September 22, 2005

Mr. D. E. Grissette
Vice President
Southern Nuclear Operating
Company, Inc.
P.O. Box 1295
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS THAT REVISE THE SPENT FUEL POOL RACK CRITICALITY ANALYSES (TAC NOS. MC4225 AND MC4226)

Dear Mr. Grissette:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 139 to Facility Operating License NPF-68 and Amendment No. 118 to Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated August 13, 2004, as supplemented by letters dated on May 3 and July 7, 2005.

The amendments revise the TSs to reflect updated spent fuel rack criticality analyses for Units 1 and 2. The amendments also corrects a typographical error on Page vi of the TSs Table of Contents that occurred during the issuance of Amendments 130 and 109, for Units 1 and 2 TSs, respectively.

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,

/RA/

Christopher Gratton, Sr. Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 139 to NPF-68
2. Amendment No. 118 to NPF-81
3. Safety Evaluation

cc w/encls: See next page

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Division of Licensing Project Management
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Package Number: ML052420124 License Amendment Number: ML052420110
Tech Spec Number: ML052690018 NRR-058

OFFICE	PDII-1/PM	PDII-1/LA	DSSA	DE	OGC	PDII-1/SC
NAME	CGratton	CHawes	FAkstulewicz	LLund	SHamrick	EMarinos
DATE	9/21/05	9/21/05	6/22/05	7/15/05	9/20/05	0/21/05

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SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS THAT REVISE THE SPENT FUEL POOL RACK CRITICALITY ANALYSES (TAC NOS. MC4225 AND MC4226)

DATE: September 22, 2005

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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. 139
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 Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 13, 2004, as supplemented by letters dated May 3 and July 7, 2005, 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. 139, 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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 22, 2005

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. 118
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 Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 13, 2004, as supplemented by letters dated May 3 and July 7, 2005, 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:

    <u>Technical Specifications and Environmental Protection Plan</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 118, 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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 22, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 139

FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 118

FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
vi	vi
3.7.18-3	3.7.18-3
3.7.18-4	3.7.18-4
4.0-2	4.0-2
4.0-3	4.0-3
4.0-4	4.0-4
4.0-6	4.0-6
4.0-7	4.0-7
4.0-8	4.0-8
4.0-9	4.0-9
4.0-10	4.0-10
4.0-11	4.0-11
4.0-12	4.0-12
4.0-13	4.0-13
4.0-14	4.0-14
-	4.0-15

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NPF-68

AND AMENDMENT NO. 118 TO FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated August 13, 2004, as supplemented by letters dated May 3 and July 7, 2005, Southern Nuclear Operating Company (SNC, the licensee), submitted for Nuclear Regulatory Commission (NRC) staff review and approval of license amendments to the Vogtle Electric Generation Plant (Vogtle), Units 1 and 2, facility operating license and technical specifications (TSs). The proposed changes to the TSs will reflect updated spent fuel rack criticality analyses for Units 1 and 2. The analyses revised the enrichment, burn-up, and integral fuel burnable absorber (IFBA) limits required to comply with the allowed storage configurations. The request assumes full credit for the Boral® neutron poison material in Unit 1 and no credit for the degrading Boraflex neutron poison material in Unit 2 (using credit for soluble boron). In addition, the requested changes would allow use of credit for soluble boron in the spent fuel pool (SFP) criticality analysis. This criticality analysis was performed using methodology analogous to that developed by the Westinghouse Owners Group (WOG) and described in WCAP-14416-NP-A, Revision 1, “Westinghouse Spent Fuel Rack Criticality Analysis Methodology.

SNC also identified an typographical error in the Vogtle, Units 1 and 2 TSs that occurred during the issuance of Amendments 130 and 109 to Vogtle, Units 1 and 2 TSs. That error is corrected herein.

The supplemental letters dated May 3 and July 7, 2005, provided clarifying information that did not change the scope of the August 13, 2004, application or the initial proposed no significant hazards consideration determination.

