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

December 4, 2017

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 – AUDIT PLAN RE: INCORPORATE SEISMIC PROBABILISTIC RISK ASSESSMENT INTO 10 CFR 50.69 CATEGORIZATION PROCESS (CAC NOS. MF9861 AND MF9862; EPID L-2017-LLA-0248)

Dear Mr. Hutto:

By letter dated June 22, 2017, Southern Nuclear Operating Company (SNC) submitted license amendment requests regarding the Vogtle Electric Generating Plant, Units 1 and 2 (VEGP). The proposed amendments would modify the licensing basis to implement a change to the approved voluntary implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants," by incorporating the use of the VEGP seismic probabilistic risk assessment into the previously approved 10 CFR 50.69 categorization process.

The U. S. Nuclear Regulatory Commission staff will conduct a regulatory audit to support its review of the proposed license amendments. The audit will be conducted at SNC corporate headquarters in Birmingham, Alabama on December 12-13, 2017. The audit plan is enclosed. The logistics and scope of the audit was discussed with your staff on December 4, 2017.

If you have any questions, please contact me by telephone at 301-415-3229 or by e-mail at Michael.Orenak@nrc.gov.

Sincerely,

Michael D. Orenak, 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:
1. Audit Plan
2. Audit Information Needs

AUDIT PLAN

INCORPORATION OF SEISMIC PROBABILISTIC RISK ASSESSMENT INTO 10 CFR 50.69

CATEGORIZATION PROCESS SOUTHERN NUCLEAR OPERATING COMPANY

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

I. BACKGROUND

By letter dated June 22, 2017 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML17173A875), the Southern Nuclear Operating Company (SNC, the licensee) submitted a license amendment request (LAR) regarding the Vogtle Electric Generating Plant, Units 1 and 2 (VEGP). The proposed amendment would modify the licensing basis to implement a change to the approved voluntary implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants." The proposed amendment would incorporate the use of the VEGP seismic probabilistic risk assessment (SPRA) into the previously approved 10 CFR 50.69 categorization process.

II. REGULATORY AUDIT BASES

The purpose of the audit is to obtain a more detailed understanding of the technical adequacy of the SPRA model credited in the LAR. The U.S. Nuclear Regulatory Commission (NRC) staff has determined that an audit is the most efficient approach toward a timely resolution of issues associated with this review. The use of an audit will minimize the potential for multiple rounds of requests for additional information (RAI) and ensure that unnecessary burden will not be imposed by requiring the licensee to address issues that are unnecessary to make a safety determination.

At the end of the audit, the audit team expects to have a complete understanding of the SPRA to support the risk-informed categorization process. Additionally, upon completion of this audit, the NRC staff expects to issue RAIs to ensure that the licensee-provided information is sufficient to complete the LAR review.

III. REGULATORY AUDIT SCOPE AND METHODOLOGY

The scope of the audit includes key components of the SPRA, specifically:

- Determination of the acceptability of the SPRA for the 10 CFR 50.69 program.
- Review of the unique aspects of using the SPRA in the risk-informed categorization process.
- Review of the key assumptions and sources of uncertainty.

To accomplish these objectives, the NRC staff:

- Requests the licensee to provide an overview presentation of the SPRA
- Review the requested documentation listed in Enclosure 2, "Documentation to be Available for NRC Staff Review," to gain a better understanding of technical approaches used in the SPRA, such as fragility analysis, human reliability analysis, and consideration of key assumptions and sources of uncertainty.
- Discuss any questions with the licensee.

The audit will be performed consistent with NRC Office of Nuclear Reactor Regulation Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195).

IV. INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The information needed for the regulatory audit is listed in Enclosure 2. The audit team will not remove non-docketed information from the audit site.

V. AUDIT TEAM ASSIGNMENTS

The members of the audit team will be:

- Mehdi Reisi-Fard, Technical Reviewer, PRA, NRC
- Shilp Vasavada, Technical Reviewer, PRA, NRC
- Kaihwa Hsu, Technical Reviewer, fragility analysis, NRC

VI. LOGISTICS

The NRC staff will conduct the audit on December 12-13, 2017, in the SNC corporate offices in Birmingham, Alabama, or other locations agreed upon by the licensee and NRC staff that facilitates access to the licensee's computer models, documentation, and technical experts performing the work on the SPRA model. The NRC Project Manager will coordinate any changes to the audit schedule and location with the licensee.

