Enclosure 1

Basis of the Proposed Change

Table Of Contents

LIST OF TABLES

1 SUMMARY DESCRIPTION

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 (SSCs) for Nuclear Power Plants.” The proposed amendment would incorporate the use of the peer reviewed; plant-specific VEGP seismic PRA (SPRA) into the previously approved 10 CFR 50.69 categorization process, as allowed by the NRC endorsed industry guidance. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The 10 CFR 50.69 categorization process has been reviewed and approved by the NRC for Plant Vogtle. Categorization includes an integrated assessment of total risk and the regulations and categorization guidance allows licensees to implement different approaches depending on the scope of their PRA models. The currently approved risk assessment tools are:

1. Internal event PRA for internal risk
2. Fire PRA for fire risk
3. Seismic margin analysis for seismic risk
4. Individual Plant Examination of External Events (IPEEE) screening to asses risk from other external hazards (high winds external floods)
5. Safety Plan to assess shutdown risk

This proposed amendment request **only substitutes a seismic PRA in place of the SMA to assess seismic risk**. This type of change was envisioned by the regulations and guidance as new PRA tools became available. All other aspects of the program remain as the NRC approved in Reference 6. It is important to note the Vogtle program was approved by the NRC using a detailed pilot plant process taking over two years. As recently as the summer of 2016, the NRC conducted a pilot inspection of the process with favorable results (Reference 7).

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (NRC) has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. Those SSCs necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as “special treatments,” designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Special treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related "and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

The previously approved VEGP 50.69 categorization process conforms to the guidance in NRC RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006 (Reference 3). The categorization process also conforms to the guidance in NEI 00-04, "1 0 CFR 50.69 SSC Categorization Guideline," Revision 0 dated July 2005 (Reference 4), as endorsed by RG 1.201. With this change, to utilize the Seismic PRA model rather than the seismic margins approach (SMA), the VEGP categorization process will continue to conform to these guidance documents.

2.2 REASON FOR PROPOSED CHANGE

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, Probabilistic Risk Assessment (PRA) addresses credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner. To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.
The rule contains requirements on how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety significance is performed by an integrated decision-making process, as described by NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline” (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability, and is a function of how SSC is categorized. Finally, assessment activities are conducted to make

adjustments to the categorization and treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees 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, the rule allows a reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows an improved focus on equipment that has safety significance resulting in improved plant safety.

The VEGP 10 CFR 50.69 categorization process has previously been reviewed and approved by NRC (Reference 6). The proposed change implements a modification to the process, as allowed by the 10 CFR 50.69 guidance endorsed by NRC in Regulatory Guide 1.201 (Reference 2), to incorporate use of the peer reviewed plant-specific VEGP seismic PRA (SPRA).

2.3 DESCRIPTION OF THE PROPOSED CHANGE

SNC proposes the addition of the following condition to the operating license[s] of VEGP Unit 1 and Unit 2 to document the NRC's approval of the use 10 CFR 50.69.

> SNC is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the license amendment dated [DATE] using PRA models to evaluate risk associated with internal events including internal flooding, internal fire, and seismic events.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

3 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
> (i) A description of the process for categorization of RISC–1, RISC–2, RISC–3 and RISC–4 SSCs.
> (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i).
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

The above information was previously provided to NRC as part of the VEGP pilot application of 10 CFR 50.69 (Reference 5). The VEGP 10 CFR 50.69 categorization process (overall process, including active and passive categorization elements) has been reviewed and approved by NRC (Reference 6) and NRC has performed an audit of its implementation (Reference 7) and found it to be in conformance with the criteria specified in the rule and in RG 1.201 (Reference 2).

In its review and approval of that application, NRC reviewed the technical adequacy of the VEGP internal events at power and internal fire Probabilistic Risk Assessment (PRA) models and approved their use for 10 CFR 50.69 categorization (Reference 6). The VEGP 50.69 process addresses seismic risk through the use of the Individual Plant Examination of External Events (IPEEE) seismic margin assessment (SMA) results, following the process defined in NEI 00-04 (Reference 1) and endorsed in RG 1.201 (Reference 2). The purpose of this license amendment request is to replace, within the approved VEGP 50.69 program, the use of the SMA process with use of the VEGP seismic PRA (SPRA), also in accordance with NEI 00-04 (Reference 1) and RG 1.201 (Reference 2) guidance. Therefore, the remainder of this technical evaluation is focused on establishing the technical adequacy of the VEGP SPRA for this application.

3.1 SEISMIC PRA TECHNICAL ADEQUACY EVALUATION (10 CFR 50.69(b)(2)(II))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The SPRA model described below has been peer reviewed and there are no PRA upgrades that have not been peer reviewed.

3.1.1 Seismic Hazards

The approved VEGP categorization process uses the SMA performed for the IPEEE in response to GL 88-20 (Reference 4) for evaluation of safety significance related to seismic hazards. Through this requested change, the VEGP categorization process will instead use the peer reviewed plant-specific VEGP seismic PRA model. The SNC risk management process ensures that the SPRA model used in this application reflects the as-built and as-operated plant for each of the VEGP units. Table 1 at the end of this enclosure identifies the applicable Seismic PRA model.

3.1.2 PRA Maintenance and Updates

The SNC risk management process, which was previously reviewed by NRC as part of the VEGP 50.69 approval (Reference 6). ensures that the applicable PRA models, including the SPRA model, used in this application continue to reflect the as-built and as-operated plant for each of the VEGP units. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

In addition, SNC has implemented a process that addresses the requirements in NEI 00-04 (Reference 1), Section 11, “Program Documentation and Change Control.” The process reviews the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades to any of the PRA models used in support of the VEGP 50.69 process will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.1.3 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA models used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the peer review processes as discussed in Section 3.2 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04 (Reference 1).

In the overall risk sensitivity studies SNC utilizes a factor of 3 to increase the unavailability or unreliability of LSS components. Consistent with the NEI 00-04 guidance (Reference 1), SNC performs both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs, including the SPRA once this amendment request is approved, for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The SPRA assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. If the VEGP SPRA model used a non-conservative treatment, or methods which are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. Only those assumptions or sources of uncertainty that could significantly impact the configuration risk calculations were considered key for this application.

Potentially key SPRA PRA model specific assumptions and sources of uncertainty for this application are identified and dispositioned in Table 3. The conclusion of this review is that no additional sensitivity analyses are required to address VEGP SPRA model specific assumptions or sources of uncertainty beyond those specified in NEI 00-04.

3.2 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(B)(2)(III))

The VEGP SPRA model has been assessed against RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Revision 2 (Reference 4). Specifically, the model was subject to a self-assessment and a peer review conducted in November 2014.

Table 1 summarizes the SPRA model results and provides the date of the industry peer review performed against RG1.200 (Reference 4). Table 2 provides a summary of:

- VEGP SPRA peer review Fact and Observation findings and disposition relative to the 50.69 application.

- Identification of and basis for any sensitivity analysis performed to address issues identified in the peer review findings.

Of the 73 peer review finding-level Facts and Observations (F&Os) listed in Table 2, 60 were associated with PRA Standard supporting requirements (SRs) that were deemed by the peer reviewers to be either Met or met at capability category II. This indicates, as can be seen from the finding details, that these findings deal with relatively focused issues that have been adequately dispositioned within the reviewed methodologies, for the SPRA and for the 50.69 application. Many of these were documentation-related.

The remaining 13 finding-level F&Os are associated with SRs deemed by the peer reviewers to be not met, or not met at capability category II. These are as follows:

Findings associated with Not Met/CC-I SRs	
SR	Findings
SHA-C4	12-18, 12-36
SHA-H1	12-18, 12-36
SHA-I1	12-15
SHA-J1	12-1, 12-2, 12-11, 12-16
SHA-J3	12-8
SFR-A2	14-1, 14-7, 14-10
SPR-B2	16-4, 16-6

As this list indicates, there are only 7 not met / capability category I SRs associated with the finding F&Os.

- Of these, 5 are seismic hazard-related SRs, for which the findings are associated with: (a) inadequate documentation of the hazard analysis performed; (b) demonstration that sufficient consideration has been given to more recent geologic events and associated modeling; or (c) sensitivity calculations for the models and parameters used in the site hazard. The documentation items have been addressed, as noted in the dispositions for the affected findings in Table 2. *[NOTE: At the present time there are some items that are being addressed part of the VEGP Units 3 and 4 closure.]* The other items are site-related and are being addressed as part of the development of the Vogtle Units 3 and 4 SPRA which will be peer reviewed in early 2017. The resolutions of these items for Units 3 and 4 demonstrate that there is no significant impact on the site hazard information as used in the plant fragilities and hazard curve development. Because these are site hazard issues, there is also no significant impact for Units 1 and 2, and closure of these items for Units 1 and 2 will be completed as part of a future SPRA model update.

- One of the SRs is fragilities-related. Two of the 3 findings associated with this SR deal with conservatisms that the reviewers noted, which have now been addressed within the analytical methodology that the peer reviewers found acceptable. The remaining finding is associated with a specific polar crane fragility issue, which has also been addressed within the reviewed methodology.