2.0 REGULATORY EVALUATION

Regulatory Requirements and Guidelines

a) Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, “General Design Criteria (GDC) for Nuclear Power Plants,” provides a list of the minimum design requirements for nuclear power plants. According to GDC 62, “Prevention of criticality in fuel storage and handling,” the licensee must limit the potential for criticality in the fuel handling and storage system by physical systems or processes. The NRC staff reviewed the amendment request to ensure that the licensee complied with GDC 62.

b) 10 CFR Section 50.68, “Criticality accident requirements,” provides the NRC regulatory requirements for maintaining subcritical conditions in SFPs. Since the licensee currently uses 10 CFR 50.68 as the licensing basis for its SFP, the NRC staff has reviewed the proposed changes against the appropriate parts of the section.

c) Standard Review Plan 9.1.2, “Spent Fuel Storage,” provides guidance to ensure that there are no potential mechanisms that will: (1) alter the dispersion of the strong fixed neutron absorbers incorporated in the design of the storage racks, and (2) cause physical distortion of the tubes retaining the stored fuel assemblies.

3.0 TECHNICAL EVALUATION

The Vogtle spent fuel storage racks were re-analyzed by WOG, as documented in the licensee’s August 13, 2004, submittal. The methodology described in the Westinghouse report takes partial credit for soluble boron in the SFP criticality analyses and requires conformance with the regulatory requirements in Section 2.0 above.

3.1 Current Vogtle Licensing Basis

SNC had previously submitted a license amendment request to credit boron in the SFPs. The previously submitted analysis was based on the methodology described in Westinghouse topical report WCAP-14416-NP-A, Revision 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology." The topical was approved by the NRC on October 25, 1996, and issued by Westinghouse in November 1996.

The methodology provided for limited credit for soluble boron in the SFP to maintain K_{eff} < 0.95. The criteria set forth in this topical report were that K_{eff} remains less than unity with zero soluble boron and that K_{eff} remains less than or equal to 0.95 with credit for soluble boron with a 95 percent probability at a 95 percent confidence level (95/95). Fuel enrichments up to 5 weight percent (5 w/o) U-235 were considered in the analyses. In some cases, Vogtle found it necessary to credit burnup or IFBA in spent fuel rack geometry, or to maintain a reference K_{eff} below some specified value in reactor geometry in order to ensure that the SFP K_{eff} remained less than unity with zero soluble boron. Several storage configurations as well as interfaces between these configurations were considered.

3.2 Issues With Current Licensing Basis Methodology

Westinghouse issued two Nuclear Safety Advisory Letters reporting potential non-conservatisms in the methodology described in WCAP-14416-NP-A. NSAL-99-003 discussed potential non-conservatisms in the calculated IFBA requirements using the reference K_{eff} technique for reactivity equivalencing. NSAL-00-015 discussed potential non-conservatisms in the axial burnup shape reactivity bias. In addition, the NRC issued Regulatory Information Summary 01-012 to notify licensees of the potential for non-conservatisms in SFP criticality analyses if reactivity equivalencing is used. An evaluation of these issues was performed by Westinghouse and included in the August 13, 2004, submittal for Vogtle.

The re-analyzed SFP rack criticality results demonstrate that the current TSs for Vogtle, Units 1 and 2 continue to provide margin to the intended level of protection.

3.3 Description of Revised Analyses

SNC decided to update the Vogtle spent fuel rack criticality analyses utilizing methods that address the issues described above. The goals of the reanalyses were to show that the acceptance criteria set forth in WCAP-14416-NP-A continue to be met and that the currently permissible storage configurations described in the TSs continue to be acceptable. The results of the analyses provided updated soluble boron, burnup credit, and IFBA credit requirements. No physical plant changes are being made (i.e., no changes to the SFPs or racks, heat loads, supporting systems, etc.), only the criticality analyses are being updated.

In determining the acceptability of SNC's amendment request, the NRC staff reviewed three aspects of the licensee's analyses: 1) the computer codes employed, 2) the methodology used to calculate the maximum K_{eff}, and 3) boron dilution. For each part of the review, the NRC staff evaluated whether the licensee's analyses and methodologies provided reasonable assurance that adequate safety margins in accordance with NRC regulations were developed and could be maintained in the Vogtle SFP.