VII. SPECIAL REQUESTS

The NRC staff would like access to the following equipment and services:

- Telephone with a speaker or speaker phone.
- Enclosed conference room (or comparable space) with a table, chairs, and white board.
- A projector and screen.
- Wireless internet access (if available in the work space).
- Computer access to any e-portal applicable for this audit.

VIII. DELIVERABLES

An audit summary will be prepared within 90 days of the completion of the audit. If information evaluated during the audit is needed to support a regulatory decision, the NRC staff will identify it in an RAI. The NRC staff will provide the request for additional information to the licensee in separate docketed correspondence.

IX. REFERENCES

(1) Southern Nuclear Operating Company, "License Amendment Request to Modify Approved 10 CFR 50.69 Categorization Process," June 22, 2017 (Agencywide Document Access and Management System (ADAMS) Accession No. ML17173A875).

(2) U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-111, "Regulatory Audits," December 29, 2008 (ADAMS Accession No. ML082900195).

(3) Nuclear Energy Institute, NEI 12-13, "External Hazards PRA Peer Review Process Guidelines", August 2012 (ADAMS) Accession No. ML122400044).

(4) U.S. Nuclear Regulatory Commission, Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (ADAMS Accession No. ML090410014).

(5) Nuclear Energy Institute, NEI 12-13, "External Hazards PRA Peer Review Process Guidelines", November 16, 2012 (ADAMS) Accession No. ML12321A280).

(6) Southern Nuclear Operating Company (SNC), "License Amendment Request for Approval to Utilize the Tornado Missile Risk Evaluator (TMRE) to Analyze Tornado Missile Protection Non-Conformances," October 11, 2017 (ADAMS Accession No. ML17284A348).

(7) Nuclear Energy Institute, NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (ADAMS Accession No. ML052910035).

(8) U.S. Nuclear Regulatory Commission, Pressurized Water Reactor Owners Group (PWROG)-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shut-Down Seal," (ADAMS Accession No. ML17200C876).

(9) Southern Nuclear Operating Company (SNC), "Seismic Probabilistic Risk Assessment," March 27, 2017 (ADAMS Accession No. ML17088A130).

(10) Sanida National Laboratories, NUREG/CR-4840, "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," November 1990, (ADAMS Accession No. ML063460465).

AUDIT INFORMATION NEEDS

Documentation to be Available for NRC Staff Review

- SPRA Peer Review Report
- SPRA Self-Assessment Report
- SPRA Walkdown Report
- SPRA Human Reliability Analysis Notebook
- SPRA Seismic Equipment List Notebook
- SPRA Seismic Plant Response Notebook
- SPRA Quantification Notebook
- Any other documents that include discussions on key assumptions and sources of uncertainty for the seismic probabilistic risk assessment (SPRA) with licensee disposition relevant to this application

Other Information Needs

1. Section 3.2 of the Enclosure to the June 22, 2017, submittal cites the licensee's SPRA peer-review report, stating that the peer review was performed using the process defined in Nuclear Energy Institute (NEI) 12-13, "External Hazards PRA Peer Review Process Guidelines" (Agencywide Document Access and Management System (ADAMS) Accession No. ML122400044). The Enclosure also states that no exceptions to the use of NEI 12-13 were noted in the peer-review report.

   While NEI 12-13 follows a process similar to U.S. Nuclear Regulatory Commission NRC endorsed peer review processes, NEI 12-13 has not been endorsed by the NRC in Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014). Please provide the following additional information to justify the use of NEI 12-13:

   a) Please describe how the qualifications of the VEGP SPRA peer review team comply with the peer review requirements in Sections 1-6.2 and 5-3.2 of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) Standard (RA-Sa-2009), as endorsed in RG 1.200.

   b) Please identify the unreviewed analysis methods (UAMs) used in the VEGP SPRA, as determined by the peer review team, and describe each UAM with a level of detail appropriate for the NRC staff's evaluation of its acceptability.

c) Please describe if the VEGP SPRA relies on expert judgement to meet any supporting requirement (SR) and, if so, demonstrate conformance to the expert judgment requirements of Section 1-4.3 of the 2009 ASME/ANS PRA Standard (RA-Sa-2009). Also, please cite any information from the peer review report related to the evaluation of the use of expert judgment by the peer review team and whether the peer review team found the use of expert judgment to be appropriate.

d) Please clarify whether VEGP SPRA was reviewed against Capability Category (CC) I for any SR. Provide a list of all SRs that were reviewed against CC I or found to meet only CC I without an associated finding. For each such SR, please justify why not meeting the SR at CC II does not impact this application.

e) Please clarify whether an "in-process" peer review was performed for the VEGP SPRA. If an "in-process" approach was utilized, confirm that (i) the approach met the requirements for an independent peer review as stated in the PRA Standard endorsed in RG 1.200, Revision 2, and the process described in NEI 12-13, (ii) a final review by the entire peer review team occurred after the completion of the SPRA, and (iii) peer reviewers remained independent throughout the PRA development activity since the peer reviewers in the early interim peer reviews must still participate in the final peer review, per the NRC staff's comments on NEI 12-13 in the letter dated November 16, 2012 (ADAMS Accession No. ML12321A280).