- One of the SRs is PRA modeling-related. The 2 findings associated with this SR are related to implementation of the seismic performance shaping factor approach in the human reliability analysis. The comments in those findings have been addressed and implemented in the SPRA model, within the reviewed methodology, without significant impact on the results.

The information in the tables identified above demonstrate that the PRA is of sufficient quality and level of detail to support the categorization process, and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required 10 CFR 50.69(c)(1)(i).

3.3 RISK EVALUATIONS (10 CFR 50.69(B)(2)(IV))

The VEGP 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04 (Reference 1). The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions, and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in Section 8 of the guidance will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, human errors, etc.). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data, and provide timely insights into the need to account for any important new degradation mechanisms.

4 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations at 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 Reactors."
- NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, April 2015.
- Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, US Nuclear Regulatory Commission, March 2009.

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

SNC proposes to modify the licensing basis to amend 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 (SSCs) for Nuclear Power Plants" to include use of the VEGP SPRA in place of the VEGP IPEEE SMA. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed change replaces the use of the VEGP SMA with use of the peer reviewed VEGP SPRA within the NRC approved risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The use of an SPRA in place of an SMA is allowed by the 50.69 process guidance defined in NEI 00-04 (Reference 1) as endorsed by NRC in RG 1.201 (Reference 2). The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change continues to permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change will continue to permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, SNC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 CONCLUSIONS

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6 REFERENCES

1. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.
2. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
3. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991.
4. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, US Nuclear Regulatory Commission, March 2009.
5. Ajluni, M. J. (Southern Nuclear Operating Company) to U. S. Nuclear Regulatory Commission, Vogtle Electric Generating Plant Pilot 10 CFR 50.69 License Amendment Request, August 31, 2012 (ADAMS Accession No. ML 12248A035).
6. U. S. Nuclear Regulatory Commission to C.R Pierce. (Southern Nuclear Operating Company), Vogtle Electric Generating Plant, Units 1 And 2 -Issuance Of Amendments Re: Use of 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473), December 17, 2014 (ADAMS Accession No. ML14237A034).
7. U. S. Nuclear Regulatory Commission to B.K Taber (Southern Nuclear Operating Company) Vogtle Electric Generating Plant - NRC Evaluation Of Risk Informed Categorization And Treatment Of Systems, Structures, And Components, Inspection Report 05000424/2016008 AND 05000425/2016008, August 10, 2016 [ADAMS Accession No. ML12061A245]

Table 1: Seismic PRA Model Summary Information

Unit	Model	Baseline CDF	Baseline LERF	Comments
1	VEGP-SPRA-1, 12/31/16	2.9E-6/yr	7E-7/yr	one model applicable to both units
2	VEGP-SPRA-1, 12/31/16	2.9E-6/yr	7E-7/yr	one model applicable to both units

The peer review was performed in March 2014 against RG 1.200 R2.

Table 2: Disposition and Resolution of SPRA Peer Review Findings

Finding Number	Supporting Require-ment(s)	CC	Finding Description	Finding Basis	Finding Resolution	Disposition for 50.69 and other applications
11-3	SHA-E2	II/III	While variability in the mean base-case Vs profile is incorporated in the site response analysis, no epistemic uncertainty in the base-case profile is represented. Documentation of the justification for this assessment should be expanded. (This F&O originated from SR SHA-E2)	To maintain hazard-consistent ground motion hazard at the control point, the site response analysis needs to incorporate appropriate epistemic uncertainty and aleatory variability in its inputs. The Vs profile for the Vogtle Units 1&2 site is represented by a single Vs profile, indicating there is no epistemic uncertainty in the mean base-case profile. Documentation of this assessment needs to be expanded. Discussion with staff indicates that consideration of the combined data for the Vogtle site (Units 1&2, Units 3&4, ISFSI) provides sufficient confidence that a single mean base-case profile characterizes the site. This conclusion is based on the quantity and quality of the combined data and an evaluation showing the site is relatively uniform with respect to Vs. For some depth ranges, data from the nearby Savannah River Site (SRS) are used to support the profile interpretation. Bechtel Document 23162-000-G65-GEK-00010 (SNC #SV0-GB-X7R-011-001) presents summaries of velocity data, but does not provide sufficient	Expand documentation to demonstrate that a single base-case Vs profile adequately represents the Units 1&2 site. Or if that is not the case, include epistemic uncertainty in the characterization of Vs profile and evaluate the impact on control point ground motions.	There is an abundance of site-specific Vs data from VEGP Units 3&4, which reduces epistemic uncertainty to an insignificant level. At a future SPRA model maintenance update the documentation will be expanded to demonstrate that a single base-case shear-wave velocity (Vs) profile adequately represents the Vogtle site. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed applications.

				information to support the lack of epistemic uncertainty at the Units 1&2 site over the complete depth range of the Vs profile. This would typically require multiple measurements throughout the depth range that provide a consistent picture of natural variability about a single mean base-case profile. The technical basis and justification that a single base-case profile is appropriate should be provided in more detail. This should include the basis for applying conclusions from other Vogtle locations to the Units 1&2 site. [A related Suggestion 11-2 addresses specifically potential epistemic uncertainty in the Blue Bluff Marl stratum.]		
11-8	SHA-E2	II/III	Upper crustal site attenuation of ground motion (kappa) is, generally, an uncertain parameter. Thus, to maintain hazard-consistent ground motion at the control point, this uncertainty should be incorporated in the site response analysis, or the basis for not including it should be provided. In either case, the technical basis and justification should be documented. (This F&O originated from SR SHA-E2)	Calculation X2CFS129 Ver2 notes that the damping associated with the base-case profile corresponds to a total kappa value for the soil column of 0.01 sec. The report does not address epistemic uncertainty in kappa. In discussion with staff during the peer review, it was noted that randomization of the damping associated with the profile layers represents both random variability and epistemic uncertainty. It was also noted that kappa was expected to be small for the Vogtle site and uncertainties in that small value would not be expected to have a significant impact on site amplification. Staff also noted that the approach used had been reviewed by the NRC for the Vogtle ESP and COLA.	Provide a basis in the documentation for representing base-case kappa at the site by a single value. The basis might include sensitivity analyses to show the impact of epistemic uncertainty in kappa.	At a future SPRA model maintenance update the basis for representing base case kappa at the site by a single value will be documented and referenced to the Vogtle ESP and relevant literature. Deep soil damping ratio was taken as 1% based on the analysis that was done earlier for VEGP Units 3&4 for the ESP and COL applications. Although there is still epistemic uncertainty about that estimate, given that it is not backed up by measured site-specific data, the epistemic uncertainty is likely to be very small

				The SPID (EPRI, 2013) provides guidance accepted by the NRC for response to NTTF 2.1 Recommendation: Seismic that indicates kappa is difficult to measure and thus subject to large uncertainty (SPID Section B-5.1.3.2). Documentation of the technical basis for kappa characterization should be expanded.		when compared to other sources of uncertainty such as rock PSHA and the basis would be documented. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed applications.
12-1	SHA-J1	Not Met	As part of the PSHA implementation, the analyst has different alternatives for modeling the earthquake occurrences in the calculations. The PSHA documentation does not describe the approach that was used to model earthquakes. (This F&O originated from SR SHA-J1)	The approach that was taken to model earthquakes in the PSHA calculation was not identified. There are two basic alternatives that can be used to model earthquake events; as extended fault ruptures, or as point sources. The approach that is used influences how the CEUS ground motion model is implemented. No documentation is provided on either of these subjects (earthquake source modeling and use of the ground motion attenuation models). From questions posed to the PSHA analysts, it is our understanding that earthquakes were modeled as point sources and the appropriate ground motion aleatory uncertainty was used in the calculation.	Documentation should be provided that describes how seismic sources are modeled in the PSHA (i.e., how the SSC and GMMs) were implemented in the Vogtle PSHA.	Background source earthquakes were modeled as point sources with appropriate correction factors for distance and aleatory uncertainty. At a future SPRA model maintenance update the documentation will be enhanced to include additional discussion of the modeling used. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed applications.

12-11	SHA-J1	Not Met	As part of the PSHA implementation, the analyst has alternatives for modeling the earthquake occurrences in the calculations. The PSHA documentation does not describe the approach that was used to model earthquakes in RLME sources. (This F&O originated from SR SHA-J1)	The PSHA analysts were asked to describe the approach that was used to model earthquakes in the Charleston RLME seismic source. The response indicated that earthquakes in the Charleston RLME source were modeled using 'pseudo faults'. The PSHA report does not: 1. Describe that a 'pseudo fault' approach was used to model earthquakes in the Charleston RLME source. 2. Provide a definition of 'pseudo faults'. 3. Describe how the 'pseudo fault' approach was implemented for the Charleston RLME seismic source (e.g., what was the fault spacing that was used; how was the earthquake rate distributed to the faults, etc.). 4. Document the fault rupture model that was used. 5. Describe how earthquake events are distributed on the faults.	Provide a description of the earthquake modeling approach that was used to model the Charleston RLME seismic source and how the approach was implemented.	At a future SPRA model maintenance update a description of how RLME sources were modeled with pseudo-faults, including spacing, rate distribution, rupture model, and location distribution) will be written. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed applications.
12-15	SHA-I1, SHA-I2	Not Met	A screening assessment was performed for soil liquefaction and is described in seismic fragility calculation (PRA-BC-V-14-025). A screening assessment was not performed for other potential seismic hazards. (This F&O originated from SR SHA-I1)	A screening analysis was not performed for hazards such as settlement, fault displacement, tsunami, seiche, etc. It is anticipated these other seismic hazards will be screened out.	A screening analysis for other seismic hazards should be performed and documented as part of the PSHA and SPRA. It is expected that information in the FSAR for Vogtle 1 & 2 and in the COLA for Units 3 & 4 can be used to support this requirement.	This evaluation was done for the V3&4 COLA and is noted in the ESP SAR. The Vogtle 3&4 evaluation is applicable to, and has been cited in, the Vogtle 1&2 SPRA Fragility report. This issue is resolved. No impact on 50.69 or other risk-informed applications.