3.3.1 Computer Codes

The analysis of the fuel storage reactivity effects for Vogtle, Units 1 and 2 spent fuel racks was performed with SCALE-PC, a personal computer version of the SCALE-4.3 code package (which includes KENO-Va, NITAWL, CSAS-2, and BON-AMI), with the 44-group ENDF/B-V neutron cross section library. Since the KENO-Va code package does not have burnup capability, depletion analyses were made with the two-dimensional integral transport theory code, DIT, which uses an 89-group structure collapsed from the ENDF/B-VI library. The SCALE-PC models used in the reactivity analysis have been benchmarked against experimental data for fuel assemblies similar to those for which the Vogtle racks are designed and have been found to adequately reproduce the critical values. The selected critical experiments included the Babcock & Wilcox experiments carried out in support of close proximity storage of power reactor fuel and the Pacific Northwest Laboratory program carried out in support of the design of fuel shipping and storage configurations. This experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to Vogtle storage rack conditions.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The DIT code and its cross section set have been used in the design of reload cores and extensively benchmarked against operating reactor history and test data. The NRC staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Vogtle storage racks with a high degree of confidence.

Models were made for each storage configuration as well as for the entire pools. Each configuration was modeled as an infinitely repeating pattern in the X-Y plane. A water reflector was modeled above and below the spent fuel storage cells to account for axial reactivity effects. The KENO model for the entire pool modeled each individual storage rack module.

KENO was used to calculate K_{eff} for the various storage configurations considered and for the entire pool. For demonstrating that K_{eff} remains below unity for zero soluble boron, SNC chose to apply an acceptance criterion of 0.995. For the storage configurations considered, the target value of K_{eff} for these calculations was selected to be less than 0.995 by an amount

sufficient to cover the magnitude of the analytical biases and uncertainties. KENO was also used to determine burnup and IFBA versus enrichment requirements to meet the target K_{eff} for configurations that credit burnup or IFBA.

Calculations were performed for the entire pool with various fuel storage configurations to demonstrate that the K_{eff} for the entire pool remains below 0.995 with zero soluble boron. KENO was used to determine the soluble boron requirements for non-accident and accident conditions to ensure that K_{eff} remains less than or equal to 0.95.

3.3.2 Methodology

In accordance with the guidance contained in References 1 and 2, the licensee performed criticality analyses of its SFP. The licensee employed a methodology that combines a worst-case analysis based on the bounding fuel, rack, and control element assembly conditions, with a sensitivity study using 95/95 analysis techniques. The major components in this analysis were a calculated k_{eff} based on the limiting fuel assembly, SFP, and code biases and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances.

In performing its criticality analysis, the licensee first calculated a k_{eff} based on nominal core conditions using the KENO-Va code. The licensee determined this k_{eff} from the limiting (highest reactivity) fuel assemblies stored in the SFP. The licensee performed its reactivity analyses for various enrichments, cooling times, burnups, and the bounding cladding thicknesses. In performing these calculations, the licensee assumed appropriately conservative conditions such as an infinite radial checkerboard array. The Westinghouse 17x17 assembly served as the nominal assembly for all calculations the licensee performed in its respective SFP region.

To the calculated k_{eff}, the licensee added the methodology bias as well as a reactivity bias to account for the effect of the normal allowable range of SFP water temperatures. The licensee determined the methodology bias from the critical benchmark experiments. For each of the proposed storage configurations, the licensee analyzed the reactivity effects of the SFP water temperature. The licensee added a reactivity bias corresponding to the most reactive temperature range of 50 degrees Fahrenheit (°F) to 185 °F. The licensee also calculated the reactivity bias associated with a temperature decrease to the maximum density of water, 4 degrees Celsius.

Finally, to determine the maximum k_{eff}, the licensee performed a statistical combination of the uncertainties and manufacturing tolerances. The uncertainties included the KENO bias uncertainty and the KENO uncertainty. The licensee determined both of these uncertainties to a 95/95 threshold, which is consistent with the requirements of 10 CFR 50.68. For each tolerance, the licensee used the DIT code to calculate a delta-k between the nominal condition and the most limiting tolerance condition. By using the most limiting tolerance condition, the licensee calculated the highest reactivity effect possible. This results in conservative margin since the tolerances will always bound the actual parameters. Once the reactivity effects for each of the tolerances were determined, the licensee statistically combined each of the manufacturing tolerances with the 95/95 uncertainties. The NRC staff reviewed the licensee's methodology for calculating the reactivity effects associated with uncertainties and manufacturing tolerances as well as the statistical methods used to combine these values. The NRC staff finds the licensee's methods conservative and acceptable.