2. Section 5-2.3 of Part 5, "Requirements for Seismic Events At-Power PRA," of the ASME/ ANS PRA Standard (RA-Sa-2009) assumes that a full-scope internal-events at-power Level 1, and Level 2 large early release frequency (LERF), PRA exists and that those PRAs are used as the basis for the SPRA systems analysis. High Level Requirement (HLR)-SPR-B of ASME/ANS Standard (RA-Sa-2009) calls for the incorporation of seismic analysis aspects that are different from the at-power internal events PRA systems model. Therefore, the technical adequacy of the internal events PRA model used as the foundation for the SPRA needs to be established. Please identify internal events PRA finding-level facts and observations (F&Os) that were not closed per an NRC-approved process and any internal events PRA upgrades that had not been peer-reviewed prior to the development of the SPRA. For each finding-level F&O identified, describe the resolution or the impact of the F&O on this application.

3. Section 4.3 of Enclosure 3 to the October 11, 2017, submittal "License Amendment Request for Approval to Utilize the Tornado Missile Risk Evaluator (TMRE) to Analyze Tornado Missile Protection Non-Conformances," (ADAMS Accession No. ML17284A348) discusses the 2015 internal events update and states that the major change during the update was the addition of Westinghouse Owners Group (WOG) shutdown seal modeling. The discussion proceeds to state that a peer review was not required for these revisions. Per Section 4.1 of the October 11, 2017, submittal, the most recent peer-review of the licensee's internal events PRA model was performed in 2009. Therefore, it appears that the addition of the WOG shutdown seal model to the licensee's internal events PRA, which forms the basis of the VEGP SPRA, was never peer-reviewed.

   Further, Section 3.1 of the Enclosure to the June 22, 2017, submittal states that the only additional sensitivity analysis required beyond those specified in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," (ADAMS Accession No. ML052910035) is:

> [...]to evaluate the impact of the possibility that, following actuation of the reactor coolant pump shutdown seals (RCP SDS), there may be some scenarios where cold leg temperatures could exceed the rated temperature in a timeframe insufficient to credit operator action following a seismic event, leading to RCP seal LOCA [Loss-of-Coolant Accident] not currently included in the SPRA.

The Enclosure to the June 22, 2017, submittal further states that the issue is under investigation with the vendor of the RCP SDS.

The NRC staff's safety evaluation for Pressurized Water Reactor Owners Group (PWROG)-14001-P, Revision 1, "PRA Model for the Generation III Westinghouse Shut-Down Seal," (ADAMS Accession No. ML17200C876) imposed limitations and conditions on the use of the models and parameters presented in the report. The limitations and conditions included a requirement for an analysis to be performed to demonstrate that the SDS remains below its maximum qualified temperature limit in the event of the cold leg temperature exceeding a certain threshold value and a requirement to develop plant-specific Human Error Probabilities (HEPs) for certain specific actions relevant to the issue cited in the submittal dated June 22, 2017.

a. Please clarify whether the addition of the WOG shutdown seal model to the internal events PRA model has been peer-reviewed. If the addition of the WOG shutdown seal model has not been peer-reviewed, justify why the addition of the WOG shutdown seal model in the internal events PRA, which forms the base for the VEGP SPRA, is not considered a PRA upgrade requiring a focused-scope peer review. If this change qualifies as an upgrade, provide the results from one of the focused-scope peer reviews addressing the associated F&Os and their resolutions.

b. Please demonstrate how the limitations and conditions in the NRC safety evaluation for PWROG-14001-P, Revision 1, are being met for the scenarios identified by the licensee in Section 3.1 of the Enclosure to the June 22, 2017, submittal. Alternately, describe a sensitivity study that evaluates the impact of RCP seal LOCAs not currently modeled in the VEGP SPRA on the current application.

