June 28, 2005

Mr. L. M. Stinson
Vice President - Farley Project
Southern Nuclear Operating
Company, Inc.
Post Office Box 1295
Birmingham, Alabama 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MC6987 AND MC6988)

Dear Mr. Stinson:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 169 to Renewed Facility Operating License No. NPF-2 and Amendment No. 161 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 17, 2005, as supplemented by letter dated June 13, 2005.

The amendments revise the TS Section 3.7, "Plant Systems," and Section 4.3, "Design Features," to establish cask storage area boron concentration limits and to restrict the minimum burnup of spent fuel assemblies associated with spent fuel cask loading operations.

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/
Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 169 to NPF-2
2. Amendment No. 161 to NPF-8
3. Safety Evaluation

cc w/encls: See next page

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Package Number: ML0518060219 Tech Spec No.: ML051860380
Amendment Number: ML0518060200 *No legal objection NRR-058

OFFICE	PDII-1/PM	PDII-1/LA	SRXB	IROB/SC	OGC	PDII-1/SC
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DATE	06/ 27 /05	06/ 29 /05	memo dated 06/21/05	06/ 28 /05	06/27/05	06/28/05

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Joseph M. Farley Nuclear Plant, Units 1 & 2

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SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MC6987 AND MC6988)

Date: June 29, 2005

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 169
Renewed License No. NPF-2

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated May 17, 2005, as supplemented by letter dated June 13, 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 set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 169, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 29, 2005

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 161
Renewed License No. NPF-8

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated May 17, 2005, as supplemented by letter dated June 13, 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 set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-8 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 161, are hereby incorporated in the license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 29, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 169

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

ATTACHMENT TO LICENSE AMENDMENT NO. 161

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

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 vertical lines indicating the areas of change.

Remove	Insert
iii	iii
-	3.7.17-1
-	3.7.17-2
-	3.7.18-1
-	3.7.18-2
4.0-3	4.0-3
4.0-4	4.0-4
4.0-5	4.0-5
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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 169 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 161 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, Commission) dated May 17, 2005 (Reference 1), as supplemented by letter dated June 13, 2005 (Reference 2), the Southern Nuclear Operating Company, Inc. (SNC, the licensee), submitted a request for changes to the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Technical Specifications (TSs). The proposed changes would revise the TS Section 3.7, "Plant Systems," and Section 4.0, "Design Features," to establish cask storage area boron concentration limits and to restrict the minimum burnup of spent fuel assemblies associated with spent fuel cask loading operations. The licensee's amendment request would ensure subcritical conditions were maintained in the spent fuel pool (SFP) during dry cask loading operations by relying on a realistically conservative fuel burnup credit.

The June 13, 2005, letter provided clarifying information that did not change the scope of the amendment request as originally noticed, and did not change the NRC staff's initial proposed no significant hazards consideration as published in the *Federal Register*.

The FNP TSs currently permit the licensee to store 1407 fuel assemblies in the SFP. However, since the FNP SFP was not designed with the storage capacity necessary for all the spent fuel generated over the full term of the renewed facility's operating license or for the permanent storage of the plant's spent fuel following the cessation of operations, the cask pit area provides plant operators with a safe location to load storage and transportation casks. SNC is planning to operate an independent spent fuel storage installation (ISFSI) facility at FNP in accordance with the general license provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, Subpart K, using the Holtec HI-STORM 100 Cask System Multi-Purpose Canister (MPC)-32. SNC intends to load spent fuel into the MPC-32 in its SFP cask pit area for subsequent removal and dry storage on the ISFSI.

On March 23, 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-05, "Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel

Storage Installations." (Reference 3) The NRC issued RIS 2005-05 for three purposes: (1) to alert addressees to findings at pressurized-water reactor facilities suggesting that the spent fuel pool licensing and design bases and applicable regulatory requirements may not be met during loading, unloading, and handling of dry casks in the SFPs; (2) to emphasize the importance of maintaining subcritical conditions for spent fuel storage in moderated environments; and (3) to encourage addressees to review the current SFP and ISFSI licensing and design bases at their facilities to ensure compliance during dry cask loading, unloading, and handling operations. Based on SNC's review of RIS 2005-05, the licensee determined that it required a license amendment to facilitate the loading, unloading, and handling of dry storage casks in its SFP.