12-16	SHA-J1	Not Met	The Vogtle PSHA has gone through a number of changes and revisions since 2012 due to changes in models, input data, etc. As new calculations were performed and reports generated, sensitivity results, were not carried forward. As a result, there does not exist a current report that includes all PSHA results, deaggregations, etc. that is based on the current PSHA model. (This F&O originated from SR SHA-J1)	The documentation of the PSHA is provided in a collection of documents that were prepared in the 2012-2014 time frame. There does not exist a single document that contains a set of results that is based on the current PSHA model.	Prepare a complete and up-to-date PSHA document that includes all results, sensitivity calculations, deaggregation results, etc. that is based on the current model.	At a future SPRA model maintenance update an overall report will be written that summarizes the existing seismic hazard results (including contributions by source and hazard deaggregations) and includes the sensitivities and responses developed for peer review comments. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed applications.
12-18	SHA-B2, SHA-C4, SHA-H1	I/II Not Met Not Met	The Vogtle PSHA is based on the CEUS SSC seismic source model which was completed in 2012. The SSC model was a developed at a regional scale that was based on data gathered up until about 2010. (Note, the date when data was gathered varied; for example the earthquake catalog was complete through 2008.) In the sense that the CEUS SSC model was not specifically performed as a site-specific PSHA for the Vogtle site. (This F&O originated from SR SHA-B2)	As part of a site-specific PSHA, there is a need to gather, review and evaluate new geological, seismological, or geophysical information or information that is defined at a scale that was not considered in the development of the CEUS SSC model. As part of the Vogtle SPRA, no effort was made to gather up-to-date and local (local to the Vogtle site) information to evaluate whether any new information has become available on active faulting and/or the development new seismic sources or the revision of sources in the CEUS SSC model in the vicinity of the Vogtle plant. Since up-to-data was not gathered, consideration of alternatives could not be addressed.	A data gathering effort should be undertaken to identify new information that post-dates the CEUS SSC data collection effort. The data gathering effort should also look for information local to the Vogtle site region that was not considered, or at a scale that was not addressed as part of the CEUS SSC regional evaluation. Some of this information may be available in the COLA for Vogtle Units 3 & 4.	At a future SPRA model maintenance update new geological, seismological, or geophysical information that might be relevant to seismic hazard at the site will be collected and evaluated. This effort will be concentrated on the site region, since the CEUS SSC model was developed. Documentation for Vogtle Units 3&4 will be used, as well as other available technical information. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed

						applications.
12-2	SHA-J1	Not Met	The method that is used in the Vogtle PSHA to estimate the soil site hazard is not described or referenced. (This F&O originated from SR SHA-J1)	For soil sites, the soil hazard is generally (though not exclusively, since other methods could be used) determined in two steps; probabilistic rock hazard results are estimated which are then combined with probabilistic estimates of the site response. The method used in the Vogtle PSHA to estimate the soil hazard is not described.	The documentation should include a description of the methodology that is used to combine the rock hazard results and the site amplification factors to determine the soil hazard at the Vogtle site.	Approach 2A in NUREG/CR-6728 is used to calculate the mean surface hazard at the foundation and surface horizons. The comparison of control point UHRS (at surface), which was developed using Approach 2A, with the one developed using Approach 3 will be documented at a future model maintenance to demonstrate that, in the case of Vogtle, the two approaches yield very similar results; therefore, the Approach 2A hazard can be used for SPRA without loss of accuracy. This issue is resolved. No impact on 50.69 or other risk-informed applications.
12-22	SHA-E2	II/III	The site response Calculation X2CFS129 Ver. 1 (2012) and Ver. 2 (2014) does not describe a framework for evaluating and characterizing sources of aleatory and epistemic uncertainty and how the approach was implemented. (This F&O originated from SR SHA-E2)	The site response calculation does not present a clear description of how aleatory and epistemic uncertainties are identified and evaluated. As a result it is difficult to track the propagation of uncertainties is carried out in the site response analysis. It is worth noting that there is some epistemic site response uncertainty that is accounted for in	A framework and approach for evaluating and modeling uncertainties in the site response should be developed and implemented. The site response calculation documentation should fully describe the methodology and its implementation.	At a future SPRA model maintenance update the documentation of the site response analysis would be expanded to include a more detailed description of the sources of aleatory and epistemic uncertainty and their propagation through the model. The analysis included the identification and propagation of

				the rock GMPEs.		uncertainties, therefore, the update is not expected to have an impact on the hazard mean or fractiles. Resolution of this issue will not have a significant impact on SPRA results. No impact on 50.69 or other risk-informed applications.
12-23	SPR-E5	II	The quantification process has included the uncertainties in the seismic hazard, fragility and systems-analysis elements of the SPRA. The results in Table 5.1 are internally inconsistent and are inconsistent with the results reported in Sections 3 and 4 for CDF and LERF, respectively. (This F&O originated from SR SPR-E5)	Table 5-1 presents the results of three different uncertainty calculations for CDF and LERF. In addition, point estimates for CDF and LERF are calculated and reported in Section 5.1.1. Thus the table reports two estimates of the mean CDF and LERF respectively from different uncertainty calculations and a 'Point Estimates' result for each. All of these results are different than the point estimate (approximate mean) reported in Sections 3 and 4 for CDF and LERF, respectively. The documentation in the report does not describe the basis (inputs) for these calculations, or offer an interpretation of the results.	Develop and document an understanding of the earlier point estimate results for CDF and LERF (as reported in Sections 3 and 4) and of uncertainty results.	Additional detail has been added to the QU report to document the uncertainty, importance, and sensitivity analyses and relate the uncertainty analysis mean CDF and LERF to the point estimate values. This finding is resolved. No impact on 50.69 or other risk-informed applications.
12-24	SPR-E5	II	The Quantification report does not provide documentation of the uncertainty analysis results. (This F&O originated from SR SPR-E5)	The uncertainty analysis is presented in Section 5.1 with the results reported in Table 5.1. The report provides limited discussion of the results and the insights that might be gained from them. The two sets of results that are reported in Table 5-1 are not discussed in terms of their relationship to each other. For instance the mean values should be the same (but are not). The	Provide documentation of the uncertainty analysis that describes the results, how they are being interpreted and the insights that are derived from them.	Additional detail has been added to the documentation of the seismic plant response model, model implementation, and quantification in the QU report. In addition, the uncertainty, importance, and sensitivity analyses are described in more detail.

				uncertainty estimates provide insight to the total uncertainty and the contribution of the basic event uncertainty to the total. In addition, neither Table 5.1 or the discussion identifies what is the 'final' uncertainty result that includes the propagation of uncertainties of all elements of the SPRA to the estimates of CDF and LERF.		This finding is resolved. No impact on 50.69 or other risk-informed applications.
12-26	SPR-E5	II	There are differences in the results for CDF and LERF that are reported in Table 5.1. A possible contributor to these differences may be due to the number of Monte Carlo simulations that were performed. (This F&O originated from SR SPR-E5)	The report does not present the results of sensitivity calculations with regard to the number of Monte Carlo simulations that are needed to produce stable results. It is our understanding from discussion with the PRA staff that these types of sensitivity calculations were performed.	Document the results of sensitivity calculations on the number of Monte Carlo simulations required to produce stable results.	Monte Carlo sensitivity runs were performed to document the uncertainty analysis. *[DRAFT; additional sensitivity results to be described]* This finding is resolved. No impact on 50.69 or other risk-informed applications.
12-27	SPR-F2	Met	Documentation should be provided that describes how the plant model analysis is quantified. (This F&O originated from SR SPR-F2)	The current quantification document does not provide a clear description of the how the plant model is quantified. For example the discussion does not identify how calculations are performed, what the limitations of these quantifications are and how they affect the results.	Provide clear and complete documentation of the approach used to quantify the seismic plant response model, to perform the risk quantification, uncertainty analysis, and importance analysis.	The QU report documentation has been updated to describe the quantification process, including the technique for combining cutsets over the 14 acceleration intervals, and obtaining the importance measures. This finding is resolved. No impact on 50.69 or other risk-informed applications.