4. Section 3.3 of the Enclosure to the June 22, 2017, submittal states that the licensee will follow industry guidance and common practice in determining whether an update of the SPRA may be warranted due to new availability of new consensus seismic hazard information. Section 3.1.2 of the Enclosure to the June 22, 2017, submittal describes the licensee's PRA maintenance and updates process and states that the process includes provisions for monitoring potential areas affecting the PRA models and for assessing the risk impact of unincorporated changes. Further, the licensee states that the assessment of the impact of the changes will be performed no longer than once every two refueling outages. Section 12.1 of NEI 00-04 states that the assessment of new technical information should be performed during the normally scheduled periodic review cycle. The VEGP SPRA, including the hazard information used therein, is unique to the site and the as-built, as-operated plant. Please justify

the reliance on industry guidance and common practice, which will include non-site specific considerations and timeframes, instead of the licensee's periodic PRA maintenance process, which is site-specific, to identify and determine the incorporation of new information into the seismic hazard results for VEGP.

5. Section 3.3.2 of RG 1.200, Revision 2, states "[f]or each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application." Further, Section 4.2 of RG 1.200 states that "[t]hese assessments provide information to the NRC staff in their determination of whether the use of these assumptions and approximations is appropriate for the application, or whether sensitivity studies performed to support the decision are appropriate." RG 1.200, Revision 2, defines the terms "key assumption" and "key source of uncertainty" in Section 3.3.2, "Assessment of Assumptions and Approximations."

   Section 3.1.3 of the Enclosure to the June 22, 2017, submittal states that key VEGP SPRA model specific assumptions and sources of uncertainty for this application have been identified and dispositioned. Section 5.7 of VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, (ADAMS Accession No. ML17088A130) describes several sensitivity studies that were performed to examine different input information and assumptions on the VEGP SPRA results. Section 3.1.3 of the Enclosure to the June 22, 2017, submittal states that no additional SPRA-specific sensitivities have been identified that would be expected to have an important impact on categorization results.

   a. Please describe the approach used to identify and characterize the "key" assumptions and "key" sources of uncertainty in the licensee's SPRA. The description should contain sufficient detail to identify: (1) whether all assumptions and sources of uncertainty related to all aspects of the hazard, fragility, and plant response analysis were evaluated to determine whether they were "key," and (2) the criteria that were used to determine whether the modeling assumptions and sources of uncertainty were considered "key." Also, please explain how the approach adequately resolves the concerns raised in F&Os 12-23, 12-24, 12-27, and 12-29 from Attachment 2 of the submittal dated June 22, 2017.

   b. Please describe each key assumption and key source of uncertainty identified in the VEGP SPRA that was not provided in the VEGP response to the March 2012,10 CFR 50.54(f) letter, dated March 27, 2017. The description should contain sufficient detail to identify whether key assumptions used in the SPRA involve any changes to consensus approaches.

   c. Please discuss how each key assumption and key source of uncertainty either identified above or presented in VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, was dispositioned for this application. If available, provide sensitivity studies that will be used to support this application or use a qualitative discussion to justify why different reasonable alternative assumptions would not affect this application.

6. Section 5.3 of NEI 00-04 indicates that components can be identified as being safety-significant following sensitivity studies. Section 5.3 also recommends the completion of several sensitivity studies, including any applicable sensitivity studies identified in the characterization of PRA adequacy.

    a. Table 5-4 of NEI 00-04 shows that one of the sensitivities to be performed for SPRA use in the categorization process is to use correlated fragilities for all structures, systems, and components (SSCs) in an area. Section 4.4.2 of the licensee's response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, states that:

        > [i]f the equipment is similar in design, with similar anchorage, and located in the same building on the same elevation, then it is treated as a correlated failure

        but continues with there were a:

        > [...]few exceptions to this general correlation rule.

        Please describe, with justification, how the sensitivity study for correlated fragilities will address the exceptions to the general correlation rule.

    b. Table 5-4 of NEI 00-04 identifies, among other SPRA sensitivity studies, any applicable sensitivity studies identified in the characterization of PRA adequacy. Please clarify whether the sensitivity analyses identified in all other requests for information above (i.e., sensitivity analyses discussed in questions 3, 5.c, 5.d, 6.c.iv, 6.d, etc.) will be performed every time SSCs are categorized under 10 CFR 50.69.

