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

February 8, 2018

Mr. James J. Hutto
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – REQUEST FOR ADDITIONAL INFORMATION RE: REVISE TECHNICAL SPECIFICATION 5.5.17 FOR PERMANENT EXTENSION OF TYPE A AND TYPE C LEAK RATE TEST FREQUENCIES (CAC NOS. MG0240, MG0241; EPID L-2017-LLA-0295)

Dear Mr. Hutto:

By letter dated September 12, 2017, Southern Nuclear Operating Company (SNC) submitted a license amendment request to revise Vogtle Electric Generating Plant, Units 1 and 2 (VEGP), Technical Specifications (TS). The licensee proposed to revise TS 5.5.17 "Containment Leakage Rate Testing Program."

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is needed for the NRC to complete its review as discussed in the Enclosure. During a clarification call on February 8, 2018, Mr. Bates agreed that SNC respond within 45 days of the date of this letter. Please note that the NRC staff's review is continuing and further requests for information may be developed.

Sincerely,

Shawn Williams

Shawn Williams, 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/enclosure: Listserv

REQUEST FOR ADDITIONAL INFORMATION

TS 5.5.17 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-424 AND 50-425

By letter dated September 12, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17257A177),Southern Nuclear Operating Company (SNC) submitted a license amendment request to revise Vogtle Electric Generating Plant, Units 1 and 2 (VEGP), Technical Specifications (TS). The licensee proposed to revise TS 5.5.17 "Containment Leakage Rate Testing Program."

The U.S. Nuclear Regulatory Commission (NRC) staff is reviewing the submittal and has determined that the following additional information is needed to complete its review.

RAI No. 1

Section 2.5.3 of Regulatory Guide (RG) 1.174, Revision 2 (ADAMS Accession No. ML100910006), states:

> The development of the PRA [probabilistic risk assessment] model is supported by the use of models for specific events or phenomena. In many cases, the industry's state of knowledge is incomplete, and there may be different opinions on how the models should be formulated. Examples include approaches to modeling human performance, common-cause failures, and reactor coolant pump seal behavior upon loss of seal cooling. This gives rise to model uncertainty.

Regarding model uncertainty, Section 2.5.3 of RG 1.174, Revision 2, states:

> ...The impact of using alternative assumptions or models may be addressed by performing appropriate sensitivity studies or by using qualitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models.

In addition, Section 2.5.5 states:

> [I]n general, the results of the sensitivity studies should confirm that the guidelines are still met even under the alternative assumptions (i.e., change generally remains in the appropriate region)."

Enclosure

In Section 3.3.2 of the license amendment request (LAR), the licensee stated:

> The Westinghouse reactor coolant pump (RCP) shutdown seals have been installed at [Vogtle Electric Generating Plant] VEGP, and are credited in the PRA.

Please provide the following information to validate and confirm the PRA technical acceptability for use in the risk evaluation performed to support the requested permanent 15-year integrated leak rate test (ILRT) extension.

a. Provide a summary of the PRA modeling of the RCP shutdown seals, addressing the following aspects:

    1. Specify the PRA models (e.g., internal events, fire, seismic) that credit the RCP shutdown seals.

    2. Describe the PRA modeling of the RCP shutdown seals. Demonstrate how any limitations and conditions delineated in the NRC-approved guidance are being met [e.g., those in Section 3 of Topical Report PWROG-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shutdown Seal," and Section 5 of NRC safety evaluation (SE) for PWROG-14001-P (ADAMS Accession Number ML17200A116)].

    3. Indicate, and provide justification, whether the incorporation of the RCP shutdown seals into the PRA model is PRA maintenance or PRA upgrade, as defined in Section 1-5.4 of the American Society of Mechanical Engineers/ American Nuclear Society, ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2 (ADAMS Accession No. ML090410014). This discussion should be of sufficient detail to allow NRC staff to independently assess whether this change is a PRA maintenance or PRA upgrade (e.g., summarize the original method in the PRA and the new method, summarize the impact that this change has on significant accident sequences or the significant accident progression sequences).

b. If PRA modeling of the RCP shutdown seals is considered a PRA upgrade and a peer review(s) was performed for this upgrade, then discuss this peer review(s). In this discussion, describe the peer review process applied to the shutdown seal model; identify the guidance used to perform this peer review(s) (e.g., ASME/ANS RA-Sa-2009, Nuclear Energy Institute NEI 05-04, RG 1.200, Revision 2); include any necessary gap- or self-assessments if current guidance/standards were not used in the peer review(s); provide all facts and observations (F&Os) characterized as findings from the peer review(s) and the associated dispositions as it pertains to this application.