3.3.3 Boron Dilution Event

The licensee revisited the boron dilution event performed for Vogtle, Units 1 and 2 as part of an August 8, 1997, amendment application. The NRC staff had evaluated the boron dilution event and reported its findings in the safety analysis report (SER) for Amendments 99 and 77, dated February 20, 1998.

The dilution evaluation addressed a dilution from the TS-required minimum boron concentration requirement of 2000 ppm (TS 3.7.17) to 600 ppm. The evaluation concluded that a dilution event would be detected by alarms and plant personnel and would be terminated prior to reaching 600 ppm.

For non-accident conditions, the boron concentration requirement to maintain K_{eff} less than or equal to 0.95 is 511 ppm for the Unit 1 pool and 394 ppm for the Unit 2 pool. Therefore, the conclusions documented in the NRC staff's February 20, 1998, SER that the dilution event will be detected and terminated prior to the SFP reactivity exceeding a K_{eff} of 0.95 remains valid and no new dilution evaluation is required.

3.3.4 Summary of Criticality Analyses

The licensee's analyses used the most reactive design and the most reactive temperature to set the storage requirements. The analyses included means to account for the bias and uncertainty associated with the bench-marking of the methodology, a bias for the under-prediction of reactivity due to boron particle self-shielding, and the uncertainty due to mechanical tolerances from the manufacturing process. The licensee also included additional uncertainties related to irradiated fuel as described in the burnup credit methodology discussed in Westinghouse report WCAP-14416-NP-A. The licensee determined these uncertainties at the 95/95 probability/confidence level, using procedures described in the regulatory guidance of Reference 2. Because the licensee followed the appropriate regulatory procedures and used conservative values for their analyses, the NRC staff finds them acceptable for use.

3.4 SFP Rack Materials Evaluation

The Vogtle, Unit 1 SFP contains high-density spent fuel racks that utilize Boral® neutron absorbing panels as the poison material. These spent fuel racks have been previously utilized at the Maine Yankee Atomic Power Plant (MYAPP) and were approved for use at Vogtle in an NRC staff letter and accompanying safety evaluation dated June 29, 1998. For Vogtle, Unit 2, the spent fuel racks utilize Boraflex as the neutron absorbing panels. The licensee stated that Boraflex is not credited in the current criticality analysis, but credit for soluble boron is taken.

MYAPP began using spent fuel racks that utilized Boral® around 1976. Approximately 9 years later, MYAPP started to replace these racks with higher density racks. Some of the replacement racks used "new" Boral® plates, while the rest of the racks used Boral® plates that were "reclaimed" from the previous racks. The "new" and "reclaimed" Boral® plates are of different thicknesses with the "reclaimed" plates being thicker. The plates also contained different B_4C weight-percents. The "new" Boral® plates were nominally 50 weight-percent B_4C while the "reclaimed" Boral® plates were nominally 35 weight-percent B_4C.

As documented in NRC memorandums dated April 27 and March 24, 2005, the NRC staff requested the licensee to clarify if the racks were inspected to ascertain the physical state of the racks and tested for any degradation (e.g., blistering, etc.) prior to installation in the Vogtle SFP.

In its response dated May 3, 2005, the licensee stated that MYAPP had implemented a surveillance procedure once per cycle prior to each refueling outage that consisted of drag testing and visual inspection to monitor for signs of bulging. The last two surveillances performed at MYAPP showed no signs of swelling or bulging. Upon receiving the racks at Vogtle, visual inspections and drag testing of the racks were performed. No problems were noted. In addition, at the time the racks were installed in the SFP, information regarding Boral® blistering was not widespread; therefore, no additional testing was performed including verification of the B-10 content of the plates.

Given that blistering of the Boral® plates is now known to occur, as documented in NRC memorandum dated August 29, 2005, on June 16, 2005, the NRC staff requested the licensee to clarify whether it had any surveillance procedures in place, and if not, did it plan to implement a surveillance procedure to monitor for bulging and blistering.

In its response dated July 7, 2005, the licensee stated that the bulging of the racks was addressed by drilling vent holes to allow for the passivation gas to escape, thereby, preventing pressure buildup in the cell walls. Therefore, drag testing would not be performed since bulging is not expected to occur and this test would not provide any new information. Regarding blistering, the main concern is the reduction of the flux trap size and the potential impact on the criticality analysis. The flux trap region is defined as the area between adjacent fuel assemblies. To address the issue of blistering, the licensee explained that the thicker “reclaimed” panels were used in the criticality analyses. The thicker panel minimizes the amount of water between adjacent assemblies, which in turn acts to increase reactivity, thereby, introducing more conservatism into the calculations.