7. The following requests for information apply to the SPRA F&Os and their corresponding resolutions as reported in Attachment 2, "Disposition and Resolution of SPRA Peer Review Findings" of the Enclosure to the June 22, 2017, submittal:

    a. F&O 14-10, related to Supporting Requirement (SR) SFR-A2, assigns a CC I to that SR and states that significant conservatisms were noted in several sampled fragility calculations. The "Finding Basis" cites the Component Cooling Water (CCW), Auxiliary Component Cooling Water (ACCW), a particular battery rack, and the turbine driven auxiliary feedwater pump as examples. The resolution to the F&O mentions updates only to the examples cited in the F&O. Using conservative fragilities can lead to incorrect categorization and therefore, treatment of SSCs based on the SPRA results. Please discuss how SNC ensured that instances of significant conservatism in the fragilities of SSCs, other than the examples cited by the peer-review team, were identified and adequately addressed in the VEGP SPRA in the context of the June 22, 2017, application.

    b. The "Finding Basis" for F&O 14-20, related to SRs SPR-B9 and SFR-E4, states that it is understood that seismic-induced fire was a key consideration during the walkdowns, but details of the walkdown procedure for fire following an earthquake is missing. The NRC staff understands that the peer-reviewers did not have the opportunity to review the seismic-fire interaction methodology employed during the walkdowns and the results therefrom. Section 4.2 of the of the licensee's response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, provides a few examples of the type of seismic-fire interactions evaluated during the walkdown, but does not provide details of the approach used during the walkdown.

Please describe the approaches used for (1) identifying the seismic-fire interaction sources, (2) performing a systematic screening of the identified sources, (3) considering human actions during the screening, and (4) inclusion of the unscreened interactions in the seismic PRA. In responding to the items above, please cite any applicable consensus or state-of-practice approach used, including deviations from such an approach, and describe the impact of the deviations on this application.

c. F&O 16-18, related to SR SPR-B8, briefly describes the licensee's unique approach to screening out Very Small Loss-of-Coolant Accidents (VSLOCAs) based on walkdowns. The F&O states that little documentation exists of such walkdowns, and the resolution treats the F&O as a documentation issue only. However, the F&O statements appear to indicate that the peer-review team, due to the limitations cited in the F&O, did not review or only partially reviewed the associated documentation to determine the adequacy of the VSLOCA treatment.

   i. Justify the disposition of the F&O as a documentation issue, including a description of the information that was available to the peer-review team in the context of the F&O statements.

   ii. Describe the methodology followed for the systematic evaluation of the possible sources of VSLOCAs based on the walkdown.

   iii. Provide justification for the evaluation that all relevant lines were "judged to be rugged" capacity such that the initiator can be completely screened out, recognizing that the 2009 ASME/ANS PRA Standard (RA-Sa-2009) states that:

   > [...]breaks in one or a very few such lines cannot always be precluded, given the large number of such lines and their unusual configurations in many cases.

   iv. The documentation of the licensee's response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, specifically Section 5.7.2, indicates that the impact of the VSLOCA was captured via a sensitivity performed using the small LOCA (SLOCA) event tree. Provide the justification for using the SLOCA logic, including accident progression, sequence timing, success criteria, and the availability of makeup in the logic model, as a surrogate for VSLOCAs in the cited sensitivity study and describe the impact on this application. Confirm that this sensitivity analysis will be performed as part of the categorization process. Alternatively, justify why a sensitivity analysis related to the VSLOCA approach is not warranted as part of the categorization process to address the technical adequacy of PRA for this application.

d. F&O 16-5, related to SRs SPR-B1 and SPR-F1, cites concerns with the LOCA modeling and fragility selection. The resolution states that the LOCA basis has been reevaluated and updated. Based on the discussion in the "Finding Basis" and the resolution, the NRC staff assumes that the surrogate component(s) used to represent the fragility for LOCAs was changed, subsequent to the peer-review, to that for the RCP supports. Representative fragilities for SLOCAs and medium LOCAs (MLOCAs) in available industry guidance documents, as well as in NUREG/CR-4840 "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," (ADAMS Accession No. ML063460465) can be appreciably lower than those for the RCP supports. Using unrealistic fragilities can lead to incorrect categorization and therefore, treatment, of SSCs based on the SPRA results.

   i. Please provide the technical justification, including a discussion of the impact on seismic cored damage frequency (SCDF) and SSC categorization, for the selection of the currently used surrogate components and the corresponding fragilities for modeling SLOCAs and MLOCAs.

   ii. Please confirm that a sensitivity analysis will be performed as part of the categorization process to address the uncertainty associated with the surrogate fragility. Alternatively, justify why a sensitivity analysis is not warranted as part of the categorization process to address the technical adequacy of PRA for this application.

