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Vogtle Electric Generating Plant – Units 1&2
License Amendment Request to Incorporate Seismic Probabilistic Risk Assessment into the 10 CFR 50.69 Categorization Process
Response to Request for Additional Information (RAIs 4-11)

Ladies and Gentlemen:

By letter dated June 22, 2017 (Agencywide Documents Access and Management System Accession No. ML17173A875) Southern Nuclear Operating Company, Inc. (SNC) submitted a License Amendment Request (LAR) for Vogtle Electric Generating Plant (VEGP), Units 1 and 2 and requested U.S. Nuclear Regulatory Commission (NRC) approval to use the Seismic Probabilistic Risk Assessment model in the existing 10 CFR 50.69 categorization process. By letter dated January 5, 2018, the NRC staff notified SNC that additional information is needed for the staff to complete their review. The Enclosure provides the SNC response to the NRC requests for additional information (RAIs), specifically, questions 4, 5, 6, 7, 8, 9,10, and 11. SNC provided responses to RAIs 1, 2, 3, and 12 by letter dated February 6, 2018. As noted in the NRC letter dated January 5, 2018, portions of these responses may be used for the NRC review of the VEGP Systematic Risk-Informed Assessment of Debris Technical Report.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at 205.992.7369.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on the 21 day of February 2018.

J. J. Hutto
Regulatory Affairs Director

JJH/PDB/CBG

Enclosure: SNC Response to NRC Request for Additional Information

cc: Regional Administrator, Region II
NRR Project Manager – Vogtle 1 & 2
Senior Resident Inspector – Vogtle 1 & 2
State of Georgia Environmental Protection Division
RType: CVC700

Vogtle Electric Generating Plant – Units 1&2
License Amendment Request to Incorporate Seismic Probabilistic Risk Assessment into the 10 CFR 50.69 Categorization Process
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Enclosure

SNC Response to NRC Request for Additional Information

NRC RAI 4

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

SNC Response to RAI 4

Characterization of the seismic hazard applicable to any particular plant site is a complex process that relies largely on availability and interpretation of consensus seismic hazard catalog and ground motion models. Such consensus models and data are updated through NRC and industry initiatives. For example, the NRC may identify an actual safety-significant change in hazard information that should be evaluated such as the "NGA East" program that is currently underway. The conclusion of Fukushima Near Term Task Force (NTTF) Recommendation 2.2 was that if there is no new and significant information, the hazard catalog should be updated every 10 years. As such updates occur, the impact on the VEGP SPRA will be evaluated within the VEGP PRA maintenance process.

However, should site-specific seismic events occur or new site-specific information become available that would have bearing on the SPRA, that information would be considered as part of the PRA maintenance process as well. The PRA maintenance process ensures that the impact of new site-specific hazard information, should it become available, will be appropriately addressed in 50.69 categorization.

NRC RAI 5

Section 3.3.2 of RG 1.200, Revision 2, states "[f]or each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making

associated with the application." Further, Section 4.2 of RG 1.200 states that "[t]hese assessments provide information to the NRC staff in their determination of whether the use of these assumptions and approximations is appropriate for the application, or whether sensitivity studies performed to support the decision are appropriate." RG 1.200, Revision 2, defines the terms "key assumption" and "key source of uncertainty" in Section 3.3.2, "Assessment of Assumptions and Approximations."

Section 3.1.3 of the Enclosure to the June 22, 2017, submittal states that key VEGP SPRA model specific assumptions and sources of uncertainty for this application have been identified and dispositioned. Section 5. 7 of VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, (ADAMS Accession No. ML 17088A130) describes several sensitivity studies that were performed to examine different input information and assumptions on the VEGP SPRA results. Section 3.1.3 of the Enclosure to the June 22, 2017, submittal states that no additional SPRA specific sensitivities have been identified that would be expected to have an important impact on categorization results.

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

b. Please clarify which sensitivities in Section 5. 7 of the VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, address key assumptions and key sources of uncertainty. Describe each key assumption and key source of uncertainty identified in the VEGP SPRA that was not provided in the VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017. The description should contain sufficient detail to identify whether key assumptions used in the SPRA involve any changes to consensus approaches.

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

SNC Response to RAI 5a (GSI-191)

As defined in RG 1.200:

- A *key source of uncertainty* is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA.
- A *key assumption* is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results.
- The term "different results" refers to a change in the risk profile and the associated changes in insights derived from the changes in the risk profile. A "reasonable alternative" assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

Assumptions are made during PRA development to address a modeling uncertainty. For the SPRA, two sets of assumptions, and associated uncertainties can affect the results:

1. Assumptions in the internal events PRA that were identified in the GSI-191 submittal as potential sources of uncertainty that could impact the internal events PRA results, and thus the seismic PRA results.

2. Assumptions made specifically for the SPRA.

These assumptions were reviewed to determine if they are related to a source of modeling uncertainty. If so, then that uncertainty was characterized, and the potential impact on the GSI-191 application was determined.

1. Assumptions identified in the GSI-191 Submittal: From the characterization of potential sources of uncertainty in the baseline Internal Events PRA model (*ADAMS Accession No. ML17116A098*, Encl. 3, Table 3-18), the items in Table 5.a-1 may impact the internal events PRA results, and thus the SPRA results. For each of the sources of uncertainty, the potential impact on the SPRA is addressed to determine the impact on the GSI-191 application.

Table 5.a-1: Impact of Internal Event PRA Uncertainties on the Seismic PRA and GSI-191				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
High RCS pressure impacts the potential for induced steam generator tube rupture (SGTR). Medium and large LOCA and reactor vessel rupture are treated as low RCS pressure scenarios. All other core damage sequences are considered high pressure sequences where the induced SGTR failure mode is possible. Therefore, the baseline PRA model may overestimate the contribution of induced SGTR to LERF.	Scenarios with significant RCP seal leakage, a stuck open pressurizer valve, or a pressurizer PORV open for feed-and-bleed cooling are conservatively considered high RCS pressure scenarios.	Sensitivity cases were performed for the baseline Internal Events PRA by reclassifying the identified scenarios as low RCS pressure to determine impact on LERF.	The GSI-191 risk assessment demonstrates that only large LOCAs could result in debris related failures. Therefore, the possible overestimation of induced SGTR for high pressure scenarios has no impact on the GSI-191 risk assessment.	The SPRA disposition is the same as for the IE PRA disposition. There is no impact on the GSI-191 seismic risk assessment.

Table 5.a-1: Impact of Internal Event PRA Uncertainties on the Seismic PRA and GSI-191				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
Certain initiating events can be affected by seasonal variations (e.g., loss of offsite power (LOSP), loss of service water (SW), etc.) and baseline PRA does not address seasonal variations.	The generic industry frequency for the LOSP event developed in NUREG/CR-6890 is applicable to the VEGP site. The NSCW cooling towers are not required during cold weather months.	None	The GSI-191 phenomena are of concern for initiating events that could generate debris from insulation materials and coatings inside containment, which could then be transported to the containment sump and fail the ECCS sump suction strainers during the recirculation phase needed to maintain core cooling. For VEGP, the initiating events that meet these criteria are LOCAs and Secondary Side Break Inside (SSBI) containment. Therefore, seasonal variations of certain other initiating events have no impact on the GSI-191 risk assessment	Seasonal variations of initiating events do not impact the SPRA since a seismic event is not seasonal. Therefore, there is no impact on the GSI-191 seismic risk assessment.

Table 5.a-1: Impact of Internal Event PRA Uncertainties on the Seismic PRA and GSI-191				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
The method of calculation of human error probabilities (HEPs) for the Human Reliability Analysis (HRA) may introduce uncertainty based on the particular methodology applied.	Detailed evaluations of HEPs are performed for the risk significant, pre- and post-initiator human failure events (HFEs) using industry consensus methods. The Technique for Human Error Rate Prediction (THERP) method is applied for pre-initiator HFEs. The Cause-Based Decision Tree Method(CBDTM) is used for cognitive errors and THERP for execution errors for post initiator HFEs.	The overall modeling uncertainty associated with the general basis for HEPs is addressed by the standard baseline PRA HEP sensitivity cases for the internal events PRA.	Since the VEGP PRA model is based on industry consensus modeling approaches for its HEP calculations, and there are no additional HFEs added for the GSI-191 risk assessment, this is not considered a significant source of epistemic uncertainty and therefore has no impact on the GSI-191 risk assessment.	The SPRA disposition is the same as for the IE PRA disposition. There is no impact on the GSI-191 seismic risk assessment.

Table 5.a-1: Impact of Internal Event PRA Uncertainties on the Seismic PRA and GSI-191				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
The VEGP PRA medium LOCA frequency is based upon data from NUREG/CR-6928, which is an order of magnitude higher than the previous data used from NUREG/CR-5750.	None	A sensitivity was performed for the Internal Events to determine the impact of the increased medium LOCA frequency from NUREG/CR-5750. A more than 10% increase in CDF and nearly 9% increase in LERF occurs due to the updated data.	The LOCA frequency values in NUREG/CR-6928 are in turn based on the LOCA frequency data from NUREG-1829. NUREG-1829 data are used to develop LOCA frequencies for the GSI-191 risk assessment. The GSI-191 risk impact, however, is not sensitive to the initiating event frequency for medium LOCAs. No medium LOCAs result in sump strainer or core failures due to the effects of debris.	Seismic-induced LOCA frequencies are not developed from these NUREGs. The seismic LOCA frequencies are based on plant specific stress data. There is no impact on the SPRA LOCA frequencies from the NUREGs, and therefore no impact on the GSI-191 seismic risk assessment.

Table 5.a-1: Impact of Internal Event PRA Uncertainties on the Seismic PRA and GSI-191				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
Steam generator (SG) tube condition affects the probabilities of induced SGTR. The current VEGP 3-18SG tube condition is pristine.	If SG tube condition degrades, the induced SGTR probability during secondary side break or anticipated transient without scram for pressure- or thermal-induced SGTR in the LERF analysis would increase.	A sensitivity analysis was performed with average vs. pristine SG tube conditions. CDF increased by slightly more than 1%, while LERF nearly tripled.	The GSI-191 risk assessment demonstrates that only large LOCA could result in sump strainer failure. Therefore, the possible under-estimation of induced SGTR for high pressure scenarios has no impact on the GSI-191 risk assessment.	The SPRA disposition is the same as for the IE PRA disposition. There is no impact on the GSI-191 seismic risk assessment.

Table 5.a-1: Impact of Internal Event PRA Uncertainties on the Seismic PRA and GSI-191				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
The presence of water in the reactor cavity at the time of vessel breach would affect the probability of containment failures (early release due to steam explosion and late release due to base mat melt through).	The base internal events VEGP Level 2 PRA assumes a dry reactor cavity condition.	A sensitivity study was performed for a wet reactor cavity. CDF increased by less than 2%, and LERF increased by more than 12%.	The risk increase for large early release due to GSI-191 is nearly three orders of magnitude below the RG 1.174 Region III risk acceptance criteria. A 12% increase in the large early release frequency would still be well below the Region III threshold.	The dominant contributor to seismic LERF is failure of containment isolation valves to close. Accident sequence progression factors such as a wet reactor cavity would not impact seismic LERF. There is no impact on the GSI-191 seismic risk assessment.

2. Assumptions Identified in the SPRA: The approach used to identify and characterize "key" assumptions and "key" sources of uncertainty in the SPRA consisted of the following steps:

- Seismic-specific assumptions or bases made for each of the VEGP Seismic PRA technical elements are noted in the individual SPRA notebooks. These assumptions or bases were collected from each notebook, including those documenting the seismic hazard, structural models, fragilities, and plant response.
- Based on the definitions of key assumptions and uncertainties, the following questions were evaluated for each of the potential assumptions and associated uncertainties:
  - Is there a reasonable alternative to the assumption that has both:
    - Broad acceptance?
    - Basis at least as sound?
  - Is the assumption an approximation made for modeling convenience?
  - Is there no consensus approach for the uncertainty issue?
- If the answers to all of the above questions was "no", then the assumption or uncertainty is not "key". If the answer to any of the three questions was "yes", then the following question was evaluated:
  - Does the assumption or uncertainty impact the risk profile or insights?
- If a review of the results for CDF and LERF demonstrated that the risk profile and insights were not significantly impacted, then the assumption or uncertainty was not "key".
- For the remaining assumptions and uncertainties, a more detailed review was performed.

Three assumptions were identified for a more detailed review, as shown in Table 5.a-2. Two of these could potentially impact the SPRA risk baseline, but there is no "delta" change in risk due to GSI-191. Therefore, these two would have no impact on the GSI-191 risk assessment. The only assumption that could impact GSI-191 concerns the apportionment of potential seismic failures to each of the LOCA categories (i.e., large, medium, and small). The following Table 5.a-2 provides a description of the uncertainties, the sensitivity study performed, and the dispositions. The result is that the potential increase in seismic CDF and LERF is small in absolute terms, and the total risk increase for GSI-191 would remain one order of magnitude below the RG 1.174 Region III risk acceptance criteria.