12-29	SPR-E2	Met	The Quantification report provides limited documentation of the process and methods that were used to perform the uncertainty analysis. (This F&O originated from SR SPR-E2)	There is limited documentation of the process and the numerical methods that were used to perform the uncertainty analysis. Based on the documentation that is provided and discussions with the PRA staff there is limited but not complete understanding of the methods that were used and the relationship of these methods to the results were obtained (reported in Table 5.1). In some cases (as described in the documentation) the results from the uncertainty analysis (Table 5.1) are not the same as the results reported in Sections 3 and 4 for CDF and LERF (though this connection is not clearly stated in the report). However, it would seem the results in Table 5.1 should be internally consistent.	Document the process and methods that were used to perform the uncertainty analysis. Where appropriate document where consistencies and potential inconsistencies in results might be expected.	Additional detail has been added to the QU report to document the uncertainty, importance, and sensitivity analyses and relate the uncertainty analysis mean CDF and LERF to the point estimate values. This finding is resolved. No impact on 50.69 or other risk-informed applications.
12-31	SPR-F1	Not Met	The standard requires a level of documentation that provides an understanding of the seismic plant response model and the quantification. This requirement is not met. (This F&O originated from SR SPR-F1)	There is limited documentation that describes the seismic plant response analysis and quantification; how the model was implemented, how the quantification was performed and a discussion of the analysis results. To meet this requirement, the documentation must be in considerable detail in order to support the review process and future updates. Part of the documentation should include a detailed discussion of the results, sensitivity calculations, and the uncertainty analysis.	Documentation should be provided in sufficient detail that describes the seismic plant model, how it is implement and quantified.	Additional detail has been added to the documentation of the seismic plant response model, model implementation, and quantification in the QU report. In addition, the uncertainty, importance, and sensitivity analyses are described in more detail. This finding is resolved. No impact on 50.69 or other risk-informed applications.

12-32	SPR-F3	Met	The documentation of the sources of model uncertainty and a description of the analysis assumptions is not complete in the SPRA quantification report. In addition, there is not a clear description of the uncertainty analysis and the contributors to the total uncertainty beyond a simple report from UNCERT. (This F&O originated from SR SPR-F3)	The purpose of this supporting requirement is that documentation should be presented that addresses the sources of epistemic (knowledge) uncertainty that are modeled and their contribution to the total uncertainty in CDF and LERF. In addition, the documentation should discuss elements of the seismic plant model where there may be latent sources of uncertainty that are not modeled and assumptions that are made in performing the analysis.	Document and discuss the contribution of the different sources of uncertainty that are modeled in the SPRA.	The documentation of the uncertainty analysis has been expanded in the Quantification report, App. G. A discussion of sources of model uncertainty has been added to the report, and potentially important sources have been addressed in the sensitivity analysis. This finding is resolved. No impact on 50.69 or other risk-informed applications.
12-36	SHA-B3, SHA-C4, SHA-H1	I/II, Not Met Not Met	As part of a site-specific PSHA, an up-to-date earthquake catalog should be used. The CEUS SSC study involved the development of a comprehensive earthquake catalog based on data through 2008. The Vogtle site-specific PSHA should consider the impact SSC of any additional seismicity since 2008 up to the time the study started. (This F&O originated from SR SHA-C4)	As part of the Vogtle PSHA an effort was not made to gather data on earthquakes that occurred since 2008. As such, the analysts did not assess whether more recent seismicity is consistent with the characterization parameters estimated as part of the CEUS SSC study (NRC, 2012). We note that as part of the Vogtle PSHA, calculations were performed to recompute the seismic hazard at the site to take into account changes in the CEUS SSC earthquake catalog through 2008 that were made following the completion of the CEUS SSC study. These changes reflect the identification of reservoir induced seismicity earthquakes and the re-interpretation of the location of some earthquakes in the Charleston, SC area that occurred in the 1880's (EPRI, 2014). References	An up-to-date earthquake catalog for the Vogtle site region should be developed to assess whether modifications to the seismic source recurrence parameters or required. The updated catalog, resources used in compiling the update and the results of the evaluation should be documented as part of the PSHA. If more recent seismicity is not consistent with the existing CEUS SSC seismic source parameters, the parameters should be updated and the PSHA should be updated.	A future model maintenance task will compare seismicity rates from the EPRI-CEUS project with seismicity rates calculated from an updated catalog in the vicinity of the Vogtle site. The EPRI-CEUS project rates will be calculated from available grid cell data (through 2008), the updated catalog will be calculated from seismicity data (2009—2016), for a circular area (e.g. radius 320 km) around the Vogtle site. It is expected that the comparison will show no statistically significant increase in seismicity in the vicinity of the site since 2008. Resolution of this issue will not have a significant impact on SPRA results.

				EPRI (2014). Review of EPRI 1021097 Earthquake Catalog for RIS Earthquakes in the Southeastern U. S. and Earthquakes in South Carolina Near the Time of the 1886 Charleston Earthquake Sequence, transmitted by letter from J. Richards to R. McGuire on March 5, 2014.		There is no impact on the 50.69 application or other risk-informed applications.
12-8	SHA-J3	Not Met	A foundational element of PSHA as it has evolved over the past 30 years is the development and implementation of methods to identify, evaluate, and model sources of epistemic (model and parametric) uncertainty in the estimate of ground motion hazards. As such fairly rigorous analyses are carried out (SSHAC studies) to quantitatively address model uncertainties. At the same time there is within any analysis sources of uncertainty that are not directly modeled and assumptions that are made for pragmatic or other	The documentation of the sources of model uncertainty analysis and a description of the analysis assumptions is not complete in the PSHA report in its current form such that a clear understanding of the contribution of individual sources of uncertainty to the estimate of hazard are understood. Limited information on the contribution of seismic sources to the total mean hazard is presented, but information on the contributors to the uncertainty is not provided. With respect to addressing model uncertainties and associated assumptions there are some examples that can be identified in the Vogtle PSHA. For example, in	The resolution to this finding could involve: 1. Documentation and discussion of the contribution of different sources of uncertainty that are modeled in the PSHA. The documentation of the contribution of different sources of uncertainty can be shown by means of 'tornado plots' that quantify the sensitivity of the hazard at different ground motion levels to the various branches in the logic tree. These plots show which sources of epistemic uncertainty are most important. It should include the source model uncertainty, ground motion model uncertainty, and	A future model maintenance task will document and discuss the contribution of different sources of uncertainty in seismic hazard that are modeled in the PSHA. This will include contribution of seismic sources (background and RLME) to total hazard, alternative background source models, alternative background source seismicity smoothing assumptions, alternative maximum magnitude values, alternative ground motion models, and alternative

			reasons. There are also sources of model uncertainty that are embedded in the context of current practice that are 'accepted' and typically not subject to critical review. For instance, in the PSHA it is standard practice to assume that the temporal occurrence of earthquakes is defined by a Poisson process. This assumption is well accepted despite the fact that it violates certain fundamentally understanding of tectonic processes (strain accumulation). A second practice is the fact that earthquake aftershocks are not modeled in the PSHA, even though they may be significant events (depending on the size of the main event). In the spirit of the standard it seems appropriate that sources of model uncertainty that are modeled as well as sources of uncertainty and associated assumptions as they relate to the site-specific analysis should be identified/discussed and their influence on the results discussed. As SPRA reviews and the use of the standard has evolved, it would seem the former interpretation is reasonable, but potentially incomplete. It is reasonable	the site response analysis the assumption is made that the 1D equivalent linear model (SHAKE type) to estimate the site amplification and ground motion input to plant structures is appropriate.	site response uncertainty. Currently, the total uncertainty is shown by the hazard fractiles, but it is not broken down to provide understanding as to what is most important. 2. Identification and discussion of model assumptions that are made.	site amplification models. For the latter, the alternative site amplification models will represent uncertainties in “site amplification; alternative soil profiles, estimates of soil parameters, etc.” (FOID 12-5). These contributions to uncertainty will be presented and discussed using “tornado plots”. Resolution of this issue will not have a significant impact on SPRA results. There is no impact on the 50.69 application or other risk-informed applications.

			from the perspective that documentation of the sources of model uncertainty and their contribution to the site-specific hazard results is a valuable product that supports the peer review process and assessments in the future as new information becomes available). Similarly, documenting assumptions provides similar support for peer reviews and future updates. The notion that model uncertainties and related assumptions that are not addressed in the PSHA is at a certain level an extreme requirement that may not be readily met and may not be particularly supportive of the analysis that is performed. For purposes of this review, the following approach is taken with regard to this supporting requirement: 1. The documentation should present quantitative results and discussion the sources of epistemic uncertainty that are modeled and their contribution to the total uncertainty in the seismic hazard. 2. The documentation should discuss elements of the PSHA model where their may be latent sources of model uncertainty that are not modeled and assumptions that are made			

			in performing the analysis. (This F&O originated from SR SHA-J3)			
14-1	SFR-A2	I	The conservatisms that exist in structural demand were not properly accounted for in the estimation of component and structure fragilities. (This F&O originated from SR SFR-A2) .	SFR-A2 requires that seismic fragilities be based on plant-specific data and that they are realistic and median centered with reasonable estimates of uncertainty. The structural response factor used in all component fragilities reviewed is reported as 1.0. This factor will be greater than 1.0 because of the conservatism introduced in the demand through the structural analysis. Because of this, the component and structural fragilities are biased low. The fragilities developed for structures and components that are mounted in those structures will be biased low because the input structural demands include conservatisms. Time histories used for the SSI analysis have been processed such that each record envelopes the target UHRS. This will introduce some level of conservatism. The input motion at the control point has been scaled to produce resultant FIRS that envelopes the FIRS coming out of the site-consistent input motion analysis. In structure response spectra coming out of the SSI analyses were not peak clipped when computing anchorage demands. Structure response at the calculated equipment fragility levels is considerably higher than the 1E-4 UHRS considered in the building	Account for conservatism in the building response analyses in the structure response factor for component fragility evaluations. Use clipped spectra for assessing anchorage capacities.	ASCE 4 methodology is used in clipping instructure response spectra for evaluating anchorage fragilities, and the methodology is documented in the fragility notebook. Structure response is dominated by the soft soil on which Vogtle 1 and 2 structures are founded. This would cause higher damping at higher hazard frequency levels and lead to stress similar to the stress calculated for the buildings at 1E-4. As a result the structural response factor is close to 1 and is accounted for appropriately in the fragility evaluations. The input motion at the control point in the SSI analysis is scaled to produce resultant FIRS that reasonably matches the FIRS from the site-consistent input motion analysis. Clipped in-structure response spectra are used for evaluating anchorage fragilities and the documentation has been updated accordingly.

