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

May 1, 2018

Ms. Cheryl A. Gayheart
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 (CAC NOS. MF9685 AND MF9686; EPID L-2017-TOP-0038)

Dear Ms. Gayheart:

By letter dated April 21, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17116A096) as supplemented by letters dated July 11, 2017 (ADAMS Accession No. ML17192A245), November 9, 2017 (ADAMS Accession No. ML17314A014), January 2, 2018 (ADAMS Accession No. ML18004A070), January 9, 2018 (ADAMS Accession No. ML18009A841), February 6, 2018, (ADAMS Accession No. ML18037B121), February 12, 2018 (ADAMS Accession No. ML18045A094), and February 21, 2018 (ADAMS Accession No. ML18052B342), Southern Nuclear Operating Company, Inc. submitted a plant-specific technical report (TR) for Vogtle Electric Generating Plant, Units 1 and 2, and requested U.S. Nuclear Regulatory Commission (NRC) approval. The plant-specific TR describes a risk-informed methodology to evaluate debris effects with the exception of in-vessel fiber limits.

The NRC staff has reviewed the submittal and has determined that additional information is needed to complete its review. Enclosed is the NRC staff's request for additional information (RAI). The RAIs were discussed with your staff on April 30, 2018, and it was agreed that your response would be provided within 30 days from the date of this letter.

If you have any questions regarding this request, please contact me at (301) 415-2871 or Michael.Marshall@nrc.gov.

Sincerely,

Michael L. Marshall, Jr.

Michael L. Marshall, Jr., Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosure:
Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION

REGARDING SYSTEMATIC RISK-INFORMED ASSESSMENT OF

DEBRIS TECHNICAL REPORT

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 April 21, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17116A096) as supplemented by letters dated July 11, 2017 (ADAMS Accession No. ML17192A245), November 9, 2017 (ADAMS Accession No. ML17314A014), January 2, 2018 (ADAMS Accession No. ML18004A070), January 9, 2018 (ADAMS Accession No. ML18009A841), February 6, 2018, (ADAMS Accession No. ML18037B121), February 12, 2018 (ADAMS Accession No. ML18045A094), and February 21, 2018 (ADAMS Accession No. ML18052B342), Southern Nuclear Operating Company, Inc. (SNC, licensee) submitted a plant-specific technical report (TR) for Vogtle Electric Generating Plant, Units 1 and 2, and requested U.S. Nuclear Regulatory Commission (NRC) approval. The plant-specific TR describes a risk-informed methodology to evaluate debris effects with the exception of in-vessel fiber limits. In its letter dated April 21, 2017, the licensee has stated that the risk-informed methodology in the TR will be used to support future requests for licensing actions. In the letter dated April 21, 2017, SNC stated that the TR supersedes previous responses to Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," (ML042360586).

The NRC staff reviewed the submittal and has determined that the enclosed additional information is needed to complete its review. The requests for additional information (RAIs) listed below are not a complete listing of the additional information needed to complete the NRC staff's review. Additional RAIs were provided via separate correspondence. RAIs number 1 through 3 were sent in a separate letter dated October 12, 2017 (ADAMS Accession No. ML17264A282). RAIs number 4 through 10 were sent in a separate letter dated November 15, 2017 (ADAMS Accession No. ML17275A026). RAIs number 11 through 15 were sent in a separate letter dated November 22, 2017 (ADAMS Accession No. ML17318A035). RAIs number 16 through 36 were sent in a separate letter dated January 11, 2018 (ADAMS Accession No. ML17355A101).

Unless stated otherwise, all references to enclosures, sections, and page numbers in the RAIs are concerning the letter dated April 21, 2017.