To ensure its continued compliance with NRC regulations governing the safe handling of irradiated fuel in the SFP, the licensee proposed a number of changes to the FNP TSs. Section 2.2 of this report provides a descriptive summary of the proposed changes and Section 3.0 provides the NRC staff's Technical Evaluation of the proposed changes.

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements and Review Documents

10 CFR Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," (Reference 4) 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.

10 CFR Section 50.68, "Criticality accident requirements," (Reference 5) provides the NRC regulatory requirements for maintaining subcritical conditions in SFPs. Although SNC has not committed to nor demonstrated compliance with all of the regulatory requirements of 10 CFR 50.68, the current FNP SFP licensing basis, as approved by the NRC in Amendments 133 and 125 (Reference 6) to the FNP, Units 1 and 2 renewed facility operating licenses, respectively, is consistent with the relevant regulatory requirements of 10 CFR 50.68 for ensuring subcritical conditions are maintained in the SFP. Therefore the NRC staff finds it appropriate and acceptable to use the applicable regulatory criteria from 10 CFR 50.68 as acceptance criteria for the review of the proposed changes to the FNP TSs governing cask loading, unloading, and handling operations in the SFP.

The 10 CFR 50.68 acceptance criteria for criticality prevention in the SFP that are applicable to the licensee's proposed amendment are the following:

1. Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water;

2. The effective 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; and

3. 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.

Under 10 CFR 72.124, "Criteria for nuclear criticality safety," (Reference 7) the NRC regulates dry cask storage activities to ensure that subcriticality is maintained during the handling, packaging, transfer, and storage of spent fuel assemblies. The NRC regulations for dry cask criticality prevention rely on favorable geometric configurations and fixed neutron absorbers. However, unlike 10 CFR 50.68, the 10 CFR Part 72 regulations for criticality prevention in dry casks allow licensees to credit the SFP soluble boron for maintaining subcritical conditions during cask loading, unloading, and handling operations in the SFP. Therefore, many cask designs have incorporated soluble boron credit in lieu of a burnup credit as a means of increasing dry cask storage capacity while maintaining subcritical conditions. FNP's amendment request proposes to demonstrate that it can satisfy the applicable 10 CFR 50.68 criticality prevention requirements, with a burnup credit, during cask loading, unloading, and handling operations in the SFP.

The NRC defined acceptable methodologies for performing SFP criticality analyses in three documents:

1. NUREG-0800, Standard Review Plan, Section 9.1.2, "Spent Fuel Storage," Draft Revision 4; (Reference 8)

2. Proposed Revision 2 to Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," (Reference 9); and

3. Memorandum from L. Kopp (NRC) to T. Collins (NRC), "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants" (Reference 10).

The NRC staff used the guidance contained in these documents to assist in its review of the licensee's amendment request.

The requirements of 10 CFR 50.36 established the categories of items for inclusion in the TS but not the particular requirements of the plant's TS. Guidance was provided for the specific contents of the TS in the "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement), 58 FR 39132 (July 22, 1993). The Final Policy Statement established four criteria for determining the items required for inclusion in the TS, as follows:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition for a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that

either assumes the failure of or presents a challenge to the integrity of the fission product barrier.

Criterion 4 A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

These criteria have been codified in 10 CFR 50.36. See Final Rule, "Technical Specifications," 60 FR 36593 (July 19, 1995). As a result, TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement are included in the TS.

2.2 Description of Proposed Technical Specification Changes

In Enclosures 3 and 4 of Reference 1, SNC provided marked-up TSs and corresponding bases pages. The NRC staff reviewed each of the TS changes against the acceptance criteria described in Section 2.1 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 the changes proposed by the licensee in Enclosures 3 and 4 of Reference 1:

1. Table of Contents: SNC revised the TS Table of Contents to include listings for new Limiting Condition for Operation (LCO) Sections 3.7.17, "Cask Storage Area Boron Concentration - Cask Loading Operations," and 3.7.18, "Spent Fuel Assembly Storage - Cask Loading Operations."