e. F&Os 16-4, 16-6, and 16-9, related to SRs SPR-B2, SPR-B1, and SPR-B4b, respectively, question the Human Reliability Analysis (HRA) method employed in the model available for peer-review. The resolutions of these F&Os state that the Electric Power Research Institute (EPRI) guidance for HRA implementation for SPRAs (i.e., EPRI Report Number 3002008093, "An Approach to Human Reliability Analysis for External Events with a Focus on Seismic") was utilized and the "[...]bins (breaking points) have been updated with additional breaking points...to reflect seismic binning applicable to Vogtle."

   i. Please justify the selection of the breaking points, including the 'critical breaking point' (beyond which all human error probabilities are set to unity), with respect to the cited EPRI guidance and the fragilities of key SSCs in the SPRA.

   ii. Please confirm that sensitivity analyses described in VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, will be performed as part of the categorization process to address the HRA assumptions and uncertainties related to issues identified by F&Os 16-4, 16-6, and 16-9. If other sensitivity analyses will be performed to support this application, describe those sensitivity analyses. Alternatively, justify why a sensitivity analysis is not warranted as part of the categorization process to address the technical adequacy of PRA for this application.

f. F&O 16-11, related to SR SPR-E2, states that the review of the potential for additional dependencies introduced by the SPRA model is missing. The resolution states that the dependency analysis has been performed using the

EPRI HRA Calculator. The "Suggested Finding Resolution" states that the licensee plans to transition to a different dependency analysis method (based on HRA calculator). The 2009 ASME/ANS PRA Standard (RA-Sa-2009) defines a PRA upgrade as,

> [...]incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences.

Non-mandatory Appendix 1-A of the 2009 PRA Standard cites "a different HRA approach to human error analysis..." as a potential PRA upgrade. Based on Section 1-5, "PRA Configuration Control," of the standard and RG 1.200, Revision 2, Regulatory Position 1.4, "PRA Development, Maintenance, and Upgrade," a PRA upgrade must be peer-reviewed.

Please provide a justification as to why the use of the EPRI HRA Calculator in the VEGP SPRA is not considered a PRA upgrade requiring a focused-scope peer review. Include a comparison of the implemented methods and base values for the top 25 HEPs, as well as HRA dependencies (if not part of the top 25 HEPs), pre- and post-Calculator use. Please use this comparison to demonstrate that the same methods, steps, and sequence that had been used in the pre-existing, manual HRA calculations were exactly mirrored when adopting the HRA Calculator. If this change qualifies as an upgrade, provide the results from the focused-scope peer review addressing the associated F&Os and their resolutions.

g. The "Finding Basis" for F&O 14-1, related to SR SFR-A2, states that structural response factor used in all component fragilities reviewed by the peer-review team is reported as 1.0. The "Finding Basis" further states this factor will be greater than 1.0 because of the conservatism introduced in the demand through structural analysis. Because of this, the component and structural fragilities are biased low.

   The NRC staff recognizes that in the design-analysis, the structural response was computed using specific (often conservative) deterministic response parameters for the structure. Because many properties, such as damping, soil property and the response combination method, etc. are random (often with the wide variability), for a given peak ground acceleration level, the actual structural response may differ substantially from the design analyses calculated response. Please provide the basis for deriving 1.0 as structural response factor and provide logarithmic standard deviation of the randomness (β_R) and state of knowledge uncertainty (β_U). Cite any applicable consensus approach used, including deviations from such approach, and describe the impact of the deviations on this application.

h. The "Finding Basis" for F&O 14-17, related to SR SFR-D2, states that the reactor internals fragility evaluation determined the demand based on an average spectral acceleration over the range of 2 to 3 Hertz (Hz), rather than using the peak acceleration in this range of the in-structure response spectra (ISRS), and did not consider the contribution of higher modes. The licensee indicated that this was done to avoid an overly conservative capacity, but agreed that the contribution of higher modes should be addressed. The licensee stated in their disposition that the reactor internals fragility has been updated in the calculation. The NRC staff notes that using the average spectral acceleration over the range of 2 to 3 Hz may result in fragilities which are lower than the actual fragility.

