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

August 1, 2014

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
Post Office Box 1295, Bin - 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (VEGP) – REQUEST FOR ADDITIONAL INFORMATION ON NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION (TAC NOS. ME9472 AND ME9473)

Dear Mr. Pierce:

By letter dated August 31, 2012, as supplemented on September 13, 2013 May 2 and July 22, 2014, Southern Nuclear Operating Company (SNC) submitted a license amendment request to revise the licensing basis for the Vogtle Electric Generating Plant, Units 1 and 2, to implement the regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) § 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." The Nuclear Regulatory Commission staff finds that additional information is needed as set forth in the Enclosure.

Sincerely,

Robert Martin

Robert Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure: Request for Additional Information

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

REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

I Introduction

By letter dated August 31, 2012, as supplemented on September 13, 2013, May 2 and July 22, 2014, (Agencywide Documents Access and Management System Accession Nos. ML12248A035, ML13256A306, ML14122A364 and ML14003A252, respectively, Southern Nuclear Operating Company (SNC) submitted a license amendment request (LAR) to revise the licensing basis for the Vogtle Electric Generating Plant, Units 1 and 2 (VEGP), to implement the regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) § 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." The Nuclear Regulatory Commission (NRC) staff finds that additional information is needed as discussed below.

The licensee's LAR proposed to add license conditions (LCs) for the VEGP to allow for the voluntary implementation of 10 CFR 50.69. As indicated in § 50.69, a licensee may voluntarily comply with § 50.69 as an alternative to compliance with the following requirements for certain SSCs; (i) 10 CFR part 21, (ii) a portion of § 50.46, (iii) § 50.49, (v) certain requirements of § 50.55a, (vi) §50.65, (vii) § 50.72, (viii) § 50.73, (ix) Appendix B to Part 50, (x) certain containment leakage testing requirements, and (xi) certain requirements of Appendix A to part 100.

In the *Federal Register* notice issuing 10 CFR 50.69 on November 22, 2004 (69 FR 68008-68048), the following is stated:

> It is important to note that this rulemaking effort, while intended to ensure that the scope of special treatment requirements imposed on SSCs is risk-informed, is not intended to allow for the elimination of SSC functional requirements or to allow equipment that is required by the deterministic design basis to be removed from the facility (i.e., changes to the design of the facility must continue to meet the current requirements governing design change; most notably § 50.59). Instead, this rulemaking should enable licensees and the staff to focus their resources on SSCs that make a significant contribution to plant safety by restructuring the regulations to allow an alternative risk- informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., "reasonable confidence") that these SSCs will satisfy functional requirements. However, continued maintenance of the health and safety of the public will depend on effective implementation of § 50.69 by the licensee or applicant applying the rule at its nuclear power plant.

Enclosure

II Regulatory Requirement

The regulations in 10 CFR 50.91, "Notice for public comment; State consultation," require licensees, as stated in § 50.91(a)(1), to submit a No Significant Hazards Consideration (NSHC) analysis using the standards in § 50.92 "Issuance of amendment," with each LAR. The standards of § 50.92(c) are as follows:

(c) The Commission may make a final determination, under the procedures in § 50.91, that a proposed amendment to an operating license or a combined license for a facility or reactor licensed under §§ 50.21(b) or 50.22, or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(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 NRC staff issued Regulatory Issue Summary (RIS) 2001-22, "Attributes of a Proposed No Significant Hazards Consideration Determination" (ADAMS ML011860215) to provide guidance on preparing a NSHC analysis for a license amendment request. The RIS should be read in full for complete information, but selected aspects of the RIS are noted as follows;

(a) The RIS states that; "Licensees are required by 10 CFR 50.91(a)(1) to submit an NSHC analysis using the standards in 10 CFR 50.92 along with each request for a license amendment. 10 CFR 50.91(a)(2) requires the NRC to publish a notice of each proposed amendment and the staff's proposed determination, under the standards in 10 CFR 50.92, in the *Federal Register*. This notice provides the public an opportunity to comment or request a hearing on the proposed amendment request. Fundamental to an NSHC analysis is a discussion of whether the proposed change would significantly affect the current plant design, operation, or analyses using the standards of 10 CFR 50.92."

(b) The following RIS guidance should be considered for the subject LAR:

- State the three criteria in 10 CFR 50.92(c) separately and provide a separate analysis for each criterion.
- Identify previously evaluated accidents that are affected by the proposed change and explain why any change in the probability, consequences, or margins of safety is or is not significant.
- Give only information required to address each criterion, not the information required to demonstrate the acceptability of the proposed amendment. Be clear and concise.

(c) The RIS also provides additional guidance as follows:

For the first standard, "... if the proposed change increases the likelihood of the malfunction of an SSC, the potential impact on analyzed accidents should be considered (e.g., an increased likelihood of an SSC [structure, system or component] malfunction may increase the probability or consequences of an accident). If there is no impact on previously evaluated accidents, explain why. Discuss the differences in the probability and consequences of these accidents (or the bounding scenario) before and after the change and whether the differences are significant. If the change is not considered significant, explain why. ..."