2. New LCO 3.7.17: SNC proposed the addition of a new TS governing the boron concentration in the cask storage area. The applicability of this TS is restricted to "whenever any fuel assembly is stored in the cask storage area." Additionally, SNC included proposed actions if the LCO is not met and surveillance requirements for sampling.

3. New LCO 3.7.18: SNC proposed the addition of a new TS to control the loading of spent fuel assemblies into casks in the cask storage area. The applicability of this TS is identical to that for 3.7.17. Additionally, SNC included proposed actions if the LCO is not met and surveillance requirements for verifying the design requirements of the assemblies are met prior to their loading into spent fuel storage casks.

4. New TS Figure 3.7.18-1: SNC proposed the addition of this figure to support the new LCO 3.7.17 loading requirements. The figure provides a curve of acceptable fuel assembly burnups as a function of initial Uranium-235 enrichment. Spent fuel assemblies with a burnup greater than the limits proposed by the curve would be acceptable for storage in a spent fuel cask.

5. New Design Feature Section 4.3.1.3: SNC proposed to add design feature requirements describing the safety limits and criteria that govern loading of spent fuel assemblies into storage casks in the SFP. These requirements are modeled on those in current Design Feature Section 4.3.1.2 that governs spent fuel storage racks.

3.0 TECHNICAL EVALUATION

In determining the acceptability of the 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) the storage configuration and limitations proposed. 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 acceptance criteria were developed and could be maintained in the FNP SFPs during cask loading, unloading, and handling operations.

3.1 Computer Codes

The licensee performed the analysis of the reactivity effects for the MPC-32 with SCALE version 4.4 (Reference 11) with the 44 and 238 group ENDF/B-V neutron cross section libraries. SCALE version 4.4 includes the KENO V.a code, a three-dimensional Monte Carlo criticality code. The KENO V.a code was benchmarked against criticality experiments under conditions that reflect the variables for fuel storage in the MPC-32. The critical benchmark experiments included 19 Babcock and Wilcox experiments (Reference 12) carried out in support of the Close Proximity Storage of Power Reactor Fuel and 11 experiments from the Pacific Northwest Laboratory (PNL) experimental program (Reference 13). The NRC has previously accepted the use of this data for benchmarking the KENO V.a code under storage conditions similar to those proposed in the SNC amendment request (References 10 and 14). The experimental data is sufficiently diverse to establish that the method bias and uncertainty will apply to MPC-32 under the proposed storage conditions in the FNP SFP. The licensee determined the KENO V.a code calculation (methodology) bias is 0.0283 with a 95/95 bias uncertainty of +/- 0.0316 using the 44-group ENDF/B-V neutron cross section library. Additionally, the licensee determined the reactivity effect (delta-k) for each manufacturing tolerance of the fuel assemblies and storage lattice based on a full-scale model representation of the MPC-32.

In addition to using the KENO V.a code to perform the criticality analyses, the licensee employed the Discrete Integral Transport (DIT) code to perform the fuel depletion analyses that were used to develop the proposed TS Figure 3.7.18-1. DIT performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly to determine the isotopic composition of the spent fuel as a function of fuel burnup and initial feed enrichment. 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. In accordance with NRC guidance documents, the licensee applied a five percent burnup measurement uncertainty to ensure that the results obtained for the depletion analysis were conservative.

The NRC staff reviewed the licensee's application of the codes to determine whether each could reasonably calculate, based on conservative assumptions and inputs, the appropriate parameters necessary to support the maximum k_{eff} analyses. In Reference 10, the staff stated that KENO V.a was an acceptable computer code for the analysis of fuel assemblies stored in the SFP. Additionally, in Reference 10, the NRC staff stated that the Babcock and Wilcox series of criticality experiments, as described in Reference 12, provided an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control and close-packed arrays of fuel. Additionally, the NRC has previously accepted the use of the PNL

experimental database employed by the licensee for performing benchmarking calculations of the codes used in the analysis (References 14, 15, and 16). Therefore, the NRC staff concludes that the licensee's use of the KENO V.a code for calculation of the nominal k_{eff} was appropriate since it was benchmarked against experimental data which adequately reflects the proposed fuel assembly and storage conditions proposed in the SNC amendment request. Additionally, the NRC staff has previously approved the use of the DIT code for performing depletion analyses in support of burnup credit for spent fuel storage (References 14, 15, and 16); therefore, the NRC staff finds that the licensee's use of the DIT code was acceptable for performing the fuel depletion analyses.