Recent test results of blistered coupons do not appear to negatively impact the neutron absorbing capability of the material, however, the long-term effects of blistering in the continued performance of Boral® is not yet known. To address this issue, as documented on April 27, 2005, the NRC staff asked the licensee whether they had a Boral® coupon surveillance program in place to monitor degradation and to ensure consistent material performance by monitoring the physical and chemical properties over time.

In its response dated May 3, 2005, the licensee stated that SNC does not have a Boral® coupon surveillance program at Vogtle and that there are no Boral® surveillance coupons currently installed in the SFP. The licensee explained that the analyses and results of the criticality analysis contain several conservatisms (e.g., minimum areal density and maximum thickness) that would offset the reactivity effects of blistering. In addition, the licensee stated that it continues to monitor issues regarding the application of Boral® in spent fuel racks through its Operating Experience and Corrective Actions Program in addition to monitoring the internal operating experience at another SNC plant that has a Boral® surveillance program.

As documented on August 29, 2005, the NRC staff requested the licensee to describe the Boral® experience at the other SNC plant, including the coupon tests performed and frequency and the similarity of the environment to that of Vogtle.

In its response dated July 7, 2005, the licensee stated that any information coming from the operating experience of the other SNC plant is considered part of the database of operating experience. The licensee also stated that it is a participant in the Electric Power Research Institute User's Group that follows industry issues with regards to Boral®. Being a member of this user's group provides the licensee with access to research data and industry operating experience information. The licensee reiterated that any issue regarding the use of Boral® at SFP will be included in the plant's Corrective Actions Program.

As part of its criticality analysis, the licensee stated in its application dated August 13, 2004, that an areal density of 0.0238 g/cm^2 was assumed for the criticality analysis. This is the same value used in the criticality analysis approved by the NRC staff in its safety evaluation dated June 29, 1998. As documented on April 27, 2005, the NRC staff asked the licensee to explain the technical basis for using the same areal density for spent fuel racks that have been exposed to radiation for many years at MYAPP and approximately 7 years at Vogtle, since tests had not been performed to verify the areal density of the panels prior to installation.

In its response dated May 3, 2005, the licensee stated that fabrication data were used to determine the areal density for both "new" and "reclaimed" Boral®, as demonstrated in calculation X6CKA.01 (Reference 4 of Enclosure 5 to the November 20, 1997, license amendment request). The NRC staff reviewed this calculation to determine if the assumptions taken for the areal density were still applicable for the current criticality analysis.

This calculation demonstrates that the B-10 areal density used in the criticality analysis is bounded by the real B-10 density of the "reclaimed" Boral® panels. The areal density was calculated by two methods. Both methods calculated areal density values significantly greater than the value of 0.0238 g/cm^2 used in the criticality analysis. The results for both methods demonstrate that the criticality assumptions are bounded by the manufacturing specifications of both types of panels. However, it was not clear to the NRC staff whether manufacturing tolerances were factored into the calculations. As documented on August 29, 2005, the NRC staff requested the licensee to discuss whether the manufacturing tolerances factored into the calculations.

In its response dated July 7, 2005, the licensee stated that all the values used in the calculations were minimum values after accounting for manufacturing tolerances, therefore, manufacturing tolerances were included in the calculations. In addition, the licensee stated that the value of 0.0238 g/cm^2 (which is the value of the areal density for the "new" Boral® panels) is the minimum requirement in the manufacturing specification confirmed in the Boral® manufacturer's certificate of compliance.

As documented on August 29, 2005, the NRC staff requested the licensee to confirm the receipt of the Boral® material certification from MYAPP for any testing and to discuss the results of the tests performed on the Boral® plates prior to its installation in Vogtle. In its response dated July 7, 2005, the licensee stated that for the "reclaimed" Boral®, SNC has the manufacturer's report of quantitative testing of Boral® samples that provided the necessary data for the Boral® thickness and B-10 areal density used in the criticality calculations. In addition, SNC has the specification developed by the rack fabricator for the "new" Boral® B-10 areal density requirements as well as the manufacturer's certificate of conformance. The licensee also stated that no testing of the Boral® was performed prior to installation.