Please justify that using the average spectral acceleration will not generate non-conservative fragility analysis and describe, in specific detail, how the contribution of higher modes was revised.

i. F&O 14-7, related to SRs SFR-A2 and SFR-F4, states that the fragility evaluation for the containment polar crane did not address the impact of variation in the fundamental frequency on the applicable seismic demand. The licensee's disposition states that the fragility evaluation has been updated to address potential uncertainty in the fundamental frequency and contribution of higher modes.

   Please describe how the fragility evaluation was updated to address the potential uncertainty in the fundamental frequency and contribution of higher modes. Cite any applicable consensus approach used, including deviations from such approach, and describe the impact of the deviations on this application. Clarify whether a sensitivity analysis will be performed as part of the categorization process to address the uncertainty associated with this finding.

j. The "Finding Basis" for F&O 14-8, related to SR SFR-F3, states that the median capacity for two relays identified in the F&O is not realistic. The licensee's disposition states that the relay fragilities have been updated using the appropriate response and in-cabinet amplification factors.

   Please describe how the relay evaluations were revised. Cite any applicable consensus approach used, including deviations from such approach, and describe the impact of the deviations on this application. Clarify whether a sensitivity analysis will be performed as part of the categorization process to address the uncertainty associated with this finding.

k. The "Finding Basis" for F&O 14-9, related to SR SFR-D2, states that it was determined that, among other issues, valve operator heights/weights that are outside of EPRI guidelines would require further effort for resolution. The licensee's disposition states that valve operator heights and weights that were outside EPRI guidelines have been taken into account in the fragility analysis for these components.

   Please describe how valve operator heights/weights that are outside EPRI guidelines were taken into account in the fragility analysis. Cite any applicable consensus approach used, including deviations from such approach, and describe the impact of the deviations on this application.

l. The "Finding Basis" for F&O 14-10, related to SR SFR-A2, states that the generic equipment response spectra (GERS) capacity for battery rack fragility calculations is not realistic and the fragility calculation for the turbine driven auxiliary feedwater pump did not consider the entire frequency range. This F&O

states that significant conservatisms were noted in several sampled fragility calculations. The licensee's disposition states that the fragilities were updated to account for appropriate frequency, and uncertainty has been considered in these updates.

Please describe how the fragility calculations were revised to address the identified issues. Cite any applicable consensus approach used including deviations from such approach, and describe the impact of the deviations on this application. Discuss how the licensee ensured that other potential significant conservatisms in fragility calculations in VEGP SPRA have been addressed.

8. The regulation 10 CFR 50.69(c)(1)(iv) requires that the categorization process includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, any potential increase in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment are small. The regulations 10 CFR 50.69(e)(2) and (3) require the licensee to monitor the performance of RISC-1 and RISC-2 SSCs and consider the data collected for RISC-3 SSCs and make adjustments to the categorization or treatment processes so that the categorization process and results are maintained valid.

   Section 8 of NEI 00-04 provides guidance on how to conduct risk sensitivity studies during the categorization process for all the preliminary low-safety-significant (LSS) SSCs to confirm that the categorization process results in acceptably small increases to CDF and LERF. An example is provided in the guidance to increase the unreliability of all preliminary LSS SSCs by a factor of 3 to 5, which appears to address random failures. No explicit discussion of seismic risk sensitivity studies is provided in the guidance.

   The categorization of SSCs using the SPRA is dominated by structural failure modes which are dependent on the corresponding modeling inputs such as the 'dominant failure modes' and 'fragility curves'. These modeling inputs are derived using several parameters, including the SSC design, testing, and as-built installation, all of which can be impacted by alternative treatments.

   Additionally, Section 5.3 of NEI 00-04 states that for SSCs screened out of the SPRA due to 'inherent seismic robustness', it is important that the inherent seismic robustness that allows them to be screened out of the seismic PRA is retained.

   a. Please describe and justify how the required risk sensitivity study outlined in Section 8 of NEI 00-04 will be performed for categorization using the SPRA to meet the requirements of 10 CFR 50.69(c)(1)(iv) and 10 CFR 50.69(b)(2)(iv).

   b. Please describe how it will be determined that the modeling inputs in the SPRA and those used for the risk sensitivity study continue to remain valid to ensure compliance with the requirements of 10 CFR 50.69(e).