One additional uncertainty was identified in NL-16-2382 (ML17173A875) (50.69 Categorization Process, Enclosure Section 3.1.3, PRA Uncertainty Evaluations). Following actuation of the reactor coolant pump shutdown seals (RCP SDS), there may be some scenarios where cold leg temperatures could exceed the rated temperature in a

timeframe insufficient to credit operator action following a seismic event, leading to an RCP seal LOCA scenario that is not included in the SPRA for the GSI-191 submittal. The GSI-191 risk assessment demonstrates that only large LOCA could result in sump strainer failure, and small LOCAs from RCP seal failures would not impact the sump strainer. Therefore, the potential for RCP seal LOCA scenarios has no impact on the GSI-191 risk assessment.

Therefore, the review of the assumptions and uncertainties in the SPRA technical notebooks determined that there are no key assumptions or uncertainties that would impact the risk profile or insights with respect to GSI-191.

Table 5.a-2: Seismic-Specific Uncertainties and Impact on GSI-191

Source of Epistemic Uncertainty	Sensitivity Case	SPRA disposition
In the SPRA, the frequency for seismic-induced LOCAs was split evenly among the Large LOCA, Medium LOCA and Small LOCA. Since the seismic failure location was the Reactor coolant pump, it is possible (although not probable) that the entire seismic-induced LOCA frequency would result in only a Large LOCA. This would increase the Large LOCA frequency by a factor of 3 (and decrease the MLOCA and SLOCA frequency to zero).	A sensitivity was run for the SPRA with the seismic-induced Large LOCA frequency increased by a factor of 3, and MLOCA and SLOCA decreased to zero. Seismic CDF increased (delta SCDF) by 1.0E-08/yr, and Seismic LERF increased (delta SLERF) by 1.1E-08/yr. Based on Table 1-1, VEGP Total Risk Impact due to GSI-191 Failures (*ADAMS Accession No. ML17116A098*), the total CDF increase (delta CDF) would be about 3.75E-08/yr, and total LERF increase (delta LERF) would be about 1.1E-08/yr.	A 3.75E-08/yr increase in CDF due to GSI-191 concerns would still be more than an order of magnitude below the RG 1.174 Region III risk acceptance criteria threshold. A 1.1E-08/yr increase in the large early release frequency would still be about one order of magnitude below the Region III LERF threshold. Therefore, there is no significant impact on the GSI-191 seismic risk assessment.
In a seismic LOSP with failure of the CRDM or RV internals, it was assumed that the control rods would not insert, and an ATWT would occur. In a LOSP (fragility of 0.3g), the control rods would be released immediately, so they may insert before the failure of the higher capacity failures of the CRDM (2.2g) or RV internals (>4g).	A sensitivity study using ATWT occurring only 10% of the time indicates that seismic CDF decreases 6%, with no impact on LERF.	This assumption would only impact the baseline risk and would not impact the GSI-191 risk assessment.

Table 5.a-2: Seismic-Specific Uncertainties and Impact on GSI-191

Source of Epistemic Uncertainty	Sensitivity Case	SPRA disposition
Detailed functional fragility analysis could not be performed for the turbine driven AFW pump, so the detailed anchorage fragility was used for the turbine-driven AFW pump instead of EPRI NP-6041 functional proxy fragility. This fragility was similar to the controlling fragilities for other large pumps, such as the RHR and SI pumps. It should be noted that the EPRI NP-6041 functional proxy fragility is based on the seismic demand at the location of the pumps, but not on the actual seismic capacity of the pumps. The EPRI NP-6041 functional proxy fragility is therefore very conservative for equipment with high seismic capacity, such as pumps. If detailed functional fragility analysis could be performed for these pumps, it is judged that the seismic capacity would be much greater than the EPRI screening fragility.	A sensitivity study examined the impact of a lower seismic capacity for the TD AFW pump, based on the EPRI screening fragility. The result was an increase of 20% for the SCDF, and an insignificant increase in SLERF.	This assumption would only impact the baseline risk, and would not impact the GSI-191 risk assessment.

The SPRA Peer Review identified concerns about the parametric uncertainty analysis and quantification in F&Os 12-23, 12-24, 12-27, and 12-29. These concerns were dispositioned for GSI-191 as follows:

- F&Os 12-23, 12-24, and 12-29 concern inconsistencies between the point estimates in the quantification sections of the SPRA and the uncertainty analysis, and the lack of documentation of the basis (inputs), process, and interpretation for the uncertainty analysis.

The parametric uncertainty analysis has been re-performed for the current SPRA submitted for GSI-191. The inputs and process used to perform this new uncertainty analysis have been documented in detail, and additional interpretation has been added. The parametric uncertainty analysis includes uncertainties in the seismic hazard, the fragilities, the non-seismic random unavailabilities, and the human error probabilities. The quantitative uncertainty results are now consistent with the point estimates in the quantification sections of the SPRA. Thus, these F&Os have been resolved, with no significant impact to the SPRA results or conclusions. The revised assessments provide insight into the overall parametric uncertainty of the SPRA, and do not impact the GSI-191 seismic risk assessment.

- F&O 12-27 concerns the documentation of the quantification of the plant response model, including the uncertainty and importance analyses.

The Peer Reviewers were particularly concerned with the process for combining the cutset results over the 14 seismic acceleration intervals. This impacted the presentation of the quantification results, the input to the parametric uncertainty analysis, and the calculation of component importance values. The quantification report documentation has been updated to describe the quantification process, including the process for combining cutsets over the 14 acceleration intervals, in more detail. As discussed directly above, the parametric uncertainty analysis has been re-performed, and the documentation of the inputs, process, and results has been expanded in the SPRA Quantification Notebook. There was no significant impact on the SPRA results, and the GSI-191 seismic risk assessment is not impacted by the documentation changes.

<u>SNC Response to RAI 5a (50.69)</u>

As defined in RG 1.200:

- A *key source of uncertainty* is one that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF, the set of initiating events and accident sequences that contribute most to CDF and to LERF) such that it influences a decision being made using the PRA.

- A *key assumption* is one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results.
- The term “different results” refers to a change in the risk profile and the associated changes in insights derived from the changes in the risk profile. A “reasonable alternative” assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

Assumptions are made during PRA development to address a modeling uncertainty. For the SPRA, two sets of assumptions, and associated uncertainties can impact the results:

- Assumptions in the internal events PRA that were identified in the 50.69 submittal as potential sources of uncertainty that could impact the internal events PRA results, and thus the seismic PRA results.

- Assumptions made specifically for the SPRA.

These assumptions were reviewed to determine if they are related to a source of modeling uncertainty. If so, then that uncertainty was characterized, and the potential impact on the 50.69 program was determined.

1. Assumptions identified in the 50.69 Submittal: From the characterization of potential sources of uncertainty in the baseline Internal Events PRA model (*NL-12-0932*, Encl. 1, Table 2), the items in Table 5.a-3 may impact the internal events PRA results, and thus the SPRA results. For each of the sources of uncertainty, the potential impact on the SPRA is addressed to determine the impact on the 50.69 application.

Table 5.a-3: Impact of Internal Event PRA Uncertainties on the Seismic PRA and 50.69 Program				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
Pressure-Induced SGTR: High RCS pressure impacts the potential for induced steam generator tube rupture (SGTR). Only the large LOCA, reactor vessel rupture, and the Medium LOCA scenarios are treated as low RCS pressure scenarios. (All other core damage sequences are considered high pressure sequences where the induced SGTR failure mode is possible.) Therefore, the baseline PRA model may overestimate the contribution of induced SGTR (PI-SGTR) to LERF.	Scenarios with significant RCP seal leakage, a stuck open pressurizer valve, or a pressurizer PORV open for feed-and-bleed cooling are conservatively considered high RCS pressure scenarios.	Sensitivity cases were performed by reclassifying several scenarios (RCP seal leak, stuck-open relief valve (SORV), etc.) from high pressure to low RCS pressure to determine impact on LERF. There was no change in LERF.	Since there was no change in LERF, this uncertainty would not have an impact on the 10 CFR 50.69 application, as no change in LERF will also result in no change in components safety significance. Therefore, no explicit sensitivity evaluation is needed for 50.69 impact.	In the SPRA, seismic LERF is dominated by seismic failure of the steam generators, and failure of containment isolation. Pressure-induced SGTR is an insignificant contributor, and is therefore not over-estimated. There is no impact on the 50.69 program from this uncertainty, and no explicit sensitivity evaluation is needed for 50.69 impact.

Table 5.a-3: Impact of Internal Event PRA Uncertainties on the Seismic PRA and 50.69 Program				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
Seasonal Impacts on Initiating Events: Certain initiating events can be affected by seasonal impacts (e.g., LOSP, loss of SW, etc.). Although the LOSP IE includes weather-related events, any seasonal variation is not addressed. The fraction of time the NSCW towers are in bypass is modeled rather than specific season or weather conditions. LOSP and LONSCW frequency may be a potential source of uncertainty. Applications pertaining to or affected by specific LOSP or LONSCW configurations should further evaluate seasonal impacts potential source of uncertainty.	The generic industry frequency for the LOSP event developed in NUREG/CR-6890 is applicable to the VEGP site. The NSCW cooling towers are not required during cold weather months.	N/A	Not a source of uncertainty because the VEGP PRA model reflects average conditions (e.g., overall fraction of time NSCW CTs in bypass). So for an application such as 50.69 which uses the average PRA model, this is not a source of uncertainty. No explicit sensitivity evaluation is needed for 50.69 impact.	Seasonal variations of initiating events do not impact the SPRA since a seismic event is not seasonal. Therefore, there is no impact on the 50.69 seismic risk assessment.

Table 5.a-3: Impact of Internal Event PRA Uncertainties on the Seismic PRA and 50.69 Program				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**IE PRA Disposition**	**SPRA Disposition**
Basis for HEPs: The method of calculation of human error probabilities (HEPs) for the Human Reliability Analysis (HRA) may introduce uncertainty based on the particular methodology applied.	Detailed evaluations of HEPs are performed for the risk significant, pre- and post-initiator human failure events (HFEs) using industry consensus methods. The Technique for Human Error Rate Prediction (THERP) method is applied for pre-initiator HFEs. The Cause-Based Decision Tree Method (CBDTM) is used for cognitive errors and THERP for execution errors for post initiator HFEs.	The overall modeling uncertainty associated with the general basis for HEPs is addressed by the standard HEP sensitivity cases required by NEI 00-04. Since there are no specific HFEs affected by the 50.69 application no additional explicit sensitivity evaluation is needed for 50.69 impact.	Since the VEGP PRA model uses industry consensus modeling approaches for its HEP calculations, this is not considered a significant source of epistemic uncertainty. Therefore, no additional explicit sensitivity evaluation is needed for 50.69 impact. Note that 5th and 95th percentiles sensitivity cases are part of the NEI 00-04 process.	The SPRA disposition is the same as for the IE PRA disposition. The overall modeling uncertainty associated with the general basis for HEPs is addressed by the standard HEP sensitivity cases recommended by NEI 00-04.

2. Assumptions Identified in the SPRA: The approach used to identify and characterize "key" assumptions and "key" sources of uncertainty in the SPRA consisted of the following steps:

- Seismic-specific assumptions or bases made for each of the VEGP Seismic PRA technical elements are noted in the individual SPRA notebooks. These assumptions or bases were collected from each notebook, including those documenting the seismic hazard, structural models, fragilities, and plant response
- Based on the definitions of key assumptions and uncertainties, the following questions were evaluated for each of the potential assumptions and associated uncertainties:
  - Is there a reasonable alternative to the assumption that has both:
    - Broad acceptance?
    - Basis at least as sound?
  - Is the assumption an approximation made for modeling convenience?
  - Is there no consensus approach for the uncertainty issue?
- If the answers to all the above questions was "no", then the assumption or uncertainty is not "key". If the answer to any of the three questions was "yes", then the following question was evaluated:
  - Does the assumption or uncertainty impact the risk profile or insights?
- If a review of the results for CDF and LERF demonstrated that the risk profile and insights were not significantly impacted, then the assumption or uncertainty was not "key".
- For the remaining assumptions and uncertainties, a more detailed review was performed.

Three assumptions were identified for a more detailed review, as shown in Table 5.a-4. In each case, the sensitivity study demonstrated that the categorization in the 10 CFR 50.69 program would not be impacted, and that the assumption/uncertainty was not key. No additional sensitivity studies would be needed for the 50.69 process.