RAIs

Paragraph (b) of Title 10 of the *Code of Federal Regulations*, Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires, in part, that an emergency core cooling system (ECCS) be provided for long-term cooling after successful initial operation of the ECCS. Water for long-term cooling is recirculated through the plant's sump strainer. GL 2004-02 contained a request that licensees provide verification that the sump screens (i.e., sump strainers) are capable of withstanding the loads imposed by the accumulation of debris and pressure differentials caused by blockage under flow conditions. Item 3.k of the NRC staff's revised content guide for GL 2004-02 supplement responses (ADAMS Accession No. ML073110389) requests licensees summarize the structural qualification results and design margins for various components of the sump strainer structural assembly.

(37) Table 3.k.1-3 of Enclosure 2, states that the crush pressure on the strainer due to suction strainer operation is equivalent to 10.1 ft. of head loss. This pressure is used in the load combinations for the structural analysis of the strainer. However, several locations in Enclosure 2 (e.g., Tables 3.f.14-1 and 3.g.16-1) identify strainer head loss values greater than 10.1 ft. and the supplemental response to GL 2004-02 item 3.f.7 notes that the strainer structural margin is 24 ft. It is not clear what the head loss limit is for the strainer.

a. Please identify the head loss limit for structural qualification of the strainer and explain how this value was determined.

b. If the value does not bound all postulated head loss values, please provide a justification for any exceedances.

Title 10 of the *Code of Federal Regulations*, Section 50.46, Subsection (a)(1) requires in part that cooling performance be calculated with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents (LOCA) of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Additionally, the subsection requires that the evaluation includes sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. In its risk-informed methodology, the licensee used guidance and acceptance guidelines described in Regulatory Guide (RG) 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML100910006).

(38) Section 2.3.3 of RG 1.174, Rev. 2, states that:

> ... technical adequacy will be understood as being determined by the adequacy of the actual modeling and the reasonableness of the assumptions and approximations.

Enclosure 1, Section 2.0, of the submittal states that a screening process can be used to eliminate scenarios that are not relevant, not affected by debris, or have an insignificant contribution. The results of this screening process are provided, but not a description (i.e., justification) of the systematic approach implemented to identify relevant initiating events and how scenarios are eliminated.

a. Please describe in detail the process leading to the identification of relevant internal initiating events (e.g., LOCA, open safety relief valve, water hammer-induced

LOCAs, non-piping LOCAs). Please include any criteria (quantitative or qualitative) used in the process for screening (i.e., eliminating) any initiating events or scenarios.

b. Please describe in more detail how the high-likelihood scenarios were determined and how the change in risk associated with low likelihood scenarios were determined for this application. Please include a summary of the process used to make these determinations.

c. Please describe in detail the systematic process applied to evaluate the impact of secondary side breaks. Please include a summary of how and why these breaks were screened in or out.

d. Please explain which secondary side breaks were screened from detailed analyses, and the basis for their screening.

(39) Section 2.3.1 of RG 1.174, Rev. 2, states that:

> ...the scope of a PRA is defined in terms of the causes of initiating events and the plant operating modes it addresses. Typical hazard groups considered in a nuclear power plant PRA include internal events, internal floods, seismic events, internal fires, high winds, external flooding, etc.

It is not apparent that the impacts of internal fire and external hazards, other than seismic, have been addressed in the submittal. These other external hazards may affect the total change in risk for this application of the PRA.

a. Please provide a justification (e.g., qualitative arguments or bounding analyses) that demonstrates that the risk contributions from internal fire would not affect this application of the PRA.

b. Please provide a justification (e.g., qualitative arguments or bounding analyses) that demonstrates that the risk contributions from external events other than seismic events would not affect this application of the PRA.

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ADAMS Accession No. ML18109A115 * via email **via memo

OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL2 1/LA	NRR/DE/ESEB*	NRR/DRA/APLB/TL**
NAME	MMarshall	KGoldstein	BWittick	MReisiFard
DATE	04/23/2018	04/19/2018	03/27/2018	04/09/2018
OFFICE	NRR/DORL/LPL2-1/BC	NRR/DORL/LPL1/PM		
NAME	MMarkley	MMarshall		
DATE	04/25/2018	05/01/2018		

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