9. The categorization of SSCs, including that using the SPRA, is expected to be based on importance measures and corresponding numerical criteria as described in Sections 5.1 and 5.3 of NEI 00-04. Further, Section 5.6 of NEI 00-04 discusses the "integral assessment" wherein the hazard specific importance measures are weighted by the

hazards contribution to the plant risk. Based on the information provided by the licensee in Tables 3.1-2, 5.4-3, and 5.5-3 of the licensee's response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, it appears that the modeling and subsequent quantification of the licensee's SPRA is based on 'binning' the seismic hazard and its consequences.

a. Please describe how the importance measures are determined from the SPRA in the context of the 'binning' approach employed in the VEGP SPRA. Clarify how the same basic events, which were discretized by binning during the development of the SPRA, are then combined to develop representative importance measures. Further, discuss how they are compared to the numerical criteria, justify any impact on the categorization results, and describe how the approach is consistent with the guidance in NEI 00-04.

b. In the context of the "integral assessment" described in Section 5.6 of NEI 00-04, describe how the SPRA importance measures will be used to calculate the integrated importance measures, justify any impact of the approach for calculating the SPRA importance measures on the "integral assessment" based categorization results, and show consistency of the approach with the guidance in NEI 00-04.

10. Section 3.2 of the Enclosure to the June 22, 2017, submittal states that the VEGP SPRA peer review was performed using the SPRA requirements in Addendum B of the ASME/ANS PRA Standard (ASME/ANS RA-Sb-2013). RG 1.200, Revision 2, endorses Addendum A (ASME/ANS RA-Sa-2009) but does not endorse PRA Standard Addendum B. The licensee's "basis for assessment" of the requirements in each SR of Part 5 of Addendum B of the PRA Standard (ASME/ANS RA-Sb-2013) compared to those in Addendum A in the context of the licensee's SPRA are provided in the Enclosure to the July 11, 2017, letter.

    The "basis for assessment" for the difference between Addenda A and B for SFR-C6 states that the licensee's SPRA conforms to accepted current practices and that Vogtle accounted for uncertainties in the soil-structure interaction (SSI) analysis by applying strain-compatible soil properties derived from probabilistic evaluation via the probabilistic seismic hazard analysis (PSHA). However, Sections 4.3.2 and 4.3.3 of licensee's submittal in response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, does not include an explicit discussion of the connection between the PSHA and the SSI analysis.

    Please provide details of the approach followed to determine the median response and uncertainty for the SSI analysis for the VEGP SPRA, and confirm the compatibility between the PSHA and SSI analysis. Cite any guidance and/or consensus approach that was followed in deriving the input parameters for the SSI analysis and justify any deviations.

11. Section 5.1 of NEI 00-04 provides guidance on the use of importance measures for identifying the "candidate safety significance" of components during the categorization process.

    a. Section 5.1 of NEI 00-04 states that in calculating the F-V risk importance measure, it is recommended that a CDF (or LERF) truncation level of five orders of magnitude below the baseline CDF (or LERF) value be used for linked fault tree PRAs and that the truncation level used should be sufficient to identify all functions with a risk achievement worth of greater than 2. According to the information in Section 5.7.1 of the licensee's submittal in response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, the selected truncation limit for the "higher bins" of the VEGP SPRA does not meet the guidance in NEI 00-04 and the rationale provided is related to quantification efficiency. Please demonstrate the impact of the selected truncation level for the "higher bins" in the VEGP SPRA on the importance measure criteria and the categorization.

b. According to the information in Section 4.4.1 of the licensee's submittal in response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, the value used for screening SSCs from the VEGP SPRA was adjusted until the maximum contribution was 2 percent of the final SCDF. Please describe how the selected screening level in the VEGP SPRA maintains consistency with the importance measure criteria in NEI 00-04 and demonstrate the impact of the selected screening level in the VEGP SPRA on the importance measure criteria and the categorization of SSCs.

12. Section 3.1.3 of the Enclosure to the submittal, dated June 22, 2017, discusses the licensee's current procedure for active component characterization and states that the procedure will be revised to reflect the change in the categorization approach. Please discuss any additional planned or anticipated changes to the licensee's categorization procedures based on the June 22, 2017, submittal, including items related to the:

   a. Use of a 10 percent margin to the importance measure thresholds in NEI 00-04,
   b. Use of absolute importance measures when re-evaluating previously categorized SSCs with an updated SPRA,
   c. Qualifications and training for the integrated decision-making panel members, and
   d. Approach to categorizing a high safety significance component based on the SPRA importance measures.

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – AUDIT PLAN RE: INCORPORATE SEISMIC PROBABILISTIC RISK ASSESSMENT INTO THE 10 CFR 50.69 CATEGORIZATION PROCESS (CAC NOS. MF9861 AND MF9862; EPID L-2017-LLA-0248) DATED DECEMBER 4, 2017

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