One additional uncertainty was identified in NL-16-2382 (ML17173A875) (50.69 Categorization Process, Enclosure Section 3.1.3, PRA Uncertainty Evaluations). Following actuation of the reactor coolant pump shutdown seals (RCP SDS), there may be some scenarios where cold leg temperatures could exceed the rated temperature in a timeframe insufficient to credit operator action following a seismic event, leading to an RCP seal LOCA scenario that is not included in the NL-16-2382 (ML17173A875) SPRA for the 50.69 submittal. The SPRA has since been updated to include these potential scenarios, so this uncertainty has been resolved for the 50.69 process. It is not considered a key assumption or uncertainty, and further sensitivity analysis is not required for the 50.69 program.

Therefore, the review of the assumptions and uncertainties in the SPRA technical notebooks determined that are no key assumptions or uncertainties that would impact the risk profile or insights with respect to the 50.69 program.

Table 5.a-4: Seismic-Specific Uncertainties and Impact on 10 CFR 50.69 Program		
Source of Epistemic Uncertainty	**Sensitivity Case**	**SPRA disposition**
In the SPRA, the frequency for seismic-induced LOCAs was split evenly among the Large LOCA, Medium LOCA and Small LOCA. Since the seismic failure location was the Reactor coolant pump, it is possible (although not probable) that the entire seismic-induced LOCA frequency would result in only a Large LOCA. This would increase the Large LOCA frequency by a factor of 3 (and decrease the MLOCA and SLOCA frequency to zero).	A qualitative evaluation of this uncertainty was performed. The cutsets for large, medium, and small LOCA were reviewed, and found to be very similar such that an increase in large LOCA frequency would be balanced by a corresponding decrease in medium and small LOCA frequency. Note that seismic-induced LOCAs of any size are not significant contributors to seismic CDF, and thus a shift in frequency to large LOCA from the medium and small LOCAs would not impact CDF. However, in combination with containment isolation failures, seismic LOCAs are significant for seismic LERF. Since their significance for LERF is approximately equal, the change from one LOCA size to another has no significant change to risk.	Since there is no significant impact on seismic CDF or LERF from this uncertainty, there is no significant impact on the 50.69 program risk assessment. This is not a key assumption or uncertainty.

Table 5.a-4: Seismic-Specific Uncertainties and Impact on 10 CFR 50.69 Program		
Source of Epistemic Uncertainty	**Sensitivity Case**	**SPRA disposition**
In a seismic LOSP with failure of the CRDM or RV internals, it was assumed that the control rods would not insert, and an ATWT would occur. In a seismic LOSP (fragility of 0.3g), the control rods would be released immediately, so they may insert before the failure of the higher capacity failures of the CRDM (2.2g) or RV internals (>4g).	A sensitivity study reduced the occurrence of an ATWT by reducing the failure of the control rods to insert by 90% (that is, a factor of 0.1 was used to modify the ATWT frequency and represent control rod failure). The sensitivity results indicated that seismic CDF decreased only 6%, with no impact on LERF. This is a relatively small risk decrease given the large change in control rod success. The small reduction in importance for ATWT mitigation components would be spread among the many remaining components, resulting in a very small increase in importance for any of those components.	The large change in control rod success resulted in a relatively small decrease in CDF, and no change in LERF. This relatively small CDF decrease would result in a small reduction in importance for ATWT mitigation components and operator actions. Based on the relatively small change in importance, the impact on the importance of the many remaining components would not be significant. Therefore, this assumption does not have a significant impact on the 50.69 program risk assessment, and is not a key assumption or uncertainty.

Table 5.a-4: Seismic-Specific Uncertainties and Impact on 10 CFR 50.69 Program		
Source of Epistemic Uncertainty	**Sensitivity Case**	**SPRA disposition**
Detailed functional fragility analysis could not be performed for the turbine driven AFW pump, so the detailed anchorage fragility was used for the turbine-driven AFW pump instead of EPRI NP-6041 proxy functional fragility. This fragility was similar to the controlling fragilities for other large pumps, such as the RHR and SI pumps. It should be noted that the EPRI NP-6041 proxy functional fragility is based on the seismic demand at the location of the pumps, but not on the actual seismic capacity of the pumps. The EPRI NP-6041 proxy functional fragility is therefore very conservative for equipment with high seismic capacity, such as pumps. If detailed functional fragility analysis could be performed for these pumps, it is judged that the seismic capacity would be much greater than the EPRI NP-6041 proxy functional fragility.	A sensitivity study examined the impact of a lower seismic capacity for the TD AFW pump, based on the EPRI NP-6041 proxy functional fragility. The result was an increase of 20% for the SCDF, and an insignificant increase in SLERF. While this would increase the TD AFW pump importance, it would already be categorized as a high safety significance (HSS) component. The impact on other components would be to slightly decrease their importances, which would not result in additional components being categorized as HSS. Therefore, this assumption would not impact the 50.69 results or process.	Based on the sensitivity study, this assumption would not impact the categorization of components as HSS. Therefore, this is not a key assumption or uncertainty. No additional sensitivity studies would be needed for the 50.69 program.

The SPRA Peer Review identified concerns about the parametric uncertainty analysis and quantification in F&Os 12-23, 12-24, 12-27, and 12-29. These concerns were dispositioned for the 50.69 program as follows:

- F&Os 12-23, 12-24, and 12-29 concern inconsistencies between the point estimates in the quantification sections of the SPRA and the uncertainty analysis, and the lack of documentation of the basis (inputs), process, and interpretation for the uncertainty analysis.

The parametric uncertainty analysis has been re-performed for the current SPRA submitted for the 50.69 program. The inputs and process used to perform this new uncertainty analysis have been documented in detail, and additional interpretation has been added. The parametric uncertainty analysis includes uncertainties in the seismic hazard, the fragilities, the non-seismic random unavailabilities, and the human error probabilities. The quantitative uncertainty results are now consistent with the point estimates in the quantification sections of the SPRA. Thus, these F&Os have been resolved, with no significant impact to the SPRA results or conclusions. The revised assessments provide insight into the overall parametric uncertainty of the SPRA, and do not impact the 50.69 program.

- F&O 12-27 concerns the documentation of the quantification of the plant response model, including the uncertainty and importance analyses.

The Peer Reviewers were particularly concerned with the process for combining the cutset results over the 14 seismic acceleration intervals. This impacted the presentation of the quantification results, the input to the parametric uncertainty analysis, and the calculation of component importance values. The quantification report documentation has been updated to describe the quantification process, including the process for combining cutsets over the 14 acceleration intervals, in more detail. As discussed directly above, the parametric uncertainty analysis has been re-performed, and the documentation of the inputs, process, and results has been expanded in the SPRA Quantification Notebook. There was no significant impact on the SPRA results, and the 50.69 program is not impacted by the documentation changes.

SNC Response to RAI 5b (GSI -191)

The following sensitivity analyses were documented in Section 5.7 of the VEGP response to the March 2012, 10 CFR 50.54(l) letter, dated March 27, 2017:

- Model Truncation and Convergence
- Increased Seismic Capacity of all Four 125 VDC 1E Distribution Panels
- Offsite Power Impact with Improved Plant Wilson fragility
- Small-Small LOCA
- Control Rod Insertion
- Human Reliability

- Preventing ESFAS Signal Failure on Loss of Vital AC Inverters
- Auxiliary Building Failure

The sensitivity study on model truncation and convergence was performed to demonstrate that the quantification process was identifying all significant results in terms of total SCDF and SLERF. It does not address a key assumption or uncertainty.

The sensitivity studies on offsite power impact with improved Plant Wilson fragility, and increased seismic capacity of all four 125 VDC 1E distribution panels were performed to examine the potential risk benefits of plant modifications. They do not address key assumptions or uncertainties.

The small-small LOCA sensitivity analysis was performed for the Peer Review since it is one of the items in the ASME/ANS PRA Standard. However, based on detailed walkdowns and the engineering design of potential very small LOCA (VSLOCA) piping and tubing, it was determined that the piping/tubing was very rugged. Therefore, the baseline SPRA does not include a VSLOCA in the plant response model. In addition, a VSLOCA would not impact sump strainer failure. Therefore, this sensitivity analysis does not address a key assumption or uncertainty for GSI-191.

The sensitivity study for the control rod insertion would only impact the baseline risk, not the "delta" change in risk. It would therefore not impact the GSI-191 risk assessment, and is not a key assumption or uncertainty.

The sensitivity analyses for human reliability addresses the overall modeling uncertainty associated with the general basis for human error probabilities (HEPs). Since the VEGP PRA model is based on industry consensus modeling approaches for its HEP calculations, and there are no additional HFEs added for the GSI-191 seismic risk assessment, this is not considered a significant source of epistemic uncertainty and therefore has no impact on the GSI-191 risk assessment.

The sensitivity study for preventing ESFAS signal failure on the loss of the vital AC inverters is a model conservatism. It would only reduce the baseline risk, not the "delta" change in risk. It would therefore not impact the GSI-191 risk assessment, and is not a key assumption or uncertainty.

The sensitivity analysis on Auxiliary Building failure examines a model simplification. It would only impact the baseline risk, not the "delta" change in risk. It would therefore not impact the GSI-191 risk assessment, and is not a key assumption or uncertainty.

The response to RAI 5.a provides a discussion of other potential key assumptions and uncertainties. The conclusion was that there are no key assumptions or uncertainties that would impact the GSI-191 seismic risk assessment.

SNC Response to RAI 5b (50.69)

The following sensitivity analyses were documented in Section 5.7 of the VEGP response to the March 2012, 10 CFR 50.54(l) letter, dated March 27, 2017:

- Model Truncation and Convergence
- Offsite Power Impact with Improved Plant Wilson fragility
- Increased Seismic Capacity of all Four 125 VDC 1E Distribution Panels
- Small-Small LOCA
- Control Rod Insertion
- Human Reliability
- Preventing ESFAS Signal Failure on Loss of Vital AC Inverters
- Auxiliary Building Failure

The sensitivity study on model truncation and convergence was performed to demonstrate that the quantification process was identifying all significant results in terms of total SCDF and SLERF. It does not address a key assumption or uncertainty.

The sensitivity studies on offsite power impact with improved Plant Wilson fragility, and increased seismic capacity of all four 125 VDC 1E distribution panels were performed to examine the potential risk benefits of plant modifications. They do not address key assumptions or uncertainties.

The small-small LOCA sensitivity analysis was performed for the Peer Review since it is one of the items in the ASME/ANS PRA Standard. However, based on detailed walkdowns and the engineering design of potential very small LOCA (VSLOCA) piping and tubing, it was determined that the piping/tubing was very rugged. The response to RAI 7 provides more discussion of VSLOCA, and the reasons that it is not considered to be a key assumption or uncertainty. Therefore, the baseline SPRA does not include a VSLOCA in the plant response model. Therefore, this sensitivity analysis does not address a key assumption or uncertainty for the 50.69 program.

As discussed in the response to RAI 5.a, the sensitivity study for the control rod insertion would not significantly impact the component importance. It would therefore not impact the 50.69 program, and is not a key assumption or uncertainty.

The sensitivity analyses for human reliability address the overall modeling uncertainty associated with the general basis for human error probabilities (HEPs). NEI 00-04 recommends that standard 5^{th} and 95^{th} percentile sensitivity studies be performed for the 50.69 program. Since the VEGP PRA model is based on industry consensus modeling approaches for its HEP calculations, and there are no additional HFEs added for the 50.69 seismic risk assessment, the recommended NEI 00-04 HEP sensitivity studies will be performed for the 50.69 program.

The sensitivity study for preventing ESFAS signal failure given the loss of the vital AC inverters was performed to examine the risk benefits of plant and model modifications. The current model fails containment isolation if the inverters fail since an automatic ESFAS signal would not be sent to certain air-operated containment isolation valves. In the baseline SPRA, it was assumed that operator action to isolate the valves would not be possible for large and medium LOCA scenarios. Other scenarios also have containment isolation failure unrelated to the inverters. If the model was refined, the importance of the vital AC inverters would be reduced. The impact on the other SSCs would be to slightly increase the primary LERF contributors (i.e., steam generators and structures). Other components have very little LERF importance, and would not be impacted in a significant manner. Therefore, this sensitivity study would not significantly impact the importance of other components. It would therefore not impact the 50.69 program, and is not a key assumption or uncertainty.

The sensitivity analysis on Auxiliary Building failure examines a model simplification. The current SPRA does not model failure of an internal wall in the Aux Building, so the sensitivity analysis assumed that the wall failure would be a direct to core damage sequence. Even with this very conservative assumption, the impact on CDF was not significant. This demonstrates that the current model is valid. The model simplification would therefore not impact the 50.69 program, and is not a key assumption or uncertainty.