				response analyses. The structure will have additional cracked shear walls and higher associated levels of damping at these higher ground motions.		This finding is resolved, and resolution does not affect the SPRA results. No impact on 50.69 or other risk-informed applications.
14-10	SFR-A2	I	Significant conservatisms were noted in several sampled fragility calculations. (This F&O originated from SR SFR-A2)	In the fragility calculations of heat exchangers (PRA-BC-V-14-009 Appendix A), nozzle loads significantly contribute to the seismic demands which form the basis for the median capacities. Based on in-plant walkdowns by the peer review teams and also noted in the walkdown report, the piping is well supported in all directions and will not impose significant nozzle loads during a seismic event. The CCW and ACCW capacities are below the 2.5g screening level and are significant contributors to risk so more realistic fragilities are required. Battery rack 11806B3BN3 in calculation PRA-BC-V-14-010 Appendix J2 is governed by GERS capacity. The GERS capacity is taken to be 1g, which corresponds to a frequency of 1 Hz. This is not realistic. The actual capacity is about 4g. The median capacity reported in the calculation is well below the 2.5g screening level and is not realistic. The median capacity reported for the Turbine Driven Auxiliary Feedwater Pump is reported in Calculation PRA-BC-V-14-008 as 1.56g. This fragility is based on the seismic qualification	Realistic nozzle loads should be determined for fragility evaluation of heat exchangers. The equipment capacity factor should be based on the frequency range of interest. That frequency range of interest is centered at the fundamental frequency of the pump, and considers some uncertainty in that frequency.	The CCW and ACCW heat exchanger capacities have been updated to reflect realistic nozzle loads. The equipment fragilities have been updated to account for appropriate frequency, and uncertainty has been considered in these updates. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				document. The frequency range of interest for the fragility evaluation should be centered around the fundamental frequency of the assembly and not consider the entire frequency range.		
14-14	SFR-G2	Met	The iterative process used for developing realistic fragilities is not well documented. (This F&O originated from SR SFR-G2)	In review of the seismic fragility calculation for the safety features sequencer (11821U3001), it was discovered that an iterative process was used. The initial fragility is based on EPRI 6041 screening methodology and an equipment capacity factor that is equal to the EPRI 6041 median capacity divided by the peak in structure demand. If this value is less than the screening capacity (2.5g), then the fragility may be refined by examining the component fundamental frequency. The fragility may be further refined by examining component specific qualification test reports. However, the fragility used in the logic tree by the systems analyst is generally the highest of these computed. This is reasonable and appropriate, however, this process is not described in the fragility notebook or fragility calculations.	Add a description of the iterative process for computing the component fragilities in the SPRA documentation	The description of the iterative process for computing fragilities has been documented. This finding is resolved. No impact on 50.69 or other risk-informed applications.
14-17	SFR-D2	Met	Inconsistencies and errors in NSSS fragility development. (This F&O originated from SR SFR-D2)	Fragilities for the Vogtle 1&2 Nuclear Steam Supply System (NSSS) are based on the results of the Westinghouse analysis of record (AOR) associated with the safe shutdown earthquake (SSE). In general, fragilities are developed through scaling of the SSE demands to the RLE and using the AOR seismic margins. Various deficiencies were noted in the development of the fragilities associated with these	Update SNC calculation no. PRA-BC-V-14-015 to incorporate corrections and enhancements.	The NSSS fragility calculations have been updated to reflect Westinghouse-provided critical loads and support capacities represented in the critical failure modes; The effect of inelastic energy absorption is factored in and documented in fragility calculation as appropriate. The Reactor

				components. Basis: The NSSS Seismic fragility evaluation (SNC calculation no. PRA-BC-V-14-015) includes detail calculations for each of the major NSSS components. It indicates that the critical failure modes for the components are controlled by the support capacities. During the Peer Review, the team members discussed these issues with SNC staff to obtain insights and develop potential resolution paths. Key issues included: (a) Basis for assumption that the support capacities represented the critical failure mode was not documented. SNC indicated that this was based on input from Westinghouse and NUREG-3360 and will update the fragility evaluation of provide this information. (b) Inelastic energy absorption was not credited to increase the median capacities - this does not result in realistic median capacities (overly conservative). (c) Reactor Coolant Pump fragility was based on consideration of the failure of the attached CCW piping, due to an assumption that a small-break/RCP seal LOCA was critical. It was learned during the Peer Review that failure in the system model was linked to a large-break LOCA, so the failure mode considered in the fragility evaluation is not consistent with the system model - SNC indicated that they will revise the fragility evaluation. (d) Reactor Internal fragility evaluation determined the demand based an average		Coolant Pump fragility has been updated to reflect the failure of the pump associated with LOCA; The reactor internals fragility has been updated in the calculation. The new fragilities have been reflected in the updated SPRA model. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				spectral acceleration over the range of 2 to 3 Hz, rather than using the peak acceleration in this range of the ISRS, and did not consider the contribution of higher modes. SNC indicated that this was done to avoid an overly conservative capacity, but agreed that the contribution of higher modes should be addressed, and will revise the calculation. (f) Control Rod Drive Mechanism fragility evaluation assumed that material stresses were the critical failure mode, and did not address the potential impact of deflections on rod drop. SNC indicated that information provided by Westinghouse (based on a Japanese testing program) indicated that the deflection levels associated with seismic loading does not impact rod drop, and agree to add this discussion to the calculation.		
14-20	SFR-E4, SPR-B9	Met Met	Seismic induced fire evaluations are not documented in the walkdown report or fragility calculations. (This F&O originated from SR SFR-E4)	The only mention for seismic induced fire evaluation is contained in the quantification notebook. Based on discussions during the peer review, it is understood that seismic induced fire was a key consideration during the walkdowns. However, detail of the walkdown procedure for fire following earthquake is missing. The write up should include team composition, methodology, screening criteria, and results,	Seismic induced fire is an important element of the fragility evaluation process and this should be clearly documented.	The seismic-induced fire and flood evaluations have been updated, and documented in the fragility and quantification report. This includes the details of the walkdown procedure used to evaluate the potential for seismically induced fires, including the methodology, screening criteria and results. This finding is resolved. No impact on 50.69 or other risk-informed applications.