3.2 Methodology

In accordance with the guidance contained in References 8, 9, and 10, the licensee performed criticality analyses of the MPC-32 under fully loaded conditions. The licensee employed a methodology which combines a worst-case analysis based on the bounding fuel and MPC-32 conditions, with a sensitivity study using 95/95 analysis techniques. The major components in this analysis were a calculated (nominal) k_{eff} based on the limiting fuel assembly and storage configuration, SFP temperature and code biases, and a statistical sum of 95/95 uncertainties and worst-case delta-k manufacturing tolerances.

For added conservatism, SNC assumed a bounding upper subcriticality limit. Instead of designing the loading configuration for the MPC-32 based on maintaining k_{eff} less than 1.0, as is required in NRC regulations, SNC chose to determine the limiting loading configuration based on an upper subcriticality limit of 0.97 including all applicable biases and uncertainties. This effectively incorporates added margin into the calculations performed to demonstrate compliance with NRC acceptance criteria.

In performing its criticality analysis, the licensee first calculated a k_{eff} based on nominal MPC-32 loading conditions using the KENO V.a code. SNC calculated this nominal k_{eff} for each fuel assembly design, fuel enrichment, and storage configuration that is considered in the scope of the MPC-32 loading and storage at FNP. Since the licensee only intends to load the Westinghouse 17 x17 Standard and the Westinghouse 17 x 17 Optimized Fuel Assembly (OFA) designs, it performed sensitivity calculations of each of these assemblies loaded in the MPC-32 to determine the most reactive fuel assembly design under normal and accident conditions. SNC determined that the Westinghouse 17 x17 Standard Fuel Assembly design was the most reactive fuel assembly under normal conditions and that the Westinghouse 17 x 17 OFA design was limiting under accident conditions. SNC then applied the most limiting fuel design in its normal and accident analyses to ensure that bounding k_{eff} values were determined.

In addition to determining the bounding fuel design, the licensee included the effects of the bounding SFP temperatures in the determination of the nominal k_{eff}. The licensee used the minimum and maximum permissible design basis SFP temperatures and corresponding water densities to determine which resulted in the most limiting nominal k_{eff}. In Reference 2, SNC provided the results of sensitivity calculations performed over the range of design basis SFP temperatures. The results showed that optimum moderation occurred under the full density conditions assumed in the analysis. Therefore, with respect to optimum moderation, the licensee ensured that its nominal k_{eff} criticality analyses is bounding under normal storage conditions. This is consistent with NRC acceptance criteria and guidance documents.

To the calculated k_{eff}, the licensee added the methodology bias. As stated in the description of the KENO V.a code, the licensee determined the methodology bias from the critical benchmark experiments.

Additionally, to determine the maximum k_{eff}, the licensee performed a statistical combination of the reactivity effects for code and methodology uncertainties, manufacturing tolerances, and burnup uncertainties. The code and methodology uncertainties account for the mean calculational variance and uncertainty in the benchmarking of the KENO V.a code. The licensee determined this uncertainty to a 95/95 threshold which is consistent with NRC acceptance criteria and guidance documents.

In addition to including the code uncertainty, the licensee performed analyses to determine appropriate and conservative fuel and storage cask mechanical tolerances as well as including a tolerance for eccentric positioning of the fuel assemblies in the storage cells. For each tolerance, the licensee calculated a delta-k between the nominal condition and the most limiting tolerance condition. For the fuel rod manufacturing tolerances the dominating contributors are the fuel enrichment and theoretical density. For each of these contributors, the licensee included bounding and conservative effects that result in maximizing the delta-k. In Reference 2, SNC provided additional information to support its conclusions that the theoretical densities assumed result in conservative calculations of the delta-k under burnup conditions. Likewise, for the storage cask fabrication tolerances, the licensee included conservative and bounding tolerances on key parameters, such as the cell pitch and cell inner dimension that result in maximizing the delta-k. By using the most limiting tolerance conditions, the licensee calculated the highest reactivity effect possible. This results in conservative margin since the tolerances will always bound the actual parameters. In addition to manufacturing tolerances, SNC analyzed eccentric positioning of fuel assemblies in the MPC-32 lattice cells. SNC determined that eccentric positioning of the assemblies such that the center-to-center pitch was at its minimum resulted in a minor increase in the k_{eff}. SNC appropriately included the bounding delta-k from eccentric positioning in its tolerance calculations.