The response to RAI 5.a above provides a discussion of other potential key assumptions and uncertainties. The conclusion was that there are no key assumptions or uncertainties that would impact the 50.69 program.

SNC Response to RAI 5c (GSI -191)

Each potentially important assumption and potentially important uncertainty has been identified and dispositioned in the responses to RAI 5.a and RAI 5.b above. When evaluated with respect to the definitions of key assumption and key uncertainty given in RG 1.200, none of the potential assumptions or uncertainties were determined to be key for this application. Sensitivity studies to support this conclusion have been discussed above.

SNC Response to RAI 5c (50.69)

Each potentially important assumption and potentially important uncertainty has been identified and dispositioned in the responses to RAI 5.a and RAI 5.b above. When evaluated with respect to the definitions of key assumption and key uncertainty given in RG 1.200, none of the potential assumptions or uncertainties were determined to be key for this application. Sensitivity studies to support this conclusion have been discussed above.

Therefore, the sensitivity studies that will be performed for the 50.69 categorization process are those identified in Table 5-4 of NEI 00-04. No other sensitivity studies were identified in the characterization of PRA adequacy.

NRC RAI 6

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

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

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

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

   i. Please describe the "exceptions to general correlation rule" and provide the technical basis for implementing those exceptions.

   ii. Please clarify whether the sensitivity study for correlated fragilities will be performed for those exceptions to the general correlation rule. Provide a justification if the correlated fragilities sensitivity study will not be performed for those exceptions.

b. Table 5-4 of NEI 00-04 identifies, among other SPRA sensitivity studies, any applicable sensitivity studies identified in the characterization of PRA adequacy. Please clarify whether the sensitivity analyses identified in all other requests for information (i.e., sensitivity analyses discussed in questions 3, 5.c, 7.c.iv, 7.d, etc.) will be performed every time SSCs are categorized under 10 CFR 50.69.

c. The key assumptions and sources of uncertainties identified as part of the licensee's submittal may change as SPRA model updates could affect the significance of those assumptions for this application or create new key assumptions or sources of uncertainties.

   Please describe how the VEGP 10 CFR 50.69 program continues to evaluate assumptions and sources of uncertainty when the VEGP SPRA model is updated in the future and subsequently incorporates key assumptions and key sources of

uncertainty in sensitivity analysis that is performed consistent with the guidance in NEI 00-04.

SNC Response to RAI 6a

As described in Section 4.4.2 of the 50.54(f) submittal (NL-17-0218, 10 CFR 50.54(f) NTTF 2.1 Seismic PRA Submittal), correlation of components was considered per the ASME/ANS PRA Standard. For the VEGP SPRA, if the equipment was similar in design, with similar anchorage, and located in the same building on the same elevation, then the equipment was assumed to be fully-correlated. The only exception to this standard correlation rule was that a more detailed and realistic correlation model was used for some components, as explained below.

Because detailed finite element models of the structures were developed, the seismic demand at different nodes of the buildings could be determined. Since the seismic fragility of a component is a function of the component seismic capacity and seismic demand at the component location, similar components at different locations (although at the same elevation) could have different demands, and thus different fragilities. If the difference between fragilities was small, then the components were correlated using the lower fragility value. However, if there was a significant difference in fragilities (more than about 20% difference), then the higher capacity was used to assign a higher correlated fragility to both components, but the lower capacity component was also assigned a unique seismic capacity that only failed that component. Thus, the lower capacity component could fail by itself, but was guaranteed to fail if the higher capacity component was failed.

Thus, correlation was considered for all components in accordance with the ASME/ANS PRA standard. When appropriate, the components were correlated, with some components using the more detailed correlation model. The "exception" is that a more detailed correlation is used for some components. Thus, the SPRA model is fully correlated.

Table 5-4 of NEI 00-04 recommends that that one of the sensitivities to be performed for SPRA use in the categorization process is to use correlated fragilities for all structures, systems, and components (SSCs) in an area. This means that if full correlation is not used for all appropriate components, then a sensitivity study should be performed with full correlation implemented. Since the VEGP SPRA already has full correlation implemented, this correlation sensitivity study does not need to be re-performed for the 50.69 process. As discussed above, the "exceptions" were still correlated, but used the more detailed and realistic correlation model. There are no "exceptions" that are not already fully correlated at the appropriate fragility, and an additional sensitivity study on correlation is not required

SNC Response to RAI 6b

As noted in the responses to the other questions, no additional sensitivities have been identified in the characterization of PRA adequacy that are required to be performed for each categorization. None of the discussed sensitivities are identified as a key source of uncertainty requiring specific sensitivities for categorization per Table 5-4 of NEI 00-04, and these will not be performed as part of the 50.69 categorization process. Note that the RCP seal LOCA model is included in the SPRA model to be used for 50.69 categorization and it addresses the cold leg temperature issue discussed in RAIs 3 and 5.

SNC Response to RAI 6c

Evaluation of PRA model assumptions and sources of uncertainty is part of the SNC PRA model maintenance process. When a PRA model update is performed, the model changes are evaluated to determine whether these involve new assumptions or sources of uncertainty that might be key to particular applications. If a change to the PRA model results in new assumptions or sources of uncertainty, these will be evaluated for impact on the 50.69 categorization process. Consistent with NEI 00-04 and the VEGP 50.69 program, if a new assumption or source of uncertainty is identified that should be considered by the IDP in its consideration of categorization results, it would be added to the list of required sensitivities for the SPRA.

NRC RAI 7a

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

a. F&O 14-10, related to Supporting Requirement (SR) SFR-A2, assigns a CCI to that SR and states that significant conservatisms were noted in several sampled fragility calculations. The "Finding Basis" cites the Component Cooling Water (CCW), Auxiliary Component Cooling Water (ACCW), a particular battery rack, and the turbine driven auxiliary feedwater pump as examples. The resolution to the F&O mentions updates only to the examples cited in the F&O. Using conservative fragilities can lead to incorrect categorization and therefore, treatment of SSCs based on the SPRA results.

    i. Please discuss how SNC ensured that instances of significant conservatism in the fragilities of SSCs, other than the examples cited by the peer-review team, were identified and adequately addressed in the VEGP SPRA in the context of the June 22, 2017, application.

    ii. Please describe how the fragility calculations were revised to address the issues identified in F&O 14-10. Cite any applicable consensus approach used, including

deviations from such approach, and describe the impact of the deviations on this application.

SNC Response to RAI 7a

i. F&O 14-10 identifies two conservatisms in the component fragility evaluation. The conservatisms are with nozzle loads for mechanical equipment and the frequency range of interest used in the fragility evaluation. To address this F&O, SNC used a systematic approach to account for realistic nozzle loads, when applicable, and frequency range of interest in the fragility of SSCs. The approach involved initially applying realistic frequency range of interest for risk significant components. After this refinement if a component is still risk significant component then realistic nozzle loads are applied, wherever nozzle loads are applicable.

ii. As stated in the response to 7.a.i, the revision of the fragility evaluation was a two-tiered approach. First the realistic frequency range of interest is used while computing the demand for a risk significant component. The frequencies are based on seismic qualification packages and EPRI TR-102180. A +/- 15% is applied to account for uncertainty in the frequency of the component and this is consistent with the guidance in EPRI NP-6041. If the component is a significant risk contributor to the SPRA after refining the frequency range of interest then the nozzle loads are refined for applicable mechanical components such as pumps, heat exchangers and tanks. The nozzle loads are refined based on inlet and outlet operational loads and location of the support for the inlet and outlet pipe, and valves. This is consistent with guidance in EPRI NP-6041.

NRC RAI 7b

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

Please describe the approaches used for (1) identifying the seismic-fire interaction sources; (2) performing a systematic screening of the identified sources;
(3) considering human actions during the screening; and (4) inclusion of the unscreened interactions in the seismic PRA. In responding to the items above, please cite any

applicable consensus or state-of-practice approach used, including deviations from such an approach, and describe the impact of the deviations on this application.

SNC Response to RAI 7b

The process for identifying the seismic-fire interaction sources is based on the approach given in NUREG-1407. The systematic process used for identifying fire sources is summarized below:

- Potential fires in the turbine building or yard areas would not impact safety equipment or other equipment on the seismic equipment list (SEL), and did not need to be further evaluated.
- Flammable gases are limited to hydrogen. There are hydrogen cylinders outside the MSIV area, and hydrogen piping to the volume control tank. During walkdowns, the cylinders outside the MSIV area were examined, and found to be well-restrained and supported. Piping, in general, has high seismic capacity, particularly at a newer plant such as Vogtle with seismic-designed piping. The hydrogen piping to the VCT is judged to have high seismic capacity.
- Transformers inside the SEL buildings are dry-type, with no flammable liquids. Note that many of the transformers are on the SEL, and are reviewed for seismic capacity, including potential fires
- Significant quantities of flammable liquids are limited to lube oil for pumps and the diesel generators, and fuel oil for the diesel generators. These potential fire sources were reviewed when the respective equipment was reviewed for systems interactions.
  - For the DG system, any fire would only fail the associated DG, and not propagate to impact other SEL equipment. Therefore, a fire was not modeled for this equipment.

After the systematic screening of fire sources, there were no unique seismic induced fire scenarios in the seismic PRA models.

NRC RAI 7c

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

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

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

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

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

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

SNC Response to RAI 7c

i. Because the use of a walkdown to document the VSLOCA treatment was a new approach to the Peer Review team, a dedicated discussion session was held to provide more information on the methods, documentation, and results of the walkdowns. This discussion included:

- A verbal description and discussion of the systematic walkdown process specific to the issue of VSLOCA.

- The amount and location of piping and tubing reviewed. Small bore piping and tubing was reviewed in each of the containment quadrants, and at several elevations. This included piping/tubing around the reactor coolant pumps (RCPs) and RCP motors (for seal cooling and leak-off), tubing around the instrumentation and transmitters, the reactor vessel (RV) head vent piping, and piping/tubing associated with the pressurizer.

- The search for issues that would potentially cause piping/tubing failures. This included II/I system interaction issues, the flexibility of tubing to accommodate differential movement between anchorage points, connection points for tubing to instruments and transmitters, and piping/tubing supports for longer runs.

- Photos of the in-vessel instrument lines and supports. Since the walkdown team could not access the in-core instrumentation area, the fragility team used photos to review the potential for tubing failures. The Peer Review team was shown photos of the well-supported and separated in-vessel instrument lines, since these lines had been identified in some previous SPRAs as VSLOCA and SLOCA significant.

- The engineered design for VEGP piping and supports. While older plants have non-engineered "field run" piping that can result in unique piping and support configurations, VEGP is a relatively new plant with well-designed and engineered piping/tubing and supports. This standardization reduces the potential for inadvertent failure conditions from unique piping configurations.

Based on the discussion above, the Peer Reviewers rated this supporting requirement (SPR-B8) as meeting Capability Category III, and commented in F&O 16-18 that "this would be a best practice but it also behooves to the SPRA team to provide detailed documentation of such walkdowns...." Additional documentation of the walkdowns and the systematic evaluation of VSLOCA seismic capacity has been added to the SPRA fragility notebook to resolve this documentation finding.

ii. The general methodology for the systematic evaluation of possible sources of VSLOCAs is outlined in the above response to RAI 7ci. Additional details include:

- Two factors guided the walkdowns to provide extensive coverage of potential VSLOCA piping and tubing:
  - Walkdowns sampled all areas of containment that were accessible with small-bore piping and tubing. This included walkdowns in all quadrants of containment, and multiple elevations.
  - Specific areas of containment were identified using knowledge of the P&IDs of the systems with LOCA-sensitive piping, such as the RCS and CVCS. These areas included the RCP pump and motor areas (including the seal cooling and leak-off piping), the pressurizer area, the RCS loop areas, the RV head vent piping (downstream of the flanged connection to the RV head), and the areas with instruments and transmitters.

- Specific attention was paid to the design and supports for tubing:

- o The tubing connections to the piping or instrument/transmitter were viewed to identify any issue.
- o The flexibility of the tubing was reviewed to ensure that there were bends that would accommodate differential movement between tubing anchor points. This included tubing connected to hanging piping that could move, and connected to a fixed support such as a wall.
- o Tubing runs were reviewed to ensure that there was adequate support for longer runs.
- o Areas above tubing were viewed to ensure that there were no II/I seismic interaction issues that could fail the tubing.

Photos were taken of tubing in many areas of containment to document the walkdown process. Thus, the walkdown was systematic and extensive such that there was very high confidence that the small-bore piping and tubing in containment had very high seismic capacity, and its failure could be screened out of further evaluation.

iii. After the above quote from Note 9 for SPR-B10 in the 2009 ASME/ANS PRA Standard, it further states:
"This requirement is intended to ensure that adding such a small-small-LOCA basic event to each relevant accident sequence is *considered* and is done unless a justification for omitting such can be supported."