14-4	SFR-D1	Met	A potential for sloshing induced inundation of the NSCW Pumps (11202P4007, 11202P408) and associated discharge motor operated valves (1HV11600, 11606, 11607, 11613) in the NSCW exists and was not identified either in the walkdowns or subsequent analysis. (This F&O originated from SR SFR-D1)	SFR-D1 requires that realistic failure modes of structures and equipment that interfere with the operation of that equipment be identified. The potential for earthquake induced sloshing of the water within the NSCW tower exists. From field walkdowns of the NSCW it was observed that there is a potential for sloshing of contents to potentially splash onto or flood the pumps and or motor operated valves on the attached discharge piping.	Evaluate the potential for flood induced failure of the NSCW Pumps or NSCW discharge MOVs.	The evaluation for potential flood induced failure of the NSCW pumps or the NSCW discharge MOVs has been performed and documented in the fragility calculation for the NSCW tower. There was no significant impact on the pump or MOV fragilities. This finding is resolved. No impact on 50.69 or other risk-informed applications.
14-5	SFR-D1	Met	The potential for seismically-induced differential settlements between structures was not addressed. (This F&O originated from SR SFR-D1)	Vogtle 1&2 is a soil site, with engineered fill from the rock interface to the finished grade. The in-scope Seismic Category I structures have foundations with varying embedment depths, ranging from surface founded (elev. 220 ft.) to a foundation embedment of 110 ft. (elev. 110 ft.). Since soils, including engineered fill, will consolidate/settle to some extent when subjected to high level earthquake ground motion, and the amount of settlement is proportional to the thickness of the soil layer under the foundation, the settlement of one structure relative to another structure is dependent on the depth of the foundation embedment. The Fragility Notebook (PRA-BC-V-14-025) does not address the potential differential settlement between buildings, or the potential effect on commodities (e.g., piping, electrical raceways, HVAC	Develop estimates of the differential settlements between adjacent structures and assess the fragility of commodities based on their ability to accommodate the associated differential displacements.	Documentation has been updated to include the effects of earthquake induced settlement; no significant differential settlements were computed between the structures. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				ducts, etc.) that cross the separation between adjacent structures. During the performance of the Peer Review, SNC personnel indicated that the consideration of differential settlements was not required, since the structures were founded on engineered fill.		
14-6	SFR-G2	Met	The results of the seismic gap/shake space walkdowns are not documented. (This F&O originated from SR SFR-G2)	The walkdown guidance provided in Appendix F (Checklists and Walkdown Data Sheets) of EPRI NP-6041 includes attributes of seismic gaps between structures which should be addressed in the performance of the walkdowns. These include the clearance between adjacent structures and the ability of any subsystems (e.g., piping, cable trays, HVAC ducts) spanning the gap to accommodate the differential seismic displacements. The Seismic Walkdown Report (PRA-BC-V-14-005) does not include documentation of the results/findings/observations associated with the inspection of the seismic gaps between structures or the subsystems spanning the gap. During the performance of the Peer Review, SNC personal indicated that inspection of the seismic gaps was included in the seismic walkdowns, but not explicitly described in the report. The ability of components to	Provide documentation of the results of the seismic gap walkdowns.	As noted in the basis, inspection of the seismic gaps was included in the seismic walkdowns. Piping across seismic gaps is designed with adequate flexibility to accommodate building motions, and pipe sleeves provide adequate gaps for piping movement. The documentation has been updated to reflect the inspections performed during the walkdowns. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				accommodate potential differential movement at the building separations is implied in the discussion of rugged components (piping, cable trays, and HVAC ducts) in Section 2.1 (Rationale for Screening) of the report. In addition, information from the Vogtle IPEEE Report (page 3.1-37) indicated that the seismic gaps had been inspected during the IPEEE.		
14-7	SFR-A2, SFR-F4	I, Met	The fragility evaluation for the Containment Polar Crane (in fragility notebook) did not address the impact of variation in the fundamental frequency on the applicable seismic demand. (This F&O originated from SR SFR-A2)	The determination of the fundamental frequency of structures and components involves a certain degree of uncertainty. This uncertainty must be accounted for in the determination of the seismic accelerations from the applicable in-structure response spectra (ISRS). Section 7.4 (Vogtle 1 and 2 Polar Crane) of the Fragility Notebook (SNC calculation no. PRA-BC-V-14-025) evaluates the polar crane as a potential seismic interaction source relative to the reactor vessel and other NSSS components inside the containment structure. In the determination of the vertical spectral acceleration applicable to the polar crane, the computed fundamental frequency falls within a valley in the applicable ISRS, on the low frequency side of the primary spectral peak. Uncertainty in the calculated frequency, and the contribution of high modes, could result in an increase in the applied vertical acceleration. During the performance of the Peer Review,	Update the fragility evaluation for the polar crane to address potential uncertainty in the fundamental frequency and the contribution of higher modes.	The fragility evaluation of the polar crane has been updated to address potential uncertainty in the fundamental frequency and contribution of higher modes. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				SNC personnel provided a written response indicating that it is appropriate to increase the applied acceleration by 50%, which will result in a 20% decrease in the median capacity of the polar crane.		
14-8	SFR-F3	II/III	Relay fragility calculations include conservative assumptions. (This F&O originated from SR SFR-F3)	The relay evaluation for the turbine driven auxiliary feedwater pump control panel in calculation PRA-BC-V-14-008 is based on a generic capacity for motor starters and contactors (intended for motor control centers) and an amplification factor associated with center of door panel response. Based on walkdown observations the relay is not mounted on the door panel so is likely on an internal bracket. The median capacity of 0.627g is well below the screening level and is not realistic. The relay evaluations in calculation PRA-BC-V-14-009 are governed by response in the vertical direction, and the in-cabinet amplification factors used in the calculation are associated with horizontal response. The resulting median capacities of 0.762g (Appendix M1) and 1.026g (Appendix M2) are well below the screening level and are not realistic.	Perform more realistic relay fragility evaluations.	The relay fragilities have been updated using the appropriate response and in-cabinet amplification factors, and are realistic. This finding is resolved. No impact on 50.69 or other risk-informed applications.

14-9	SFR-D2	Met	The seismic walkdown report includes a number of open items that are not are not traceable to a resolution (This F&O originated from SR SFR-D2)	The summary of the seismic walkdowns documents a number of issues identified during the performance of the walkdowns that required follow-up actions (31). These include spatial interaction issues, housekeeping issues, anchorage issues, valves having configurations that do not meet the EPRI guidelines, configuration issues, installation errors, etc. The Seismic Walkdown Report (PRA-BC-V-14-005) does not document how the issues identified during the walkdowns have been addressed, either in the field (e.g., correction of installation errors, resolution of housekeeping issues) or in the fragility evaluations (e.g., valve configurations, anchorage issues). During the performance of the Peer Review, the Peer Review Team provided a list of the walkdown issues to SNC personnel, and SNC provided a summary of how they were addressed. Most issues had been adequately addressed during the development of the SPRA, but it was determined that the following would require further effort for resolution: (a) Potential interaction between piping and deluge valve (page 19) - follow-up walkdowns required. (b) Anchorage configuration on inverter (page 40) - follow-up revision to fragility evaluation required (c) Overhead heater poses potential interaction issue (page 60) - follow-up walkdown required.	Perform resolution of open items and provide documentation of the resolution associated with each of the issues, either in the Fragility Notebook or the SPRA Database.	The noted walkdown issues have been evaluated and reflected in the revised documentation: - potential piping interaction; - the difference in inverter anchorage configuration; - potential interaction concerns with the overhead heater; this evaluation is in the fragility notebook in section 3.4.2. Valve operator heights & weights that were outside EPRI guidelines have been taken into account in the fragility analysis for these components. The Diesel Generator Exhaust Silencer was re-evaluated to the as-operated condition; the Diesel Generator Exhaust Silencer was re-evaluated from a systems functional perspective and it determined that its failure would not prevent DG operation, allowing the Exhaust Silencer to be screened from inclusion in the logic model. The fragility analysis for these components has been completed for the as built condition. This finding is resolved. No impact on 50.69 or other risk-informed

				(d) Valve operator heights/weights outside of EPRI guidelines (page 74) - follow-up walkdown required. (e) Diesel Generator exhaust silencer anchor bolt nuts (page 96) - not addressed in fragility evaluation, further evaluation required. (f) Valve operator heights outside of EPRI guidelines and potential lack of yoke support (page 105) - these valves are part of the unfinished scope described in the Fragility Notebook, which will be completed in the future. (g) Valve operator heights outside of EPRI guidelines (page 107) - further evaluation required.		applications.
16-1	SFR-F3, SPR-B4, SPR-E5	II/III, Not Met, II	The model presented for peer review did not incorporate the effects of relay chatter as the analysis was not yet complete. (This F&O originated from SR SPR-B4)	Relay chatter is consistently being observed as a significant contributor to risk profile in recently peer reviewed S-PRAs and it is therefore realistic to expect that relay chatter is a potential significant contributor. During the peer review it was discussed that the SPRA team does not believe relays will be a significant contributors but it was also said that this conclusion/ expectation is based on potentially crediting operator actions. Thus, the effects of relay chatter per se may be significant (and provide some insights) while the combination of relays and a number of HEP may not be.	Complete the analysis and incorporate the effects of relay chatter and similar devices in the PRA logic model.	The approach to screening and modeling of seismically-induced relay failures and chatter was provided to the peer review team and determined to have been performed appropriately; only the incorporation into the model of the impacts of relay chatter from unscreened relays was not complete. The final screening resulted in only 2 relays being incorporated into the model, with one having an operator action. Relay chatter fragilities and

						impacts have been incorporated into the seismic model, in a manner consistent with that used for other failures. This finding is resolved. No impact on 50.69 or other risk-informed applications.
16-10	SPR-B6	Met	The documentation about the walkdowns in support to seismic impact on HRA appear limited. (This F&O originated from SR SPR-B6)	There is only a short sentence supporting the discussion on alternative access pathways.	More detailed documentation is suggested to support the conclusion on accessibility, alternative route, availability of tools/keys, clear identification of equipment manipulated in each local action. Obviously, the goal of the enhanced documentation is not to convince the peer reviewer that the walkdowns were performed but rather to ensure that the analyst is fully convinced of the conclusions. Past SPRAs have shown examples of equipment needed for the HFE that was not in the SEL, or that has different actuators when manually actuated, or that needed ladders that were not easily accessible or that were close to block walls (or under ceiling that could collapse) that were not considered an issue because the block walls were not near safety related equipment (and therefore not addressed in the rest of the SPRA work). In this perspective, a more systematic documentation of the feasibility	Walkdown documentation on accessibility for operator actions, including photos, has been improved. Potential failure of block walls has been reviewed and documented. Required tools and equipment, such as ladders, have been identified with locations when needed. The documentation supports the seismic HRA assumptions and modeling. This finding is resolved. No impact on 50.69 or other risk-informed applications.