Finally, in lieu of performing detailed burnup uncertainty analyses, the licensee chose to apply a 5 percent burnup measurement uncertainty in accordance with NRC guidance documents (Reference 10). This uncertainty, in conjunction with the use of the other conservative assumptions and inputs, assures that NRC regulatory requirements are satisfied.

The licensee's proposed TS changes place considerable emphasis in the criticality analyses on a burnup credit; therefore, the accurate determination of the burnup profile is essential to ensure the acceptance criteria for k_{eff} are satisfied. As previously stated, the licensee employed the DIT code for determining the appropriate burnup credit. For a given spent fuel assembly, the fuel burnup is a function of axial position. In performing the depletion analysis, the licensee analyzed the burnup in discrete axial zones to ensure that the axial "end-effect" was adequately captured. SNC divided the fuel assembly into several axial zones with each zone assumed to be uniform in burnup. Additionally, SNC made the top axial zone sufficiently small (6 inches) to capture the steep burnup gradient with axial position in this region of the fuel assembly. The axial burnup profile employed in this analysis was based on the most limiting axial burnup shape from a Department of Energy (DOE) topical report (Reference 17). The NRC has approved similar axial profile models based on the DOE database in other license amendments (References 14 and 16). In order to generate the isotopic concentrations for each segment of the axial profile, appropriate fuel and moderator temperatures and soluble boron concentrations

that both reflect historical operating conditions at FNP as well as represent appropriately conservative values intended to maximize the residual reactivity of the spent fuel assemblies must be used in the depletion analysis. In Reference 2, SNC provided additional information that demonstrated the values chosen for these parameters satisfied these criteria. The data provided by SNC demonstrated that the values used in the burnup credit analysis represented conservative assumptions for the fuel and moderator temperatures and soluble boron concentrations relative to the historical operating conditions at FNP. Therefore, the NRC staff finds that the methodology employed and the assumptions used to perform the burnup credit analysis are acceptable.

In addition to a burnup credit, SNC proposed to credit the fixed neutron absorbers in the MPC-32. The MPC-32 contains Metamic neutron absorbers loaded between fuel assemblies in the cask lattice structure. Metamic acts as a poison, absorbing neutrons and holding down the k_{eff} in the MPC-32. In the criticality analyses, SNC credited the minimum areal density of the Boron-10 in the Metamic panels. Section 3.2.5.2 of the Hi-Storm Certificate of Compliance 1014 (Reference 18), Appendix B requires that the Boron-10 loading be greater than or equal to 0.0310 g/cm^2. This represents a TS minimum acceptable areal density for Metamic in the MPC-32. SNC conservatively applied the minimum areal density in its criticality analyses. Since SNC applied the worst case condition, it did not include an associated delta-k uncertainty for the areal density. The NRC staff finds that SNC used an appropriately conservative and limiting value for the minimum areal density of boron-10 in the Metamic neutron absorbers.

Once the reactivity effects for each of the tolerances and uncertainties were determined, the licensee statistically combined these results in accordance with the guidance contained in Reference 10. The NRC staff reviewed the licensee's methodology for calculating each of 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 for calculating the maximum k_{eff}, that combined a worst-case analysis based on the bounding fuel and MPC-32 conditions with a sensitivity study using 95/95 analysis techniques, as described in the above paragraphs, to be acceptable.

3.3 Proposed Storage Configuration

The primary purpose of the licensee's amendment request was to gain the NRC staff's approval for a proposed storage configuration within the MPC-32 during loading, unloading, and handling operations in the SFP. The licensee's proposed TS LCO 3.7.18 would permit unrestricted storage of spent fuel assemblies in the MPC-32 provided each assembly satisfied minimum burnup requirements as a function of initial enrichment. The minimum burnup requirements are provided in proposed TS Figure 3.7.18-1.