The justification would be a peer reviewed methodology (RAI response 7.c.ii) and the peer review team agreed that explicit modeling is not needed.

In the 2013 update to the ASME/ANS Standard (ASME/ANS RA-Sb-2013), this SR (now labeled SPR-B8) has been modified to explicitly include the use of a walkdown to examine the possible sources of a VSLOCA:

"ASSUME the existence of an earthquake-caused "very small loss-of-coolant accident" in the seismic-PRA accident sequences and system modeling, unless it is demonstrated that such a LOCA can be excluded, based on a walkdown or on another examination of the possible sources of such a LOCA."

In addition to the extensive walkdown, SNC also had detailed fragility evaluations of LOCA sensitive piping inside containment for the following systems: RCS, CVCS, RHR, SIS, RV head vent, and RVLIS. These fragility evaluations showed that the piping had very high seismic capacity (>5g) that was significantly greater than the fragility used for small, medium, and large LOCA in the SPRA.

Therefore, based on the walkdown and LOCA piping fragility evaluations, it was determined that VSLOCA could be justifiably screened out of the SPRA.

iv. A sensitivity analysis related to the VSLOCA approach is not warranted as part of the categorization process to address the technical adequacy of PRA for the 50.69 application. Based on the extensive walkdown approach described in the responses to RAI 7ci, 7cii, and 7ciii above, and the fragility evaluations of LOCA sensitive piping, the potential for VSLOCA can be justifiably screened out of the SPRA logic model.

While a sensitivity study was performed in anticipation that the Peer Review group would want to see the impact, the sensitivity study was only a rough approximation of the impact. It was performed using very conservative assumptions, including the use of SLOCA logic, success criteria, and timing. A more realistic model would have treated VSLOCA as a small leak equivalent to about 2 RCP seal LOCAs, giving a time to core damage greater than 24 hours (with AFW available). If the primary side is depressurized using the steam generators, as instructed in the procedures, then there would be greater than 2 days before core damage, giving more than adequate time to perform recoveries. Based on these conservatisms, the use of more realistic success criteria and timing is judged to lead to the same conclusion that VSLOCA could be screened out of the model.

The screening out of VSLOCA is a more realistic treatment rather than the generic assumption of a VSLOCA. It is not considered a key uncertainty since it is well-supported and demonstrated by the extensive walkdown, and the fragility evaluations of LOCA sensitive piping. Therefore, no sensitivity analysis will be performed as part of the 50.69 process.

NRC RAI 7d

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

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

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

SNC Response to RAI 7d

i. The basis for Vogtle seismic LOCA fragilities is much more realistic than the generic LOCA fragilities available in the older literature. Vogtle Seismic LOCA fragilities are plant-specific and are based on design basis calculations specifically for Vogtle 1&2. The evaluation includes multiple piping system covering a range of piping sizes and systems (e.g. RCS, SIS, RHR, CVCS, Reactor Head Vent, RVLIS), including the pressurizer surge line and pressurizer spray piping. Based on the fragility of LOCA piping and associated NSSS equipment, the RCP support was of the lowest fragility, and therefore selected as the basis for the LOCA fragility. It is not a "surrogate", but the most realistic seismic-induced LOCA failure for this plant.

ii. Since the failure of the RCP supports could cause small, medium, or large LOCAs, depending on the mode of failure, the fragility was used for all three LOCA size ranges. These plant-specific calculations are considered much better models for the SPRA, and are not considered key uncertainties. No sensitivity studies are needed since the generic LOCA fragilities are not appropriate to this plant. The uncertainty with the allocation of LOCA is addressed in RAI 5a, table 5.a-4 and concludes, since there is no significant impact on seismic CDF or LERF from this uncertainty, there is no significant impact on the 50.69 program risk assessment.

NRC RAI 7e

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

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

the SPRA. Identify any deviations from the cited guidance and describe the impact of the deviations on this application.

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

SNC Response to RAI 7e

i. The selection and justification of the "breaking points" for the HRA seismic damage bins is described in detail in the seismic HRA notebook. Per EPRI guidance (EPRI 3002008093, An Approach to Human Reliability Analysis for External Events with a Focus on Seismic), four bins were identified based on expected plant damage at the acceleration levels of the bins. These damage states are intended to account for the overall context resulting from the seismic event beyond the specific failures dictated by the cutset, including impact to local infrastructure and non-safety related systems, level of heightened stress, general increase in level of coordination and workload, and quality of working environment. The damage bins can be summarized as follows:

1. OBE to 0.3g (%G01 to %G02): Over the acceleration interval, damage ranges from insignificant damage to safety-related and non-safety SSCs, to some damage to non-safety-related SSCs. For SPRA purposes, it was assumed there were failures/unavailability of PCS, instrument air, and the Plant Wilson connection. Significant sequences start with LOSP.
2. 0.3g to 0.7g (%G03 to %G06): In addition to Bin 1 damage, some damage to non-critical safety-related equipment could start, but not to critical SSCs or CAT I structures. Non-critical SSC examples are ACCW surge tank, ECW cooling coils, and MCR filter trains. Potential access issues with masonry walls in non-Cat I buildings, and steam/water leaks in non-safety SSCs, but access to SEL components would not be impacted. No impacts on MCR controls, SSC instrumentation, panels, or lighting. Some isolated failures of non-critical equipment which could cause long-term consequences if not corrected, but recovery options are available and highly reliable.
3. 0.7g to 1.0g (%G07 to %G09): In addition to Bin 2 damage, additional damage to safety related SSCs. Some wall failures start in Cat I structures (but not near SSCs), with some interior wall failures at upper end of acceleration bin. Potential failures in MCR instrumentation and annunciators start at lower end of this bin, with increasing likelihood (25%) at upper end. Failures to significant equipment starting at lower end

of this bin, with increasing likelihood (25%) at upper end. Significant equipment includes potential for DG lube oil coolers to impact DGs, damage to important vital AC, some sequencer failure. No cliff edge impact, but increasing potential for significant damage to critical equipment and to instrumentation and control. Note that failure of I&C is directly incorporated into the logic model rather than through the HRA multipliers.

4. >1.0g (%G10 to %G14): Substantial damage to SSCs, and to critical instrumentation and control. No substantial damage to Cat I structures at lower end of acceleration range, but some potential as acceleration increases. Failure impacts may not be recoverable.

A sensitivity analysis was performed to determine the impact on seismic CDF and LERF if the HRA seismic damage state bin "breaking points" were lowered. For the sensitivity study, the breaking points for HRA bins 2, 3, and 4 were shifted down by 0.1g. Table 7e-1 shows the baseline bins and acceleration intervals, while Table 7e-2 shows the modified bins for the sensitivity study.

Table 7e-1. HRA Bin PGA Range before Bin Redefinition

HRA Bin	1	2	3	4
PGA Range (g)	<0.3	0.3 – 0.7	0.7 – 1.0	> 1.0
Seismic Ground Motion Intervals	%G01 - %G02	%G03 - %G06	%G07 - %G09	%G10 - %G14

Table 7e-2. HRA Bin PGA Range after Bin Redefinition

HRA Bin	1	2	3	4
PGA Range (g)	<0.2	0.2 – 0.6	0.6 – 0.9	> 0.9
Seismic Ground Motion Intervals	%G1	%G02 - %G05	%G06 - %G08	%G09 - %G14

The independent and dependent HEPs for the moved seismic ground motion intervals (%G02, %G06, and %G09) were increased in accordance with the multipliers for the new HRA bin assignments. The results are shown in Tables 7e-3 and 7e-4 for seismic CDF and LERF, respectively. The overall CDF and LERF increase is 0.72% and 0.00%. Therefore, it is concluded that the total CDF and LERF are not sensitive to the "breaking points" used to define the HRA bins.

Table 7e-3. Comparison of CDF

Seismic Interval	**ACUBE CDF**		
	Before	**After**	**% Increase**
%G02	1.05E-08	1.06E-08	0.95%
%G06	1.69E-07	1.81E-07	7.10%
%G09	3.60E-07	3.68E-07	2.22%
Total (%G01 - %G14)	2.79E-06	2.81E-06	0.72%

Table 7e-4. Comparison of LERF

Seismic Interval	ACUBE LERF		
	Before	**After**	**% Increase**
%G02	0.00E+00	0.00E+00	N/A
%G06	1.24E-09	1.26E-09	1.61%
%G09	1.75E-08	1.77E-08	1.14%
Total (%G01 - %G14)	3.25E-07	3.25E-07	0.00%

A "critical breaking point" beyond which all HEPs are set to unity was not used for the VEGP SPRA. While some SPRAs have made this assumption, there is no basis for such a conservative assumption. In the EPRI guidance (EPRI 3002008093, An Approach to Human Reliability Analysis for External Events with a Focus on Seismic, Section 6.5.2), it is suggested that a "minimum value of 1E-1" be used for the highest HRA seismic damage state bin based on the uncertainties in HRA for these high accelerations. However, a sensitivity study was performed that set all HEPs in acceleration intervals above 0.8g to unity. The impact on seismic CDF and LERF was insignificant (NL-17-0218, 10 CFR-50.54(f) NTTF 2.1 Seismic PRA Submittal, Section 5.75).

There were two modifications to the EPRI HRA guidance (EPRI 3002008093, An Approach to Human Reliability Analysis for External Events with a Focus on Seismic). First, the multipliers for EPRI are designed to be very conservative for screening purposes. With these screening multipliers, the dominant HEPs are identified and then refined using more realistic assessment. Instead, the VEGP SPRA used the integrated performance shaping factor (PSF) multipliers based on the EPRI sponsored Surry Pilot Plant SPRA (EPRI 1020756, *Surry Seismic Probabilistic Risk Assessment Pilot Plant Review*). The integrated PSF multipliers were developed to be more realistic estimates of operator response to earthquake-related stresses, rather than the very conservative screening multipliers in the EPRI HRA guidance. The sensitivity analysis of HEPs recommended in Table 5-4 of NEI 00-04 will provide information on the impact of increasing (and decreasing) the HEPs for the 50.69 categorization.

The second modification from the EPRI HRA guidance was the use of "time after seismic event" rather than the "time margin." It was judged that both timing factors impact the seismic HEP. The "time margin" is already included in the HEP assessment performed using the EPRI HRA Calculator. The integrated PSF multipliers then increase the HEP base value according to the location of action, the magnitude of the earthquake, and the time after the seismic event. In this way, both timing factors are included in the seismic HEP assessment. As above, the NEI 00-04 sensitivity studies on HEPs will provide the impacts of increasing or decreasing HEPs for the 50.69 process.

ii. The HRA sensitivity studies recommended in NEI 00-04, Table 5-4, that revise the HEPs to the 95th and 5th percentiles, will be performed to support this application. These sensitivity studies cover a large spectrum of potential HEPs, and therefore provide

adequate information to address the HRA assumptions and uncertainties of the SPRA for this application. No other HRA related sensitivity analyses are needed to be performed for the 50.69 process.

NRC RAI 7f

f. F&O 16-11, related to SR SPR-E2, states that the review of the potential for additional dependencies introduced by the SPRA model is missing. The resolution states that the dependency analysis has been performed using the EPRI HRA Calculator. The "Suggested Finding Resolution" states that the licensee plans to transition to a different dependency analysis method (based on HRA calculator). The 2009 ASMEIANS PRA Standard (RA-Sa-2009) defines a PRA upgrade as,

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

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

i. Please clarify whether the use of the EPRI HRA Calculator in the VEGP SPRA has been peer-reviewed. If the use of EPRI HRA Calculator has not been peer reviewed, provide a justification as to why the use of EPRI HRA Calculator is not considered a PRA upgrade requiring a focused-scope peer review. If this change qualifies as an upgrade, provide the results from the focused-scope peer review addressing the associated F&Os and their resolutions.

ii. If the use of EPRI HRA Calculator has been peer-reviewed, describe whether the use of the EPRI HRA Calculator was expanded after the peer-review to perform any HRA calculations (e.g., HRA dependencies) that were performed manually in the peer-reviewed model. If the use of EPRI HRA Calculator has been expanded, please demonstrate that the same methods, steps, and sequence that had been used in the pre-existing, manual HRA calculations were exactly mirrored when adopting the HRA Calculator. Please include a comparison of the implemented methods and base values for the top 25 HEPs that were impacted by the expansion of the use of HRA Calculator (e.g., HRA dependencies), pre- and post-Calculator use. If pre- and post-Calculator uses do not provide identical results, (a) describe the reasons for the difference, (b) provide technical bases for new methods, if the peer-reviewed methods are enhanced or changed and (c) justify why the potential differences in methods

do not constitute a PRA upgrade or, alternatively, provide the results from the focused-scope peer review including the associated F&Os and their resolutions.