The NRC staff has reviewed the licensee's request to continue crediting Boral® in its updated criticality analysis and all the responses to the requests for additional information. The NRC staff notes that in absence of a surveillance program, appropriate measures, as follows, should be taken to ensure consistent material performance.

As discussed above, the licensee does not have a coupon surveillance program, but appropriate measures have been taken to account for the possible degradation of the Boral® panels. In its criticality analysis, the licensee chose to use the maximum thickness of the two designs ("reclaimed" and "new"), as well as the minimum areal density. By modeling each spent fuel rack with the thicker Boral® panels, the amount of water between adjacent cells (the flux trap region) is minimized, thereby increasing reactivity. In addition, assuming the minimum areal density represents using the lowest amount of boron carbide available to control the criticality of the SFP. These assumptions introduce more conservatism into the criticality calculations. In addition, any issues emerging from the use of Boral® in the spent fuel racks will be monitored through its Operating Experience and Corrective Actions Program.

Based on the discussion provided above, the NRC staff finds that the licensee's assumptions to account for possible Boral® degradation in the criticality analysis and its continued monitoring of industry experience are appropriate measures for ensuring consistent material performance of the Boral® panels.

3.4.1 SFP Rack Material Evaluation Conclusion

The NRC staff reviewed the portions of the submittal addressing behavior of the materials used in the racks and the assumptions supporting the criticality analysis. Based upon its review of the information included in the submittal and the responses to the NRC staff's requests for additional information, the NRC staff finds that the neutron absorption material (Boral®) assumptions used to support the criticality analysis are appropriate in accounting for the possible degradation of the neutron poison panels, and are, therefore, acceptable.

3.5 Proposed Technical Specification Changes

SNC provided a descriptive list of requested changes to the Vogtle, Units 1 and 2 TSs. The actual marked-up and revised TSs are located in enclosure 1 of the August 31, 2004, application. The NRC staff reviewed each of these changes against the regulatory criteria described in Section 2 of this report and found them acceptable. The basis for the NRC staff's acceptance and a description of the review it performed is located in Section 3.0 of this report. The following is the descriptive list of proposed changes as provided by the licensee (Enclosure E1-1 of the August 31, 2004, application):

SNC proposes to revise TS 3.7.18, TS 4.3.1.1, and TS 4.3.1.2 to reflect the results of the revised analyses described above. In addition, the licensee included changes to the Bases for TS 3.7.17 and TS 3.7.18 to reflect the results of the revised analyses described above.

The changes to the TS pages have been reviewed by the NRC staff and are in accordance with NRC guidelines and compatible with the requested changes in Section 3.0 of this report and are, therefore, acceptable. The changes to the Bases are made in accordance with the TS Bases Control Program defined in TS 5.5.14. The NRC staff agrees that the TS Bases Control Program is the appropriate process for updating the affected TS Bases pages.

The licensee also identified an error in the Table of Contents for the TSs that occurred during the issuance of Amendments 130 and 109 for Vogtle, Units 1 and 2 respectively. During that amendment, the page number for Figure 5.5.6-1 changed from Page 5.5-22 to 5.5-23. However, at that time the Table of Contents was not revised to reflect the change to the TS page. The licensee has included a revised page vi to the TSs to correct the error in the Table of Contents. There is no change to the figure itself and the conclusions of the SER for Amendments 130 and 109 for Vogtle, Units 1 and 2, respectively, are not affected.

3.6 Summary

The NRC staff has reviewed the reports submitted by SNC, reflecting the update of the spent fuel rack criticality analyses performed, the methods used and the changes to the corresponding TSs for Vogtle, Units 1 and 2. Based on this review, we conclude that appropriate documentation was submitted and that the proposed changes satisfy the NRC staff positions and requirements in these areas. Therefore, the criticality analyses performed for Vogtle, Units 1 and 2 spent fuel storage racks are acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (69 FR 64990). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 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.

7.0 REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Appendix A, Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

2. L. Kopp (NRC), "Guidance on the Regulatory Requirements for criticality analysis of fuel storage at Light-Water reactor power plants," February 1998.

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Date: September 22, 2005

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