The first step in the process for loading an MPC-32 at FNP involves placing the canister in the SFP. The FNP cask storage area is physically separated from the spent fuel in the pool by a transfer canal used during refueling operations. In its criticality analyses, SNC assumed a 1 foot spacing between the MPC-32 and other fissile materials. This spacing is assumed due to the physical restrictions encountered during loading of canisters in the cask pit area. In its KENO V.a model, SNC incorporated a standard 2 foot reflector region in both the axial and radial directions intended to maximize the k_{eff} of the fuel stored in the MPC-32. The spacing employed in the criticality analyses will effectively preclude neutron coupling between the fuel stored in the MPC-32 and fuel stored in the SFP. The NRC staff finds that the spacing

assumed in the criticality analysis appropriately reflects the storage conditions at FNP and results in a bounding determination of the maximum k_{eff}.

In determining the acceptable burnup versus enrichment curves, SNC used the codes and methodologies described in Sections 3.1 and 3.2, respectively, of this report. In addition to the conservative assumptions previously described, SNC did not take credit for long-term cooling in the fuel assemblies. Following irradiation, the residual reactivity of spent fuel will decrease for approximately 100 years due to the decay of fissile nuclides and build-up of neutron absorbing nuclides. This will result in a lower k_{eff} in the MPC-32 under SNC's proposed storage configuration. Since SNC did not include a credit for the beneficial long-term cooling effects, the burnup versus enrichment curves contain additional conservative margin to the NRC acceptance criteria.

TS Figure 3.7.18-1 provides the fuel assembly burnup limit requirements for cask storage. This figure depicts the limiting burnup as a function of initial fresh fuel enrichment required to load spent fuel assemblies into the MPC-32 at FNP. An assembly with a burnup greater than the limits on the curve may be loaded into the MPC-32 without restrictions on its storage configuration. In developing this burnup versus enrichment curve, SNC performed KENO V.a analyses, as described previously, based on limiting storage conditions. To ensure that the NRC acceptance criteria were satisfied, SNC set its target value of k_{eff} at its self-imposed limit of 0.970 minus the magnitude of the limiting analytical biases and uncertainties. The sum of the biases and uncertainties was conservatively calculated to be 0.016. Therefore, each data point on the burnup versus enrichment curve is based on a limiting k_{eff} value of 0.954. The licensee developed a third order polynomial of limiting assembly burnup as a function of initial enrichment from this data. This polynomial will be used to determine the acceptability of assemblies for loading into the MPC-32. The NRC staff finds this to be acceptable.

In addition to analyzing the nominal MPC-32 loading configurations, the licensee performed detailed accident analyses. The accidents analyzed included the following: 1) a dropped fresh fuel assembly on top of the MPC-32; 2) a misloaded fresh fuel assembly outside of the MPC-32; 3) MPC-32 assembly-to-assembly pitch reduction due to seismic event; 4) MPC-32 water temperature greater than 180 EF; and 5) misloaded fresh fuel assembly into a MPC-32 location. SNC determined that the bounding accident was the misloading of a fresh fuel assembly into a MPC-32 location. Since the NRC staff does not require a licensee to assume two independent accidents occurring simultaneously, SNC calculated the amount of soluble boron required to mitigate the consequences of this accident. SNC determined that the 269 ppm of soluble boron would be required to compensate for the reactivity increase caused by the worst-case misloading event and maintain the k_{eff} less than 0.95. Since the new proposed TS LCO 3.7.17 will require the minimum cask storage area boron concentration to be greater than or equal to 2000 ppm, the NRC staff finds that sufficient soluble boron will be available to preclude an inadvertent criticality event for this and all less severe accidents.

In addition to the fuel handling accidents described above, the NRC staff requires licensees to perform a boron dilution analysis. During MPC-32 loading operations, the cask storage area is connected to the SFP. In Reference 6, the NRC staff approved the licensee's boron dilution analysis for the spent fuel pool. This analysis excluded the volume of water located in the cask storage area. The previous analysis remains bounding since the addition of cask storage area volume will result in longer dilution times. Additionally, since the licensee's criticality analyses are based on maintaining the k_{eff} in the MPC-32 less than 0.970, an inadvertent criticality

resulting from a boron dilution to pure water conditions is considered unlikely. Therefore, the NRC staff finds that the license's criticality analysis demonstrates adequate defense-in-depth and safety margins for boron dilution events.