SNC Response to RAI 7f

i. Basic human error probabilities (HEPs) for the independent human failure events were estimated using the EPRI HRA calculator tool for both the Vogtle internal events PRA (IE PRA) HRA, and the Seismic PRA (SPRA) HRA. These independent HEPs were peer reviewed for both the IE PRA and SPRA peer reviews. Note that the SPRA HRA started with the HRA for the IE PRA. These HEPs from the IE PRA were adjusted to estimate HEPs for the 14 different seismic acceleration levels. The seismic-unique HEPs added to the SPRA were also adjusted for the acceleration intervals.

   Therefore, for the independent HEPs, the EPRI HRA Calculator was used for both the IE PRA and SPRA, and has been peer reviewed. Consequently, the use of the HRA Calculator is not considered to be a PRA upgrade.

ii. Initially, the Vogtle PRA models used a manual HRA dependency analysis approach to identify a combination of human failure events (HFEs) in one cutset, determine the level of dependency between the HFEs in the cutset, and revise the joint HEPs to account for dependency among the HFEs. The Vogtle dependency process and rules were developed by the team that developed the HRA calculator for EPRI, and later implemented the HRA Calculator Dependency tool using these rules. The following rules were used for Vogtle for the manual dependency analysis.

   1. Identify and sort the cutsets according to the risk achievement worth (RAW) of the HFE combinations in descending order (i.e., set all HEPs to 1.0 and sort by cutset frequency).

   2. Sort the HFEs chronologically within each combination (that is, with respect to the start of the accident sequence, as the operators follow their procedures).

   3. Analyze risk significant HFE combinations, in decreasing order of importance. For Vogtle, since this was a manual process, dependent combinations in the top 100 cutsets were identified as potentially important. Note that these combinations would also appear in lower cutsets with similar SSC failures.

   4. Systematically apply the following rules to determine the level of dependence between HFEs in the same combination (note: bold words are factors affecting the dependency level):

      - HFEs with a **common cognitive element** are completely dependent.

- **Execution HFEs that overlap in time** are completely dependent, unless it can be conclusively shown that the **required man power resources** will be available in the specific scenario (i.e. the required tasks can be **performed in parallel using the available manpower**).

- Apply the following **timing rules** to **successive HFEs** in the **same location**:

    - 0-15 minutes: HD (high dependency)

    - 15 – 30 minutes: MD (medium dependency)

    - 30 - 60 minutes: LD (low dependency)

    - More than 60 minutes: ZD (zero dependency)

- Determine, by inspection, if there are any **intervening successes** between HFEs. If so, the level of dependence is considered ZD.

5. Apply the THERP dependency formulas for calculating the conditional HEPs.

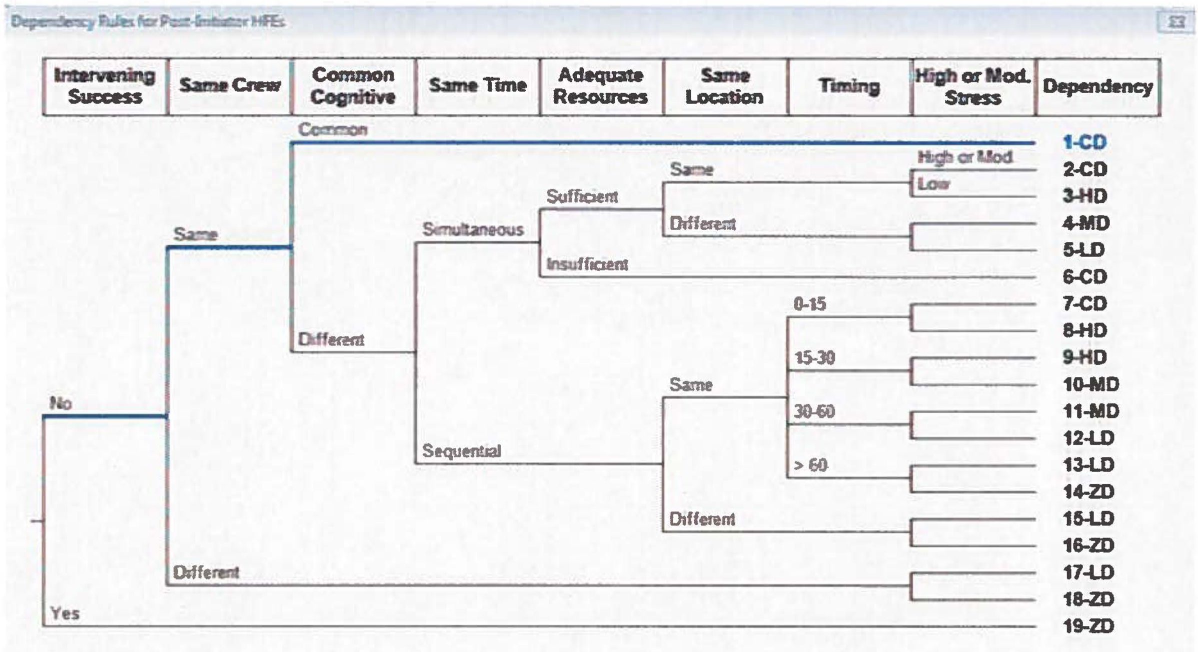
6. Incorporate the dependent HEPs into the PRA model. This was accomplished using one of the following methods:

    - Adjust the probability of the combination of HFE in a cutset using the conditional HEPs from step 5 and a recovery file. This included writing recovery rule equations and running the RECOVER program using the recovery rule equations to adjust the HEPs for the dependent cases.

    - If a dependency was assumed to always occur, then the conditional HEP from step 5 could be directly incorporated into the fault tree. For example, the operator action to fail to start feed and bleed was conservatively assumed to always follow the operator action to start AFW, since that is the scenario when feed and bleed would be used. The conditional dependent HEP for feed and bleed was therefore directly used in the fault tree, along with the independent HEP for starting the AFW. This is equivalent to using the combination HEP for the two dependent HFEs.

The dependency tool that is now implemented in the EPRI HRA Calculator uses a very similar set of rules when compared with the manual process described above. The HRA Calculator just assists the HRA analyst by automating the identification of potential dependencies, and providing the initial quantification of conditional HEPs. Because the process is automated, the number of combinations that are identified can reach into the thousands, since all of the cutsets are used to identify combinations. However, the HRA analyst must still review the dependency analysis results since the HRA Calculator provides potential dependencies that are then modified as needed for scenario realism.

The updated rules being used for determining HRA dependency were developed by the same team who developed the rules for Vogtle. Also, the rules have been reformatted into an event tree format (see the event tree rule below).



In the rules used for the original manual HRA dependency analysis approach, the following factors were considered to impact the level of dependency:

- Common cognitive element
- Overlap in time (execution HFEs)
- Required manpower resources
- Performed in parallel using the available manpower
- Location
- Timing of successive HFEs (in the same location)
- Intervening successes

Comparing the above factors with the factors considered in the event tree above, and the way each or combination of these factors affect the dependency level, these rules and usage of the rules are very similar. One additional factor, stress level, was added in the event tree rule. It should be noted that even though it was not explicitly mentioned in the list of the original rules, HRA analysts often used judgement to increase the level of dependency based on the scenario reviews. Therefore, there are no significant differences between the two sets of rules used for determining dependency level.

Once the dependency level was determined, then the HRA Calculator uses the THERP dependency formulas to calculate the conditional HEPs. This is the same as the THERP formulas used for the initial manual dependency analysis process.

In conclusion, moving from the manual dependency analysis approach to the HRA calculator dependency tool is not considered as an upgrade of the HRA. The use of the HRA calculator dependency module is a tool that automates the process for preliminary identification of dependency, which must then be reviewed by an HRA analyst to validate the results.

In response to the RAI request to include a comparison of the methods and values for the top 25 HEPs that were impacted by the expansion of the use of the HRA Calculator, Table 7f-1 provides a list of the seven dependent failure event combinations that appeared in the detailed cutsets at the model truncation limit for the SPRA (Ver. 2, Seismic Probabilistic Risk Assessment Quantification Report). These seven combinations are the only HEPs that were impacted and are contained in the SPRA cutset results. Table 7f-2 provides the description of the HEP basic event name and independent probabilities. Note that this information is based on the lowest acceleration intervals, and the dependent HEPs increase as acceleration levels increase.

It is important to note that the dependent HEP combinations together have an insignificant Fussell-Vesely (VF) importance measure total of about 0.1% for the seismic CDF, and less than 0.03% for seismic LERF. These dependent HEP combinations are not significant to seismic CDF or LERF.

Table 7f-1 Dependent HEP Combinations Found in Final Cutsets of SPRA

Base Combination ID	SEISMIC HFE1	SEISMIC HFE2	DHEP (Version 1)	DHEP (Version 2)
001	OA-OFC_1-----H	OAB_TR-------H	2.28E-04	2.54E-05
007	OA-START-AFW-H	OAB_TR-------H	1.19E-03	1.12E-03
016	OA-START-AFW-H	S-OA-CU-ISOL-15	No DHEP	3.34E-04
023	OA-START-AFW-H	OAB_SI-------H	7.23E-04	1.10E-03
026	OA-ESFAS-HE1-H	OA-START-AFW-H	No DHEP	4.00E-04
045	OA-ESFAS-HE1-H	OAB_TR-------H	No DHEP	2.03E-04
048	S-OA-BKR-LOCAL	OA-N1EBATCHG-H	No DHEP	2.14E-04

Table 7f-2 Individual HEP Names and Descriptions

Event Name	Description	Indep HEP (Version 1)	Indep HEP (Version 2)
OA-OFC_1-----H	OP. FAIL TO CONTINUE TO OPERATE TDAFWP AFTER BAT DEPL- SBO, w DEP failed	1.9E-03	3.46E-04
OA-START-AFW-H	OPERATOR ACTION TO MANUALLY START AFWPUMPS IN MCR FAILS	9.9E-03	2.18E-03

Event Name	Description	Indep HEP (Version 1)	Indep HEP (Version 2)
OA-ESFAS-HE1-H	OPERATOR FAILS TO MANUALLY START ESF EQUIPMENT	3.6E-03	2.77E-03
S-OA-BKR-LOCAL	FAILURE OF O.A. TO LOCALLY RECLOSE BKR AFTER SEISMIC EVENT	1.6E-03	2.19E-03
OAB_TR-------H	OPERATOR FAILS TO ESTABLISH FEED AND BLEED - TRANSIENT	1.2E-01(1)	2.46E-02
OA-SI-------H	OPERATOR FAILS TO ESTABLISH FEED AND BLEED COOLING-SI	7.3E-02(1)	5.67E-03
S-OA-CU-ISOL-15	OPERATOR FAILS TO ISOLATE NSCW TO CCU IN <15M	3.0E-02	1.2E-02
OA-N1EBATCHG-H	OPERATOR FAILS TO PUT THE STANDBY NON 1E BATTERY CHARGER TO SERVICE	5.0E-02	5.03E-02

(1) As discussed below, this HEP has the dependency included.

Each of the dependent combinations is discussed with respect to the initial, peer reviewed SPRA (Version 1) and the updated SPRA (Version 2, submitted for this application), which used the HRA Calculator dependency module. Also note that the HRA for the internal events PRA (IE PRA) has been updated recently, and incorporated into the SPRA, Version 2, so there is some variation in the independent HEPs (commonly called the basic HEP (BHEP)) and dependent HEPs (DHEPs) between the two versions of the SPRA. The updated HRA was performed as part of regular PRA model maintenance (to incorporate, for example, improvements in procedures, or removal of conservatisms in order to model the current plant).

Combination 001: OA-OFC_1-----H and OAB_TR-------H

This combination was directly modeled in the fault tree for the internal events PRA (IE PRA, Version 4) and the initial SPRA (Version 1). The independent HEP for OAB_TR-----H (operator fails to feed and bleed) was 5.8E-02, but was included in the seismic fault tree with the conditional dependent HEP (DHEP) of 1.2E-01. The updated Version 2 of the SPRA uses the combination HEP with the recovery file, rather than incorporating the dependency directly in the fault tree. The updated BHEP is 2.46E-02. The reduction in the DHEPs between Version 1 and 2 of the SPRA is primarily driven by the reduction in the independent HEPs for the two operator actions. These HEP reductions are due to model maintenance with improved procedures and training.

Combination 007: OA-START-AFW-H and OAB_TR-----H

This combination is similar to the combination 001 above, and was also included directly in the fault tree for the initial SPRA. Version 2 of the SPRA incorporated the dependency using the

recovery file and combination HEP. From Table 7f-1 above, the dependency is essentially equivalent.

