP Unit 1 SFP storage capacity and the revised criticality analysis, described in Section 3.1, herein. The following revisions to the TS are proposed:

(1) TS 3.7.18 would be changed to reflect that separate criticality requirements apply to the Units 1 and 2 SFPs. Currently, TS 3.7.18 references the VEGP Units 1 and 2 criticality requirements in TS 4.3.1. The proposed TS 3.7.18 references TS 4.3.1.1 for criticality requirements in the VEGP Unit 1 SFP and TS 4.3.1.2 for criticality requirements in the VEGP Unit 2 SFP.

(2) TS Figure 3.7.18-1, "Vogtle Unit 1 Burnup Credit Requirements for All Cell Storage," would be replaced with a revised figure based on the criticality analyses for the VEGP Unit 1 racks containing Boral as previously evaluated.

(3) TS 4.3.1, "Criticality," would be separated into two sections, 4.3.1.1 and 4.3.1.2, to address the design features and the criticality requirements for the VEGP Units 1 and 2 SFPs, respectively. The criticality requirements for the VEGP Unit 2 SFP would not change.

(4) TS 4.3.3, "Capacity," would be revised to increase the VEGP Unit 1 storage capacity from 288 to 1476 assemblies.

(5) TS Figure 4.3.1-4, "Vogtle Units 1 and 2 Empty Cell Checkerboard Storage Configurations," TS Figure 4.3.1-6, "Vogtle Units 1 and 2 Interface Requirements (All Cell to Checkerboard Storage)," and TS Figure 4.3.1-7, "Vogtle Units 1 and 2 Interface Requirements (Checkerboard Storage Interface)," titles would be revised to reflect the elimination of a 2-out-of-4 storage configuration for VEGP Unit 1.

(6) Administrative changes to the TSs are proposed to change the Table of Contents and renumber the sections of TS 4.3, "Fuel Storage," to accommodate the separation of TS 4.3.1 into proposed TS 4.3.1.1 and TS 4.3.1.2.

The TS changes proposed as a result of the revised criticality analysis are consistent with NRC-approved methodology. On the basis of this consistency with the approved methodology and on the preceding evaluation, the staff finds these TS changes acceptable. The proposed associated Bases changes adequately describe these TS changes and are also acceptable.

5.0 LICENSEE COMMITMENTS RELIED UPON BY THE NRC STAFF

In letters dated September 4, 1997, May 19, 1998, and June 12, 1998, the licensee committed to the following:

(1) The SFP heat loads will be managed by administrative controls. These controls will be placed in applicable procedures before transferring irradiated fuel into the VEGP Unit 1 SFP.

(2) The UFSAR will be updated to include the heat load that will ensure the temperature limit of 170 °F will not be exceeded, as well as the requirement to perform a heat load evaluation before transferring irradiated fuel to either pool. This will be included in the next appropriate UFSAR update following the installation of the VEGP Unit 1 spent fuel racks.

(3) A temporary gantry crane, with a hoist rated for 20 tons, will be erected on the existing fuel handling bridge rails to move the racks within the SFP area. This commitment will be implemented before commencing reracking operations.

(4) The licensee will implement all applicable crane, load path and height, rigging and load test guidelines of NUREG-0612 and ANSI Standard B30.2 before and during reracking operations, as appropriate.

In the May 19 and June 12, 1998, supplements, the licensee proposed that these commitments be incorporated in Appendix D of the VEGP Unit 1 Facility Operating License. In addition, Commitment 2 should be incorporated in Appendix D of the VEGP Unit 2 Facility Operating License. The NRC staff agrees that these commitments should be incorporated in Appendix D of the VEGP Units 1 and 2 Facility Operating Licenses in that fulfilment of the preceding commitments is necessary to maintain the integrity of the analyses associated with installation and use of the MYAPP spent fuel racks in the VEGP Unit 1 SFP.

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the Federal Register on June 24, 1998 (63 FR 34491).

Accordingly, based on the Environmental Assessment, the Commission has determined that issuance of the amendments will not have a significant effect on the quality of the human environment.

8.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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