3.4 TECHNICAL SPECIFICATIONS

The TS proposed by SNC were reviewed against the criteria specified in Section 2.1 of this report. The NRC staff finds that (a) the concentration of dissolved boron in the cask storage area, and (b) the configuration of fuel assemblies in the cask storage area satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii). Accordingly, the NRC staff finds SNC's proposed additions to the TS, as discussed above, to be acceptable.

3.5 SUMMARY

The NRC staff reviewed the effects of the proposed changes using the appropriate requirements of 10 CFR 50.68 and GDC 62. Based on its review of the criticality analyses supporting the proposed changes, the NRC staff finds that SNC employed realistically conservative assumptions, inputs, and methodologies in every step of the analysis. The results of the criticality analyses demonstrate that the proposed changes provide reasonable assurance that, under both normal and accident conditions, the licensee would be able to operate the plant safely and comply with the NRC regulations. Therefore, the NRC staff finds the proposed amendments to be acceptable.

Nothing in the approval of these amendments is intended or authorized to replace or supercede any requirements of the Holtec HI-STORM 100 Certificate of Compliance (CoC) 1014, Amendment 2 (Reference 19). SNC is required to comply with all of 10 CFR Part 72 approved TSs and limitations in CoC 1014.

The NRC staff also notes that, in addition to the proposed TS changes and additions described in Enclosures 3 and 4 of Reference 1, SNC provided revised TS Bases to be implemented with the TS change. The NRC staff finds no inconsistencies between the Bases and the proposed TS changes. SNC's TS Bases Control Program is the appropriate process for updating the affected TS Bases pages.

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

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The amendment has been evaluated against the three standards in 10 CFR 50.92(c). In its analysis of the issue of no significant hazards consideration, as required by 10 CFR 50.91(a), the licensee has provided the following:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Cask loading operations will not require any physical changes to Part 50 structures, systems, or components, nor will their performance requirements be altered. The potential to handle a spent fuel cask was considered in the original design of the plant. Therefore, the response of the plant to previously analyzed Part 50 accidents and related radiological releases will not be adversely impacted, and will bound those postulated during cask loading activities in the cask storage area. Accordingly, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Existing fuel handling procedures and associated administrative controls remain applicable for cask loading operations. Additionally, the soluble boron concentration required to maintain $k_{eff} \leq 0.95$ for postulated criticality accidents associated with cask loading operations was also evaluated. The results of the analyses, using a methodology previously approved by the NRC, demonstrate that the amount of soluble boron required to compensate for the positive reactivity associated with these postulated accidents (659 ppm) remains well below the existing spent fuel pool minimum boron concentration limit of 2000 ppm. Accordingly, the same limit has been proposed for cask loading operations in the cask storage area. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Does the proposed change involve a significant reduction in a margin of safety?

   An NRC approved methodology was used to perform the criticality analysis which provides the basis to incorporate a new burnup versus enrichment curve into the plant Technical Specifications to ensure criticality requirements are met during spent fuel cask loading. Accordingly, the existing minimum boron concentration limit for the spent fuel of 2000 ppm will continue to remain bounding during cask loading operations. Existing criticality limits will also be maintained should it be postulated that the spent fuel pool be flooded when connected to the cask storage area with unborated water ($k_{eff} < 1.0$) or should it become flooded with borated water to 400 ppm ($k_{eff} \leq 0.95$) during cask loading operations. This determination accounts for uncertainties at a 95-percent/95-percent probability/confidence level. Proposed Technical Specification 3.7.17 requires that the spent fuel transfer canal gate and the cask storage area gate be open except when moving the spent fuel cask into or out of the cask storage area. The cask storage area will be isolated from the spent fuel pool volume during movement of the cask into and out of the cask storage area. Due to the minimal time that spent fuel will be stored in the cask storage area with the cask storage area isolated from the spent fuel pool volume, a boron dilution event is not considered credible while the cask storage area is isolated. However, should it be postulated that a boron dilution event does occur during this time period, k_{eff} will remain less than 1.0 should

the cask storage area become fully flooded with unborated water. Therefore, there will not be a significant reduction in a margin of safety.