Combination 016: OA-START-AFW-H and S-OA-CU-ISOL-15

This combination was not identified as a significant dependency for the initial SPRA using the manual dependent HRA process. A cutset review of the initial SPRA shows only two cutsets with this combination, both in the high acceleration 13th bin. The first cutset with this combination is cutset number 23,972 of a total of 24,804 cutsets in this bin, and has a CDF contribution of about 1.1E-11. The second cutset is even lower. Therefore, this potential dependency was insignificant for the SPRA, Version 1, and would not have been identified using the manual dependency process outlined previously. It is also insignificant for the SPRA, Version 2, with a FV importance of about 0.002% for CDF, and no contribution to LERF.

Combination 023: OA-START-AFW-H and OAB_SI-----H

This combination is similar to combination 007 above, with failure to manually start AFW followed by failure to establish feed and bleed. The difference is that instead of a transient initiator, there is a safety injection signal to mitigate a LOCA. The Version 1 SPRA DHEP is lower than the Version 2 SPRA DHEP by about 34%. This difference is due to the selection of dependency by the HRA analysts for the two SPRA versions. For the initial SPRA, the HRA analyst determined that there was a low dependency between the operator actions based on the actions being initiated by different cues and emergency operating procedures (EOPs), and at different timing. There is about 30 minutes between the time that the operators are directed to start AFW, and the time limit for starting feed and bleed (Chapter 8 Human Reliability Analysis for VEGP PRA Model, Section 8.3.2.2.5). For the Version 2 SPRA, the HRA analyst was more conservative, and judged that there was a high dependency between the operator actions. This more conservative analysis resulted in a higher DHEP for the Version 2 SPRA.

Combination 026: OA-ESFAS-HE1-H and OA-START-AFW-H

This combination shows up in the same cutsets as Combination 023 above, with OA-ESFAS-HE1-H failing the actuation of feed and bleed by not actuating the SI signal. The HRA analysis for Version 1 of the SPRA determined that these cutsets should be removed, since OAB_SI-----H in Combination 023 already includes a step to actuate SI (Chapter 8 Human Reliability Analysis for VEGP PRA Model, Section 8.3.2.2.8). (Since the contribution of the cutsets was insignificant, they were not removed, but there was no dependency assigned.) Therefore, the HRA analysis did not include this dependency for the Version 1 SPRA. The HRA analysis for the Version 2 SPRA did not identify that the cutsets should be removed, so the dependency was conservatively retained for the Version 2 SPRA.

Combination 045: OA-ESFAS-HE1-H and OAB_SI-----H

This combination does not appear in the CDF cutsets, but is an insignificant cutset in the Version 2 SPRA cutsets for LERF, with a FV of 0.001%. Because this potential dependency was insignificant for the SPRA, it would not have been identified using the manual dependency process outlined previously. It was therefore not included as a DHEP for Version 1 of the SPRA.

Combination 048: S-OA-BKR-LOCAL and OA-N1EBATCHG-H

This combination appears in insignificant CDF cutsets in the Version 2 SPRA, with a FV of less than 0.0005%. Because this potential dependency was insignificant for the SPRA, it would not have been identified using the manual dependency process outlined previously. It was therefore not included as a DHEP for Version 1 of the SPRA.

In summary, the method and implementation for both the independent and dependent HRAs performed for the IE PRA and SPRA have been subject to peer review. Moving from the manual dependency analysis to the automated HRA Calculator module is not considered to be an upgrade:

- The underlying dependency analysis methodology/concept used is the same as the methodology reviewed during the peer review.
- It is not a significant change in scope or capability that impacts the significant accident sequences or the significant accident progression sequences (the impact is not significant).

Since the use of the HRA Calculator is not considered to be an upgrade, an additional focused-scope peer review is not required.

SNC Response to RAI 7g

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

The NRC staff recognizes that in the design-analysis, the structural response was computed using specific {often conservative) deterministic response parameters for the structure. Because many properties, such as damping, soil property and the response combination method, etc. are random (often with the wide variability), for a given peak ground acceleration level, the actual structural response may differ substantially from the design analyses calculated response. Please provide the basis for deriving 1.0 as structural response factor and provide logarithmic standard deviation of the randomness

(β_R) and state of knowledge uncertainty (β_U). Cite any applicable consensus approach used, including deviations from such approach, and describe the impact of the deviations on this application.

SNC Response to RAI 7g

VEGP is on a relatively soft soil site and in-structure response spectra is driven by the Soil-Structure Interaction (SSI) effects. For lower hazard levels, the structure and soil damping increases countering the higher accelerations, leading to similar in-structure response spectra amplitude computed at 1E-4 level. Conservatisms in the response analysis identified by the peer review team was removed and a median response was computed by performing a realistic SSI analysis. Since the spectral accelerations are expected to be similar at different hazard levels, structure response factor was 1.0 and agrees with the guidance in EPRI TR-103959. The variables associated with structural response that are not dependent on the frequency of the equipment are: Spectral Shape, Time History Match, Damping, Modeling, Mode Combination, Spatial Variation of Ground Motion. The randomness and uncertainty variability associated with each of these variables are based on the recommended values in EPRI TR-103959. Additionally, the uncertainty due to soil property variation is based on the in-structure response spectra (ISRS) from the results of SSI analysis on three soil cases. This is consistent with the guidance in EPRI TR-103959 and for VEGP the uncertainty is a major contributor to the variabilities, due to the strong SSI effects.

NRC RAI 7h

h. The "Finding Basis" for F&O 14-17, related to SR SFR-02, states that the reactor internals fragility evaluation determined the demand based on an average spectral acceleration over the range of 2 to 3 Hertz (Hz), rather than using the peak acceleration in this range of the in-structure response spectra {ISRS), and did not consider the contribution of higher modes. The licensee indicated that this was done to avoid an overly conservative capacity, but agreed that the contribution of higher modes should be addressed. The licensee stated in their disposition that the reactor internals fragility has been updated in the calculation. The NRC staff notes that using the average spectral acceleration over the range of 2 to 3 Hz may result in fragilities which are lower than the actual fragility.

   Please justify that using the average spectral acceleration will not generate non-conservative fragility analysis and describe, in detail, how the contribution of higher modes was revised. Please provide the basis for the fundamental frequency of 2-3 Hz for the reactor internals mentioned in F&O 14-17 and discuss the impact of those components on this application. Cite any applicable consensus approach used to address this finding, including deviations from such approach, and describe the impact of the deviations on this application.

SNC Response to RAI 7h

Since the fragility of the reactor internals was computed using the separation of variables approach, median-centered values are used for parameters that contribute to the capacity and demand. The average acceleration is used between the specified frequency range to compute a realistic median-centered demand. The uncertainty associated with the acceleration within a frequency range is captured through the uncertainty variability using maximum acceleration in the range. Since median capacity as well as variabilities are used in computing the failure probability, the uncertainty associated with the acceleration is captured and ensures that the fragility estimate is not non-conservative.

Per the guidance in EPRI TR-103959, a multi-mode factor is applied in order to capture the contribution from higher modes. The frequency range of interest is based on the seismic reliability proving tests for the reactors internals, CRDM, RCC and fuel assemblies by Nuclear Power Engineering Test Center (NUPEC). The main issue with the reactor internals is the fuel assembly where the fuel rods and control rod guide tubes in the fuel assemblies are separated by spacers that are subject to impact loads due to gap closure from the earthquake. The 17 X 17 full-scale fuel assemblies were tested and the fundamental frequency for the fuel assemblies was between 2 to 3 Hz.

NRC RAI 7i

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

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

SNC Response to RAI 7i

The polar crane has now been screened out of the SPRA model due to the geometry of the crane and crane rails, and the tight fit within the containment itself. It would be highly improbable that the crane could move such that it would fall during an earthquake. Since the crane is not loaded during power operation, the stresses are low, and a collapse would be highly unlikely that

the crane will fall or collapse during an earthquake. To address the F&O 14-7, the crane's capacity due to vertical loading was updated to include a multi-mode factor consistent with the guidance in EPRI TR-103959 to capture the contribution of higher modes and the uncertainty associated with this factor. But since the polar crane was screened out, the fragility evaluation was not required for and modeled in the SPRA. Therefore, a sensitivity study will not be performed as part of the categorization process for the polar crane.

NRC RAI 7j

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

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

SNC Response to RAI 7j

The generic seismic ruggedness spectra (GERS), EPRI NP-7147, contains seismic capacities of critical relays needed for safe shutdown of nuclear power plants. Electrical relay GERS are less generic and depend on details that vary with vintage and model number compared with GERS for equipment classes. The fragility analysis for relays follows the same methodology, presented in EPRI TR-103959 for equipment evaluation based on testing.

However, the analyst does not have an in-cabinet seismic spectral response near the location of the critical relays. For these cases, the effective peak ISRS spectrum for relays is used. This is equal to the amplification factor, AFC, times the highest value of the clipped ISRS which is input at the base of the cabinet where the relays attached. Cabinet Amplification Factors (AFC) presented in EPRI TR-103959, express the most amplification generated at the worst location for a relay located in the specified cabinet types (located in the middle of a door at the top of the cabinet). Assuming a shear beam deflected shape, the in-cabinet amplification should range linearly from 1.0 at the base to maximum value at the top.

For relays with revised fragility, location of the relay within the cabinet is found and in-cabinet amplification is computed. Vertical amplification is set equal to the calculated horizontal amplification, since amplification in the vertical direction would never be greater than amplification in the horizontal direction, for the items requiring amplification consideration. Panels needing evaluation were judged to be more rigid in the vertical direction, thus producing less amplification. This amplification is then used in computing realistic fragility for the relays. This is consistent with the methodology in EPRI TR-103959. No additional sensitivity analysis

will be performed, since there is no deviation from standard industry approach.

NRC RAI 7k

k. The "Finding Basis" for F&O 14-9, related to SA SFA-02, states that it was determined that, among other issues, valve operator heights/weights that are outside of EPAI guidelines would require further effort for resolution. The licensee's disposition states that valve operator heights and weights that were outside EPAI guidelines have been taken into account in the fragility analysis for these components.

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

SNC Response to RAI 7k

For valves that exceeded the experience guidelines, a Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment) 3g load is applied at the center of gravity of the operator in the yoke's weakest direction. If the yoke stresses are low and the relative deflections are small (to ensure that shaft binding will not occur) then the caveat is satisfied. When the caveats are satisfied, EPRI NP-6041 caveats are satisfied as well, since the EPRI NP-6041 caveats are based on the caveats in GIP.

A verification of the valve's material to not be cast iron and flexibility in the cables attached to the valve, and junction box anchorage are ensured to be adequate. No additional sensitivity analysis will be performed, since there is no deviation from standard industry approach.

NRC RAI 8

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

Section 8 of NEI 00-04 provides guidance on how to conduct risk sensitivity studies during the categorization process for all the preliminary low-safety-significant (LSS) SSCs to confirm that the categorization process results in acceptably small increases to CDF and LERF. An example is provided in the guidance to increase the unreliability of all preliminary

LSS SSCs by a factor of 3 to 5, which appears to address random failures. No explicit discussion of seismic risk sensitivity studies is provided in the guidance.

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

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

Based on the preceding discussion,

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

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

SNC Response to RAI 8a

The risk sensitivity study prescribed in Section 8 of NEI 00-04 is a check to ensure that as additional systems are categorized, there can be continued confidence that the overall change in plant risk (delta-CDF and delta-LERF) is within the guidance provided in RG 1.174. In this sensitivity study, the values for unreliability/unavailability for all SSCs that have been categorized as LSS are increased by a factor of 3 (in the VEGP 50.69 program). The factor used is not intended to be mechanistic, but instead is a test to determine that there continues to be adequate margin to the RG 1.174 delta-CDF and delta-LERF guidance assuming that alternate treatments for LSS SSCs might result in a decrease in reliability. With the inclusion of the SPRA in the VEGP categorization process, this sensitivity will be expanded to also use the SPRA to evaluate the impact on CDF/LERF of the factor of 3 change in unreliability and unavailability for LSS components. No other change in this sensitivity study is required, because the factor of 3 addresses any failure mechanism of the modeled components.

SNC Response to RAI 8b

The validity of modeling inputs in the SPRA will be maintained by reviewing changes to the plant, operational experiences, applicable plant and industry operational experience and periodic update of the seismic PRA model. In addition, periodic adjustment to the SSC categorization and

treatment process may be needed if the results of the performance monitoring of SSC indicates so.