					and accessibility analysis for each of the HFE credited in the SPRA is suggested.	
16-11	SPR-E2	Met	Missing review of the potential for additional dependencies introduced by the SPRA models (QU-C1&2) (This F&O originated from SR SPR-E2)	It is understood that the investigation performed in internal events to identify potential HFE dependency has been relied upon in the Vogtle SPRA. The SPRA logic may identify additional dependencies trends that were not identified in the internal events.	As this exercise was apparently performed for the Fire PRA (as discussed during the peer review), it is suggested that a review of the potential for unforeseen dependencies trends is performed. As it is understood that the plan is to transition to a different dependency analysis method (based on HRA calculator), this may be addressed within the same transition as it is realistic to expect that not too many (if any) new dependencies would be identified.	A detailed quantitative HRA dependency analysis based on using the HRA calculator was performed and documented. There was no significant impact on results since human actions are not significant contributors in the Vogtle SPRA. This finding is resolved. No impact on 50.69 or other risk-informed applications.
16-12	SPR-E2	Met	Missing documentation of the review of non significant cutsets QU-D5. (This F&O originated from SR SPR-E2)	It is an industry expectation (as discussed in NEI peer review task force meetings) that review of the non significant cutsets is explicitly documented. Based on discussion during the peer review, two reviews were performed to validate the overall model and cutsets. The first was a random review of cutsets at midpoints and low significance for each of the %Gxx initiators to verify that the cutsets are valid cutsets, and that the patterns are appropriate. That is, if one cutset is valid, then another cutset with slightly different seismic failures (or random failures) should also be nearby. The second review, more importantly, lowered the median	It is understood that the SPRA documentation will be revised to incorporate explicitly the two reviews discussed in the basis for this F&O. It is also recommended to document the review of cutsets following guidance from the NEI peer review task force.	The QU report has been updated to document the review of both dominant cutsets and non-significant cutsets for both CDF and LERF. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				seismic capacity for each of the seismic initiators and some of the other seismic failures to ensure that the model would properly generate valid cutsets. For example, the LLOCA fragility was reduced to 0.5g to generate LLOCA cutsets. For ATWT, the fragility of the CRDs and RV internals were reduced to 0.5g to verify that valid ATWT cutsets were generated.		
16-15	SPR-E6, SPR-F2	Met, Met	Documentation of LERF model applicability review. (This F&O originated from SR SPR-F2)	The current documentation does not explain what are the basis for retaining the LERF logic and analysis unchanged within the SPRA logic. During the peer review the following explanation was provided by the SPRA team: "The internal events Level 2 notebook (Chapter 9) was reviewed to ensure that the definition of LERF would be appropriate for seismic events. Section 9.2 provides the LERF definition, including the use of a 12 hour time period for release after event initiation, to allow for evacuation. This time period is considered to be valid for Vogtle seismic events, particularly due to the very low population density in the area. Other characteristics, such as bypass and scrubbing, are the same for seismic as for internal events. The logic for the internal events LERF model is very straightforward, with sequences	Expand the documentation to ensure that the criteria used to retain the LERF analysis in the SPRA is explained so that the same applicability review can be performed following future potential revisions of the LERF modeling.	The LERF documentation in the QU report was expanded to describe the review of applicability of the internal events PRA LERF analysis to the seismic PRA. This finding is resolved. No impact on 50.69 or other risk-informed applications.

				from the CDF model ANDed with the appropriate LERF fault tree. This logic is also appropriate for seismic events."		
16-18	SPR-B8	III	Very small LOCA have been screened from the analysis based on walkdowns but little documentation exists of such walkdowns. (This F&O originated from SR SPR-B8)	The DB has a specific entry for the incore thermocouples and provides pictures of them. Still, in-core thermocouple tubing is not the only possible source of very small LOCA that is envisioned and the only documentation of addressing the other potential sources is in section 2.3.3 of the quantification notebook: "For Vogtle 1&2, the seismic walkdowns inspected and photographed a large sample of the small piping and tubing lines connected to the primary system in order to identify any weaknesses. The piping was judged to be rugged."	To the peer review team knowledge Vogtle is the only plant that has elected to perform dedicated walkdowns in support of not modeling very small LOCA. This would be a best practice but it also behooves to the SPRA team to provide detailed documentation of such walkdowns and how they supported a systematic evaluation of the potential sources of very small LOCA.	Additional information on the walkdown for very small LOCA has been added to fragility report to provide the basis for the VSLOCA screening. This finding is resolved. No impact on 50.69 or other risk-informed applications.
16-2	SFR-C1, SPR-E1	I/II, Met	Fragilities were not corrected to reflect the 2014 hazard used for quantification. (This F&O originated from SR SPR-E1)	The 2014 hazard was only used as input to FRANX for the final quantification. It is understood that the fragility estimates have been performed based on the 2012 hazard. While it is not expected nor recommended to regenerate all the fragility work with the new hazard, some consideration on the possible change in fragility due to the use of the newer hazard should be made.	During the peer review the SNC staff answered a question on this topic by performing an initial limited investigation of the effect on fragilities correction to reflect the 2014 hazard and concluded that the effect of this scaling is not insignificant (especially for LERF). It is recommended to continue and expand this investigation to make the quantification fully consistent with the fragility values.	The fragilities have been recalculated based on the 2014 hazard and the new values incorporated into the SPRA model and quantification. This finding is resolved. No impact on 50.69 or other risk-informed applications.

16-4	SPR-B2	Not Met	The effect of seismic impact on performance shaping factors is considered in the analysis by the usage of the Surry method. (This F&O originated from SR SPR-B2)	There is no assessment of the effect of changing the breaking points in the Surry method. The Surry method is based on methods used in the past at SONGS and Diablo Canyon and the 0.8g breaking point was developed for California earthquakes. In the Vogtle analysis there is no indications on whether the breaking point at 0.8g is also applicable to Vogtle. There are also no sensitivity analyses that would support whether a change in the breaking points is significant or not.	While it is recognized that the industry is still developing methods in support to this particular topic (e.g., recently published EPRI HRA method for external events), some additional considerations should be done to understand the effect of HEPs in the model rather than simply implementing the Surry method as is. Three examples for addressing this finding may be the following: 1. Perform sensitivities on the values of the multipliers and the g levels where the breaking point happens. 2. Use a different multipliers method with more breaking points. 3. Apply the impact of seismic specific PSF at the individual PSF level (i.e., timing, stress, etc.) in the HRA calculator.	The methodology used for the seismic HRA analysis is based on defining PSFs as a function of seismic hazard level (bins), which is consistent with the EPRI seismic HRA guidance in EPRI 1025294. The Integrated PSFs and bins (breaking points) have been updated to reflect seismic binning applicable to Vogtle, in accordance with this finding and consistent with the EPRI guidance. The updated values have been applied to both internal events HFEs and seismic-unique HFEs within the plant response model. There was no significant impact on the SPRA results. This issue is resolved. No impact on 50.69 or other risk-informed applications.
16-5	SPR-B1, SPR-F1	Met, Not Met	LOCA modeling and fragility selection not clearly documented. (This F&O originated from SR SPR-F1)	The selection of the fragility data used for all LOCA is discussed in Appendix B.2 of the quantification notebook but is confusing in the mapping of selected fragilities with specific failures. It appears that the fragility selected to represent LOCA sequences are coming from specific components but then they	Documentation on the use of fragility in support to LOCA should be clarified to better represent the rationale selected and potentially addresses the modeling uncertainties associated with this selection. While this finding is expected to be addressed via	LOCA basis has been re-evaluated and updated. This was partially due to seismic fragility update and partially a matter of adding amplifying information to the LOCA basis. See appendix B of the quantification report.

				are used to represents sort of surrogate events for potential failures along the piping network. Using localized events as surrogate for pipe network failure is probably conservative and may not be fully consistent with the system success criteria and modeling in the internal events modeling. For example, the seismic-induced MLOCA fragility seems to be based on failure of the pressurizer surge line, which is a localized failure. The seismic-induced MLOCA initiator is mapped to the internal events MLOCA initiator. The internal events logic for MLOCA has a split fraction that divides MLOCA (and LLOCA) in four 25% contributors impacting all four CL/HL. Since the seismic-induced MLOCA is a localized failure, the internal events logic is not fully applicable (probably slightly conservative). Because the documentation is potentially leading to a misunderstanding of the selected approach (thus impacting ease on update), this F&O is considered a finding against the documentation SR.	documentation, some additional suggestions are provided, such as: 1. Perform a sensitivity to show that the modeling approach described is not significantly skew the results for seismic; 2. Modify the logic by mapping the seismic-induced MLOCA to a different position in the logic (e.g., a dummy event can be entered in the model to provide a target for the FRANX injection).	*[DRAFT; additional sensitivity results to be described]* This issue is resolved. No impact on 50.69 or other risk-informed applications.
16-6	SPR-B2	Not Met	The effect of seismic impact on performance shaping factors is not considered for any action that was explicitly added for the SPRA (e.g., flood isolation or DG output breaker closure).	The Vogtle SPRA elected to use Integrated Performance Shaping Factors (IPSF) multipliers. While this approach was used for the HEPs that were carried over from internal events, it was systematically not done for all the actions explicitly added for seismic.	Expand the IPSF approach to all the operator actions credited in the SPRA.	The methodology used for the seismic HRA analysis is based on defining PSFs as a function of seismic hazard level (bins), which is consistent with the EPRI seismic HRA guidance in EPRI 1025294. The Integrated

			(This F&O originated from SR SPR-B2)	Based on discussion during the peer review, the analyst believed that having designed these actions for specific scenarios following a seismic event, the impact of seismic specific PSF is already included. The objection to this conclusion is that the seismic specific PSF should realistically change with the magnitude of the event. This change addresses the change in the overall context of the plant when a small seismic event happens as opposed to when a very large seismic event happens. This seems not to be captured by the approach selected for the Vogtle SPRA. One example of this is that an action that has a 30 minute Tsw (S-OA-BKR-LOCAL) maintains an HEP of 1.60E-03 at all g levels, including the %G14 interval (i.e., >2g). It is understood that this is not expected to be quantitatively significant because failure of the recovered equipment is taken care by the logic model.		PSFs and bins (breaking points) have been updated to reflect seismic binning applicable to Vogtle, in accordance with this finding and consistent with the EPRI guidance. The updated values have been applied to both internal events HFEs and seismic-unique HFEs within the plant response model. There was no significant impact on the SPRA results. This issue is resolved. No impact on 50.69 or other risk-informed applications.