Based upon the preceding information, SNC has concluded that the requested license amendment does not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis, and based on this review, has determined that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff finds that the amendment request involves no significant hazards consideration.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change the surveillance requirements. 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 (70 FR 30148). Additionally, the Commission has made a final no significant hazards consideration with respect to this amendment. 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.

7.0 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.

8.0 REFERENCES

1. Letter from L.M. Stinson (SNC) to NRC, “Joseph M. Farley Nuclear Plant Technical Specifications Revision, Spent Fuel Cask Loading Requirements,” dated May 17, 2005, ADAMS Accession Nos. ML051390362, ML051390363, and ML051390369 .

2. Letter from L.M. Stinson (SNC) to NRC, “Joseph M. Farley Nuclear Plant Technical Specifications Revision, Spent Fuel Cask Loading Requirements, Response to Draft Request for Additional Information,” dated June 13, 2005, ADAMS Accession No. ML051660286.

3. NRC Regulatory Issue Summary 2005-05, “Regulatory Issues Regarding Criticality Analyses for Spent Fuel Pools and Independent Spent Fuel Storage Installations,” dated March 23, 2005, ADAMS Accession No. ML043500532.

4. 10 CFR, Part 50 Appendix A, General Design Criteria 62, “Prevention of criticality in fuel storage and handling.”

5. 10 CFR, Section 50.68, “Criticality accident requirements.”

6. Letter from J.I. Zimmerman (NRC) to D.N. Morey (SNC), “Issuance of Amendments - Joseph M. Farley Nuclear Plant, Units 1 and 2 (TAC Nos. M99136 and M99137),” dated January 23, 1998, ADAMS Accession No. ML013130226.

7. 10 CFR Section 72.124, "Criteria for nuclear criticality safety."

8. NUREG-0800, Standard Review Plan, Section 9.1.2, “Spent Fuel Storage,” Draft Revision 4, April 1996.

9. Proposed Revision 2 to Regulatory Guide 1.13, “Spent Fuel Storage Facility Design Basis,” December 1981.

10. NRC Memorandum from L. Kopp to T. Collins, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” dated August 19, 1998.

11. NUREG/CR-0200, “SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,” distributed by the Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1998.

12. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979

13. S.R. Bierman and E.D. Clayton, “Criticality Experiments with Subcritical Clusters of 2.35 Wt% 235U Enriched UO2 Rods in Water at a Water-to-Fuel Volume Ratio of 1.6,” NUREG/CR-1547, PNL-3314, July 1980.

14. Letter from R.B. Ennis (NRC) to J.A. Price (Dominion Nuclear Connecticut, Inc.), “Millstone Power Station, Unit No. 2 - Issuance of Amendment Re: Spent Fuel Pool Requirements (TAC No. MB3386),” dated April 1, 2003, ADAMS Accession No. ML030910485.

15. Letter from G.S. Vissing (NRC) to R.C. Mecredy (Rochester Gas and Electric Corporation), “R.E. Ginna Nuclear Power Plant - Amendment Re: Revision to the Storage Configuration Requirements Within the Existing Storage Racks and Taking Credit for a Limited Amount of Soluble Boron (TAC No. MA8443),” dated December 7, 2000, ADAMS Accession No. ML003761578.

16. Letter from B. Benney (NRC) to G.M. Rueger (Pacific Gas and Electric Company), “Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment Re: Credit for Soluble Boron in the Spent Fuel Pool Criticality Analysis (TAC Nos. MB2982 and MB2984), “dated September 25, 2002, ADAMS Accession No. ML022610080.

17. DOE Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages,” DOE/RW-0472 Rev. 2, September 1998.

18. Holtec HI-STORM 100 Certificate of Compliance 1014, Amendment 2, Appendix B, Technical Specification Section 3.2.5.2, ADAMS ML051580527.

19. Holtec HI-STORM 100 Certificate of Compliance 1014, Amendment 2, dated June 7, 2005, ADAMS Accession No. ML051580446.

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Date: June 29, 2005