For the second standard, "... Then determine whether the proposed change will create the possibility of a new or different kind of accident due to credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases. [This new accident would have been considered a design basis accident in the FSAR had it been previously identified.] A new initiator of the same accident is not a different type of accident. Finally, the accident must be credible within the range of assumptions previously applied (e.g., random single failure, loss of offsite power, no reliance on nonsafety-grade equipment.)

For the third standard, "... Licensees should identify the safety margins that may be affected by the proposed change and review the conservatism in the evaluation and analysis methods that are used to demonstrate compliance with regulatory and licensing requirements. The safety margin before the change should be compared to the margin after the proposed change to determine if the amendment will reduce the margin, and if the change is significant. ..."

III Evaluation and Request for Additional Information

Section 4.1 of SNC's LAR, "Significant Hazards Consideration" is not consistent with the guidance noted above. SNC's information to demonstrate the acceptability of the proposed amendment is included in the discussion for each standard[1] when it only needs to be included once at the beginning of the NSHC discussion. The discussion for the first standard addresses safety margins, which is the topic of the third standard and doesn't specifically address the probability or consequences of an accident previously evaluated. The discussion of the second standard is similar to SNC's discussion for the first standard and doesn't specifically address the possibility of a new or different kind of accident. The discussion of the third standard includes the topics of the first and third standard and provides little more than declarative statements. The NRC staff requests that additional information be provided, as discussed below:

1. With respect to whether operation of the facility in accordance with the proposed amendment would involve a significant increase in the probability or consequences of an accident previously evaluated, as addressed by 10 CFR 50.92(c)(1), provide responses to the following requests for information.

[1] The terms "standard" and "criterion" are used interchangeably in this NSHC discussion.

a. With respect to the analysis of design basis events in the Updated Final Safety Analysis Report (UFSAR) for the VEGP, discuss the basis for your conclusions regarding whether the proposed amendment would:

(i) adversely affect accident initiators or precursors,
(ii) adversely alter design assumptions, conditions, or configurations of the facility,
(iii) adversely impact the ability of SSCs to perform their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits,

b. Discuss the basis for your conclusions regarding whether the ability of SSCs to perform their design function as required by the accident analysis would be affected.

c. Discuss the basis for your conclusions regarding whether structures, systems, and components required to safely shut down the reactor and to maintain it in a safe shutdown condition will remain capable of performing their design functions.

d. Discuss the basis for your conclusions regarding whether any increases post-transition in core damage frequency or risk associated with the LAR submittal impact the conclusions reached with respect to the standard in §50.92(c)(1).

e. Discuss the basis for your conclusions regarding whether equipment required to mitigate an accident remains capable of performing the assumed function and accordingly, whether the consequences of any accident previously evaluated could be determined to not be significantly increased with the implementation of the amendment.

f. Discuss whether the proposed amendment will affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated and accordingly whether the applicable radiological dose criteria will continue to be met.

g. In the issuance of the final rule (69 FR 68014), it is stated that:

> The NRC recognizes that the reliability of RISC–3 SSCs could potentially decrease (RISC–3 SSC failure rates increase) due to the reduction in treatment applied to these SSCs as a result of § 50.69 implementation. This is the reason why the Commission requires in the rule that the licensee demonstrate with reasonable confidence that any potential risk increase due to implementation of the rule will be small.

Please discuss this subject with respect to ensuring that the 50.92(c)(1) standard will be met by implementation of § 50.69 at the VEGP.

2. With respect to whether operation of the facility in accordance with the proposed

amendment would create the possibility of a new or different kind of accident from any accident previously evaluated, as addressed by 10 CFR 50.92(c)(2), provide responses to the following requests for information.

a. Discuss the basis for your conclusions regarding whether the proposed change would alter the requirements or function for systems required during accident conditions, i.e., whether implementation of a new risk-informed categorization licensing basis which complies with the requirements in 10 CFR 50.69 will involve new failure mechanisms or malfunctions that can initiate a new accident.

b. Discuss the basis for your conclusions regarding whether the proposed amendment would adversely affect accident initiators or alter design assumptions, conditions, or configurations of the facility.

c. Discuss whether the proposed amendment will introduce any new accident scenarios, transient precursors, failure mechanisms, or limiting single failure modes that are not bounded by previously evaluated accidents.

d. Were all scenarios or previously analyzed accidents with potential offsite dose consequences included in the evaluation of the transition to 10 CFR 50.69?

3 With respect to whether operation of the facility in accordance with the proposed amendment would involve a significant reduction in a margin of safety, as addressed by 10 CFR 50.92(c)(3), provide responses to the following requests for information.

a. Discuss whether the proposed amendment would alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined.

b. Discuss whether the safety analysis acceptance criteria are affected by this change.

c. Discuss whether the proposed amendment would adversely affect existing plant safety margins or the reliability of equipment assumed to mitigate accidents in the Updated Final Safety Analysis Report.

d. Discuss the basis for your conclusions regarding whether the proposed changes are evaluated to ensure that risk and safety margins are kept within acceptable limits, with respect to the criterion of 10 CFR 50.92(c)(3).

e. Discuss any engineering analyses, engineering evaluations, probabilistic safety assessments or calculations that have been performed to demonstrate that the implementation of 50.69 will not result in a significant reduction in the margin of safety as addressed by 50.92(c).

August 1, 2014

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
Post Office Box 1295, Bin - 038
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Sincerely,

/RA/
Robert Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure: Request for Additional Information

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