16-7	SPR-E2	Met	Base case seismic LERF does not meet the truncation requirements from QU-B3. (This F&O originated from SR SPR-E2)	Both CDF and LERF are truncated at 1.0E-09 with 1000 cutsets managed by ACUBE. This meets the QU-B3 requirement for CDF but not for LERF.	LERF at 1E-11 truncation meets the QU-B3 truncation requirement. Rename LERF at 1E-11 as the base case for LERF.	LERF truncation, which was already considered in sensitivity studies, has been revised appropriately to meet QU-B3. A new LERF truncation limit has been established consistent with the LERF results. Quantification is at 1E-12, which is a suitably low value. This finding is resolved. No impact on 50.69 or other risk-informed applications.
16-8	SPR-E2	Met	Missing documentation of cutsets review (cfr. QU-D1) (This F&O originated from SR SPR-E2)	Section 3.1 is the only description of the most important scenarios but there is no cutset-by-cutset review.	While it is understood that the Draft. B version of the quantification notebook is still somewhat a work in process, it is expected that when the model reaches a more stable state documentation of the review of the cutsets is going to be part of the documentation.	The QU report has been updated to document the review of both dominant cutsets and non-significant cutsets for both CDF and LERF. This finding is resolved. No impact on 50.69 or other risk-informed applications.

16-9	SPR-B1, SPR-B4b	Met, Met	Screening values used for the HEPs that (at the time of the provided documentation) were in the most significant cutsets. (This F&O originated from SR SPR-B1)	At the time when the documentation was provided for peer review, the most significant operator actions (i.e., flood isolation of ACCW HX) were all screening values, which would only meet CCI for HR-G1 (directly called through SPR-B1). In addition, there is little documentation or supporting evidence to justify screening values as low as 3.00E-2	An appropriate resolution of this F&O is pending the current evolution of the model and the importance of operator actions in the SPRA. Given the expectation that operator actions will be needed to mitigate the importance of relay chatter (not yet included in the SPRA logic model) this F&O was provided to ensure care is used in the generation of HEPs if they appear in important cutsets and also to provide more justification for screening values less than 1.00E-1 because a low screening value may indeed skew the actual importance of the newly generated HEP.	The seismic HRA analysis has been revised to be consistent with the EPRI seismic HRA guidance in EPRI 1025294. The original screening HEPs have been upgraded using the HRA Calculator, consistent with the approach used in the VEGP internal events PRA. The Documentation has been updated. Operator response to relay chatter has been addressed and evaluated within the same process, and not found to be important. This finding is resolved. No impact on 50.69 or other risk-informed applications.
17-1	SPR-B1	Met	The documentation does not specifically address the applicability of the internal events accident sequences and success criteria to the SPRA model, and does not properly document the accident sequences created specifically for the SPRA model. (This F&O originated from SR SPR-B1)	The modeling approach injected seismic fragilities into fault trees that were modified from the internal events PRA model. It can be inferred from this approach, and it was verified by discussions with the staff, that the internal events sequences and success criteria were considered to be applicable to the SPRA model. This was not specifically stated in the documentation. Further, several additional seismic flooding sequences were added to the fault tree. These sequences are not discussed from an accident sequence and success criteria perspective. Inspection of the fault tree and discussions with	A separate section in the documentation that specifically addresses accident sequences and success criteria is needed to collect the information in one logical place, and is needed to support effective peer reviews and future model updates.	The discussion of accident sequences and success criteria has been expanded, and specific descriptions of the flooding scenarios has been added. This finding is documentation only and does not impact Seismic PRA model results. This issue is resolved. No impact on 50.69 or other risk-informed applications.

				the staff indicate that the sequences were appropriately developed with specific success criteria that is different from other internal events sequences. The development of these sequences needs to be included in the documentation. Including event trees for these sequences would also aid in a reader's understanding.		
17-2	SPR-E2, SPR-F2	Met, Met	The processes used to create the presented quantification results are not fully documented. (This F&O originated from SR SPR-F2)	Examples include: The top cutsets shown in table 3-1 of the quantification report are produced by combining the cutsets from all the seismic interval cutsets in a process that is not documented. While the process used to obtain the importance measures in section 5.2 of the quantification notebook is documented in that section, discussions with the PRA staff indicated that importances for some of the basic events were obtained in a different manner (setting to one or zero and requantifying). This is not documented in the notebook.	Expand the documentation to clearly explain the post-processing of the results generated by CAFTA and FRANX. Examples include: - Explain how the cutsets generated by FRANX are combined into g-level-independent cutsets. - Explain the post-processing used to generate importance measures, especially focusing on the deviation from a normal practice that is currently only mentioned in the notebook.	Documentation for QU results has been improved to describe the processes used to aggregate results over the 14 hazard intervals. The importance calculations have been requantified, and the method for presentation documented. This finding is resolved. No impact on 50.69 or other risk-informed applications.

17-3	SPR-B3, SPR-E4	I/II, I/II	Subdividing correlation groups based on weaker/stronger components resulted in retention of non-minimal cutsets in some cases, which could impact CDF/LERF results as well as model importance measures. The magnitude and acceptability of these impacts was not documented. (This F&O originated from SR SPR-E4)	To account for similar equipment that has different fragilities due to different building locations, certain correlation groups were subdivided to assign a seismic capacity to a weaker component that only failed that component. The higher capacity was then assigned to both components, and was effectively the correlated failure of both components. This can result in the retention of non-minimal cutsets in some cases. For example, for the Containment Fan Cooler Units there are cutsets in which, due to other failures, only one containment fan cooler needs to seismically fail to cause core damage. Inspection of the cutsets shows that two otherwise identical cutsets are retained: one in which the 1Fan 'group' occurs, and one in which the 4Fans group occurs. The 4Fans cutset is not minimal, and should not be included in the results. Discussions with the staff indicated that these non minimal cutsets were noted during the quantification review process, but were thought to not greatly impact overall results. No formal assessment was done, however, and no record of the informal assessment was included in the documentation.	The impact of the retention of these non-minimal cutsets on CDF/LERF and importance measures should be assessed and the results documented, or a method to remove the non-minimal cutsets should be devised. Each subdivided correlation group should be investigated for similar effects.	The non-minimal cutsets were identified and reviewed for impact, and determined to be non-significant to risk. The results were very slightly conservative due to these non-minimal cutsets. Further, in the updated model, such non-minimal cutsets no longer appear. This issue is resolved. No impact on 50.69 or other risk-informed applications.

17-4	SPR-E6	Met	No quantitative analysis of the relative contribution to LERF from Plant Damage States and Significant LERF contributors from Table 2-2.8-9 was presented in the quantification results. (This F&O originated from SR SPR-E6)	A quantitative analysis is required to meet CCII for LE-F1 & LE-G3, which are directly called from SPR-E6.	Perform the analysis and include the results in the quantification notebook.	The quantitative analysis of significant LERF plant damage states and contributors has been performed. A table and associated discussion of plant damage states and significant contributors has been added to the LERF QU documentation to resolve this finding. This finding is resolved. No impact on 50.69 or other risk-informed applications.

Table 3: Disposition of Key Assumptions/Sources of Uncertainty in the SPRA *[PRELIMINARY]*

Assumption / Uncertainty	Discussion	Disposition
Correlation of SSCs	The SPRA assumes complete correlation for identical components in the same building at the same elevation and orientation.	A sensitivity was performed to un-correlate key mitigation components. The results indicate that the model is not sensitive to this treatment. In the absence of a consensus approach to partial correlation, the approach followed represents the consensus approach and is not a key source of uncertainty.

Assumption / Uncertainty	**Discussion**	**Disposition**
The Seismic PRA HFE dependency analysis	The Seismic PRA dependency analysis assumes that once an accident sequence is initiated, the operator action timing for a seismically induced event is similar to that of an internally induced event for main control room actions.	Modification of the time available due to seismic considerations may result in a longer response or identification time and consequently a higher HEP. A sensitivity analysis was performed in the seismic PRA quantification increasing the failure probability of all internal events HEPs to 1.0, resulting in an insignificant increase in CDF. For the 50.69 application, the methodology already includes sensitivities to HEPs, using the 5th and 95th percentile values. This is a sufficient approach given the insensitivity of the VEGP model to internal events HEPs.