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

April 3, 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 (TAC NOS. ME9472 and ME9473)

Dear Mr. Pierce:

By letter dated August 31, 2012, and supplemental letters dated May 17, July 2, and September 13, 2013, Southern Nuclear Operating Company, Inc. submitted a license amendment request to implement Title 10 of the Code of *Federal Regulations*, Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors" at the VEGP.

The U.S. Nuclear Regulatory Commission staff finds that additional information is needed as set forth in the Enclosure. Please provide the additional information within thirty (30) days of the date of this letter.

Sincerely,

Robert Martin

Robert E. 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

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SOUTHERN NUCLEAR OPERATING COMPANY, INC

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 (VEGP)

By letter dated August 31, 2012, and supplemental letters dated May 17, July 2, and September 13, 2013 (Agencywide Documents Access and Management System Accession Nos. ML12248A035, ML13137A480, ML13184A267 and ML13256A306, respectively), Southern Nuclear Operating Company, Inc. (SNC) submitted a license amendment request (LAR) to implement Title 10 of the *Code of Federal Regulations*, Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors." The latter three letters were in response to a request for additional information (RAI) dated April 17, 2013 (ML13102A233) which contained 27 RAIs. The numbering of the RAIs below is consistent with that of the April 17, 2013, RAI, letter.

3 In response to RAI 3, SNC clarified that the May 2009 peer review was conducted against the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard RA-Sb-2005 that was endorsed in Revision 1 of Regulatory Guide (RG) 1.200. Revision 2 of RG 1.200 endorses ASME/ANS RA-Sa-2009, a later version of the standard. Table 6 of the LAR identifies differences between Revisions 1 and 2 of RG 1.200, and relates those differences to the VEGP Probabilistic Risk Assessment (PRA). SNC intends to demonstrate compliance with Revision 2 of RG 1.200. However, some of the items listed in Table 6 of the LAR appear inconsistent with Section 3.3 of Nuclear Energy Institute (NEI) 05-04 "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard", Revision 3, November 2009. NEI 05-04 lists a number of supporting requirements that require or may require re-evaluation for updating PRAs to Revision 2 of RG 1.200. The majority of the supporting requirements listed in NEI 05-04 (for example, HR-G3, QU-A3, QU-B5, QU-E3, IFPP-B1) are not found in Table 6 of the LAR. Many of the items listed in Table 6 are not found in NEI 05-04. Please clarify these discrepancies and the basis for assuming compliance with Revision 2 of RG 1.200 for the Internal Events (including Internal Flooding) PRA.

7, 8 SNC's response to RAI # 8 appears to indicate that the pumps and check valves of the containment spray (CS) system were categorized as low-safety-significant (LSS). In RAI # 7 the staff asked SNC to provide more details on how it answered questions 4 and 5 in Section 9.2.2 of NEI 00-04, "10 CFR 50.69 SSC [Structure, System and Component], Categorization Guideline," Revision O, July 2005, regarding the determination as to whether a function/SSC provides the "sole means" of accomplishing a specific mitigation function. The response to RAI # 7 listed only one CS system function; the function related to containment pressure indication. Were other CS system functions considered? Please clarify the categorization process as it was applied to the CS system pumps and check valves and the outcome of the licensee's trial categorization for these components.

28 The categorization process described in NEI 00-04 starts with component categorization based on risk (i.e., the PRA or defense-in-depth), followed by function categorization, followed by the final assessment of the Integrated Decision-making Panel (IDP). When a component's function is categorized as high-safety-significant (HSS) based on risk, the associated system or train function is considered HSS. System or train functions not addressed by the risk assessment are categorized based on the seven questions in Section 9 of NEI 00-04. Once a function has been identified as HSS, then all components supporting the function are assigned as preliminary HSS and the IDP must intervene to assign any of these components to LSS. SNC's process is different from the categorization process described in NEI 00-04. Based on Clarification 1 in LAR Section 3.1.3, the licensee's categorization process performs function categorization based on the seven questions in Section 9 of NEI 00-04 (qualitative assessment), followed by component categorization based on risk, followed by the final assessment of the IDP. In SNC's process, it appears that a system or train function categorized as LSS based on the qualitative assessment would remain LSS even if the risk assessment identifies the associated component's function as HSS. If so, the risk assessment does not affect the preliminary LSS safety significance of other components associated with what appears should be an HSS function. Therefore, it appears that SNC's process could have more, perhaps many more, components categorized as preliminary LSS than the NEI method. If this characterization is incorrect please clarify. Please also clarify the difference in the IDP final assessment between components that are preliminarily HSS versus those that are preliminarily LSS. Explain whether this process deviates from the NEI approach if components could follow different paths in the licensee's approach than they would in the NEI approach.

29 Clarifications 2 and 3 in Section 3.1.3 of the LAR indicate that a passive component (i.e., a pressure retaining component) whose failure could fail an active HSS function can be assigned to LSS if the passive categorization process yields an LSS category. The passive categorization is driven by the consequence of failure and not the frequency so it is unclear how the pressure retaining function of a passive component whose failure would fail an HSS function can be found LSS in the passive categorization process. Please explain and provide an example.

30 The U. S. Nuclear Regulatory Commission (NRC) staff is currently investigating methods available to perform a Level 3 PRA as described in, "Technical Analysis Approach Plan for Level 3 PRA Project," Rev 0b, October 2013 (ADAMS Accession No. ML13296A064). SNC is participating in the Level 3 PRA Project. As described in the Approach Plan, the office of Nuclear Regulatory Research has notified the Office of Nuclear Reactor Regulation of three issues, as summarized below, regarding the VEGP PRA that have not been fully resolved and that might impact the results of SNC's pilot request to implement 50.69 for VEGP. Please summarize the impact of these three issues on your PRA models supporting your application to implement 10 CFR 50.69 and your proposed resolution of each issue with respect to this application.

i) Crediting offsite recovery within 4 hours without considering the impact of earlier consequential failures that could lead to unrecoverable scenarios.

ii) Crediting battery DC power for a longer time than the design.

iii) Use of a human error probability estimation method without apparently fully exercising the attributes used to apply the method to time sensitive activities.

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Sincerely,

**/RA/**
Robert E. 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

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Request for Additional Information

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ADAMS Accession No.: ML14093A357 *By memo dated March 26, 2014

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|---|---|---|---|---|---|
| NAME | RMartin | SFigueroa | HHamzehee* | RPascarelli | RMartin |
| DATE | 04/03/14 | 04/03/14 | 03/26/14 | 04/03/14 | 04/03/14 |

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