c. If PRA modeling of the RCP shutdown seals is considered a PRA upgrade and a peer review was not performed for this upgrade, then perform an appropriate sensitivity and/or bounding analysis for the RCP shutdown seals (e.g., remove credit for RCP shutdown seals) that assesses the contribution of risk for permanently extending the ILRT to 15 years. This analysis should also address the below consideration on the regulatory guides related to fire PRA (FPRA). Discuss this sensitivity/bounding analysis and provide updated risk values that include the increase in total large early release frequency (LERF), change in LERF (ΔLERF), population dose rate (PDR), and conditional containment failure probability (CCFP) for each unit to assess the risk impact. Confirm that the results of this analysis still meet the acceptance guidelines in RG 1.174, Revision 2, and Electric Power Research Institute EPRI Technical Report 1009325, Revision 2-A. If the acceptance guidelines are exceeded, then provide qualitative or quantitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models, to support the conclusion of the LAR. This discussion should include which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

If the FPRA applied in the sensitivity/bounding analysis does not incorporate the recently approved FPRA methodologies below, provide a detailed justification for why the integration of these NRC-approved FPRA methods and studies would not change the conclusions of the LAR.

- The NRC issued a letter dated June 21, 2012, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, 'Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires'" (ADAMS Accession No. ML12171A583), providing staff positions on (1) frequencies for cable fires initiated by welding and cutting, (2) clarifications for transient fires, (3) alignment factor for pump oil fires, (4) electrical cabinet fire treatment refinement details, and (5) the EPRI 1022993 report.

- The NRC published NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2 (ADAMS Accession No. ML14141A129), which is supported by a letter from the NRC to NEI, "Supplemental Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," and supplemented in April 2014 (ADAMS Accession Nos. ML14017A135 and ML14086A165).

- The NRC published NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009" (ADAMS Accession No. ML15016A069).

- Guidance on the credit taken for very early warning fire detection system (VEWFDS) is available in NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities, (DELORES-VEWFIRE)" (ADAMS Accession Nos. ML16343A058). The guidance provided in FAQ 08-0046, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems" (ADAMS Accession No. ML093220426), has been rescinded.

RAI No. 2

Table 6-2, "Vogtle Units 1 and 2 Internal and External Events Summary," provided LERF values for Units 1 and 2. The LERF values for the Units 1 and 2 internal events PRA (IEPRA) are both 6.45E-09/year. For fire events, the LERF values for Units 1 and 2 are 1.39E-06/year and 1.56E-06/year, respectively. The LERF values across the IEPRA and FPRA hazards are approximately two orders of magnitude (i.e., 10^{-2}) in difference, while that for core damage frequency (CDF) is one order of magnitude. Confirm the LERF values across the IEPRA and FPRA hazards.

RAI No. 3

In the section of the LAR entitled, "Assessment of PRA Model Technical Adequacy," the licensee stated that in 2013 "[a] significant upgrade in [Modular Accident Analysis Program] MAAP capabilities was initiated" for the IEPRA. Further review of the technical adequacy of the VEGP IEPRA does not readily identify whether a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA standard, as qualified by RG 1.200, Revision 2, was performed to assess the upgrade in MAAP capabilities.

Provide a summary of the upgrade in MAAP capabilities and justify whether the "significant upgrade" constitutes a PRA upgrade in accordance with ASME/ANS RA-Sa-2009, as qualified by RG 1.200, Revision 2. If the significant upgrade in MAAP capabilities constitutes a PRA upgrade, provide the results from the focused-scope peer review, including the associated F&Os and their dispositions for any impact to the ILRT application; otherwise, provide a quantitative evaluation (e.g., sensitivity or bounding analysis) of its effect until a focused-scope peer review can be completed. For this quantitative evaluation, confirm that the results of this analysis still meet the acceptance guidelines in RG 1.174, Revision 2, and EPRI Technical Report 1009325, Revision 2-A. If the acceptance guidelines are exceeded, then provide qualitative or quantitative arguments, based on an understanding of the contributors to the results and how they are impacted by the change in assumptions or models, to support the conclusion of the LAR. This discussion should include which metrics are exceeded and the conservatisms in the analysis and the risk significance of these conservatisms.

RAI No. 4

In Table 6-2 of the LAR, a CDF value of 2.52E-06/year and a LERF value of 6.45E-09/year for Units 1 and 2, respectively, are provided for the internal events hazard. These CDF and LERF values are identical for each unit. Typically, differences in CDF and LERF results exist for multiple-unit plants, even if the differences are not significant. Accordingly, it is not clear whether the risk values reported in the LAR are the results of separate PRAs performed for each unit or whether PRAs were performed only for a given unit and assumed to represent both units.

a. If the PRAs were performed only for a given unit and assumed to represent both units, then for the internal events hazard justify that the PRA model is an adequate representation of Units 1 and 2. Include a discussion of systems, structures, and components that are shared between units and how these were implicitly or explicitly modeled.

b. If the PRAs were performed for each unit separately, briefly explain why the risk results are identical.

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