NRC RAI 9

The categorization of SSCs, including that using the SPRA, is expected to be based on importance measures and corresponding numerical criteria as described in Sections 5.1 and 5.3 of NEI 00-04. Further, Section 5.6 of NEI 00-04 discusses the "integral assessment" wherein the hazard specific importance measures are weighted by the hazards contribution to the plant risk. Based on the information provided by the licensee in Tables 3.1-2, 5.4-3, and 5.5-3 of the licensee's response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017, it appears that the modeling and subsequent quantification of the licensee's SPRA is based on 'binning' the seismic hazard and its consequences.

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

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

SNC Response to RAI 9a

The VEGP SPRA approach to determining the importance measures is to calculate the FV and RAW measures for a component for each seismic acceleration interval, and then develop overall seismic importance values (for FV and RAW) using the following weighted process to combine the importance values over all seismic acceleration intervals.

- For a component/basic event, the FV and RAW are calculated by ACUBE 2.0 for each of the 14 seismic acceleration intervals, resulting in 14 FV and RAW importance values by interval.
- The interval FV values are weighted based on the seismic acceleration interval CDF divided by the total seismic CDF, and summed together for each seismically failed fragility group to obtain the total FV from the seismic failure. This is essentially using the integrated FV formula given in Section 5.6 of NEI 00-04. (Note that the seismic

LOSP is removed from the importance analysis since it is virtually assured for all seismic sequences and cutsets, and does not correspond to an explicit component.)

- The RAW values are weighted and summed similarly to the FV importance values, using the integrated RAW formula given in Section 5.6 of NEI 00-04.
- The FV of the seismic failure is then combined with the FV of the random failures for that component to get a complete picture of the SPRA FV importance measure for that component.
- The maximum of the RAW for seismically induced failure and RAWs of random failures for that component is used to get a complete picture of the SPRA RAW importance measure.

A similar process and weighting is used for LERF importance measures. Thus, the formulae in NEI 00-04 for performing an integral assessment, while not specifically identified for calculation of the SPRA importance values, can be used for the SPRA.

The weighted SPRA importance measures developed above are compared to the FV and RAW criteria in NEI 00-04. If they meet the criteria for HSS, then an integral assessment is performed, as specified in NEI 00-04. If the integral assessment results in LSS, then this information, along with sensitivity information, is given to the IDP for evaluation. The process for SPRA follows the NEI 00-04 process explicitly.

SNC Response to RAI 9b

As discussed in response to RAI 9a above, the guidance in NEI 00-04 for the integral assessment is explicitly followed for seismic risk. The formulae in NEI 00-04 for integrated FV and RAW will be used to combine the seismic importance measures with the internal events and fire importance measures, and then compared to the NEI 00-04 criteria. There is no deviation from NEI 00-04.

NRC RAI 10

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

The "basis for assessment" for the difference between Addenda A and B for SFR-C6 states that the licensee's SPRA conforms to accepted current practices and that Vogtle accounted for uncertainties in the soil-structure interaction (SSI) analysis by applying strain-compatible soil properties derived from probabilistic evaluation via the probabilistic seismic hazard analysis (PSHA). However, Sections 4.3.2 and 4.3.3 of licensee's submittal in response to the March

2012, 10 CFR 50.54(f) Letter, dated March 27, 2017, does not include an explicit discussion of the connection between the PSHA and the SSI analysis.

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

SNC Response to RAI 10

SSI analyses using three soil profiles (namely the lower bound (LB), best estimate (BE), and upper bound (UB)) are performed to determine the seismic response. The soil profiles used in the SSI analyses are consistent with the ones used for generating the SSI input motion. Section 3.1.1 of VEGP response to the March 2012, 10 CFR 50.54(f) letter, dated March 27, 2017 discusses the development of Uniform Hazard Response Spectrum (UHRS) at specific horizons, along with the corresponding strain-compatible properties for subsequent SSI analysis. The SSI input response spectra (UHRS) are from the site PSHA. The site-specific dynamic soil profile properties (LB, BE, and UB) developed are strain-compatible with the SSI input response spectra.

Vogtle is a relatively soft soil site where all safety-related structures at VEGP are either founded on or embedded in the soil and strong SSI effects are expected. The effects of uncertainty in SSI are dominated by the soil response. Uncertainty is incorporated in the seismic analysis is by evaluating the SSI model for three soil profiles (LB, BE, and UB). The three results were averaged to obtain median responses (ISRS and structural demands).

SSI analyses of safety-related structures at VEGP are categorized as

1. Surface-founded structures analyzed as a surface-founded structure.
2. Embedded structures analyzed as a surface-founded structure at depth.
3. Embedded structures analyzed as an embedded structure.

The input parameters derived for the SSI analysis follows the approach outlined in DC/COL-ISG-017 "Interim Staff Guidance on Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Analyses" (ADAMS Accession No. ML092230543) and the Nuclear Energy Institute (NEI) white paper titled, "Consistent Site-Response/Soil-Interaction Analysis and Evaluation", dated June 12, 2009 (ADAMS Accession No. ML091680715). This ensures that the compatibility between the PSHA and SSI analysis is maintained throughout the seismic response analysis. The depth of soil modeled in the SSI analysis below the foundation elevation is based on the ASCE 4-98 limits.

The requirements of ISG-017 on evaluating the adequacy of time histories for the embedded SSI analysis was examined for conservatisms to disposition a finding-level F&O from the peer review. The original embedded SSI analysis, prior to the peer review, applied a set of scaling

factors to the calculated time histories such that the envelope of the convolved surface outcrop spectra from each soil stiffness case (LB, BE, and UB) envelopes the surface UHRS which agrees to the requirements of ISG-017. VEGP SPRA received a finding-level F&O stating that "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". To disposition this F&O, the calculated time histories are modified such that the average of the convolved surface outcrop spectra from each soil stiffness case (LB, BE, and UB) reasonably matches the surface UHRS. This was the modification that was performed to the time history generation process to address the finding-level F&O.

The SSI analysis of surface founded structures utilizes the SASSI Direct Method (DM) and SSI analysis of the embedded structures are based on the SASSI Modified Subtraction Method (MSM). Confirmatory analyses are performed to validate the accuracy of the MSM analysis results.

NRC RAI 11

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

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

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

SNC Response to RAI 11a

For the Internal Events Assessment described in Section 5.1 of NEI 00-04, it is recommended that the importance measures should be calculated using a truncation level of five orders of magnitude below the baseline CDF or LERF. NEI 00-04 continues with:

"The selected truncation level should support an overall CDF/LERF that has converged, and must be within the capability of the software used."

The guidance also notes that the RAW measure could still be affected, and that these values should be checked.

No specific guidance is provided for other hazards such as fire or seismic. For the SPRA, the only seismic acceleration intervals that are below this criterion are %G12 to %G14. A sensitivity study was performed where the truncation levels were lowered, and the importance measures were recalculated. For CDF, %G12 through %G14 were truncated at 2.8E-11, which is 5 orders of magnitude below the SPRA CDF of 2.8E-06/yr. For LERF, %G12 was truncated at 3.0E-12, and %G13 was truncated at 3.2E-12, which is 5 orders of magnitude below the SPRA LERF of 3.3E-07/yr. Acceleration interval %G14 was truncated at 4.0E-12, and could not be quantified at lower than this truncation.

The baseline quantification results for the top 25 SSCs equipment, ranked by FV and RAW for CDF and LERF, were compared to the lowered truncation results. The FV results for CDF are provided in Table 11a-1. While there are minor differences, they are almost all in the third digit, demonstrating that the lower truncation does not impact the FV measure.

Table 11a-1: Comparison of CDF Truncation on FV Importance Measures

Group	Group Description	FV-Orig. Truncation	FV-Lower Truncation
S_1DCBS-PN-CB180-1E	125 VDC 1E DISTR. PANEL - CB180	21.27%	21.20%
S_CRDM-ATWT	ATWT DUE TO CRDM (CONTROL ROD DRIVE MECHANISM) FAIL TO DROP	15.84%	15.79%
S_SEISMIC_RCP_ALL	SEISMIC FAILURE OF ALL RCPS	6.52%	6.49%
S_SEISMIC-LLOCA	SEISMIC INDUCED LLOCA (RCP)	4.11%	4.10%
S_SEISMIC-MLOCA	SEISMIC INDUCED MLOCA (RCP)	4.10%	4.09%
S_SEISMIC-SLOCA	SEISMIC INDUCED SLOCA (RCP)	3.99%	3.98%
S_1ACBD-MCB	MAIN CONTROL BOARD	3.89%	3.88%
SDS-F	RCP SHUTDOWN SEAL FAILS TO ACTIVATE AND SEAL FOR 24 HRS.	3.75%	3.75%
S_1AFPM-MDP	BOTH AFW MDP (MOTOR DRIVEN PUMP)	2.96%	3.02%
RCPSL-21GPM	RCP SEAL LEAK 21 GPM/PUMP AFTER 13 MIN. TOTAL LOSS OF SEAL COOLING	2.55%	2.54%
S_RX-TRIP-BKRS-SEIS	REACTOR TRIP BREAKERS	2.20%	2.19%
S_1FC-CCU-FLD	ANCHORAGE FAILURE OF MULTIPLE CCU WITH NSCW FLD (FLOOD)	2.16%	2.16%

Group	Group Description	FV-Orig. Truncation	FV-Lower Truncation
S_RV-INT-ATWT	SEISMIC INDUCED FAILURE OF RX VESSEL INTERNALS	2.10%	2.09%
S_1DCBS-SGR-CB180	125 VDC SWITCHGEAR CB180	2.00%	1.98%
S_1ACIV-120-AB220-LC-B	VITAL AC INVERTER 1BD1I12	1.84%	1.84%
S_CB-CHLR-NSCW-FLOOD	SEISMIC FAILURE OF CB ESF CHILLERS CAUSE NSCW FLOOD ON CB 260	1.68%	1.67%
S_1FC-ACU-FLD	ANCHORAGE FAILURE OF ACU WITH NSCW FLD	1.36%	1.37%
OA-OBR-------H_REC	OPERATOR FAILS TO ESTABLISH EMERGENCY BORATION, (HEP replacement)	1.18%	1.18%
S_1DGHE-LUBEOIL	DG LUBE OIL HX	1.17%	1.15%
S_1AFPM-TDP	AFW TDP (TURBINE DRIVEN PUMP)	1.12%	1.11%
S_1DCBS-MCC-ALL	ALL 125 VDC MCC	0.96%	0.95%
S_1DG	DIESEL GENERATOR	0.94%	0.93%
S_DG-BLDG	DIESEL BUILDINGS	0.94%	0.93%
S_1DCBY-CB180	125 VDC BATTERY CB180	0.88%	0.87%

A second comparison was performed for the SSCs that were near the FV criterion of 0.5%, as shown in Table 11a-2. None of the SSCs that were originally below the FV criterion increased above the criterion.

Table 11a-2: Comparison of SSCs near the FV Criterion

Group	Group Description	FV-Orig. Truncation	FV-Lower Truncation
S_1DGFN-FAN	DG BLDG ESF SUPPLY FAN	0.528%	0.528%
1DGDGG4001---X	DG1A FAILS TO RUN BY RANDOM CAUSE (24 HR MISSION TIME)	0.508%	0.508%
S_1DCBC-CB180	BATTERY CHARGER CB180	0.501%	0.494%
S_1DGPN-GEN	DGGEN CTL PNL	0.479%	0.477%
1AFPTP4001---A	TDAFWP (P4-001) FAILS TO START	0.465%	0.474%
S_1AFW-AOV-RLY	RELAY FOR AFW PUMP TURB TRIP & THROTTLE VLV	0.443%	0.446%
S_1CCHE-4	CCW HEAT EXCHANGER	0.419%	0.392%
S_NSCW-TOWER	NUCLEAR SERVICE COOLING WATER TOWER	0.401%	0.398%

Tables 11a-3 and 11a-4 present the same results for the RAW criterion. As with FV, there is no significant difference with the lower truncation. Based on Table 11a-4, none of the SSCs that

were below the RAW criterion of 2 would increase above the criterion with a lower truncation, except for the Control Building. However, as mentioned in the NEI 00-04 methodology, selection of truncation level of five orders of magnitude below the baseline CDF or LERF would have identified Control Building as potentially HSS. When factoring in 10% margin that SNC will use when categorizing SSCs, the results of the sensitivity show that importance measures of the SSCs will be captured appropriately.

Table 11a-3: Comparison of CDF Truncation on RAW Importance Measures

Group	Group Description	RAW-Orig. Truncation	RAW-Lower Truncation
S_1DCBS-PN-CB180-1E	125 VDC 1E DISTR. PANEL - CB180	362.4	361.7
S_SEISMIC_RCP_ALL	SEISMIC FAILURE OF ALL RCPS	93.7	93.5
S_1ACBD-MCB	MAIN CONTROL BOARD	93.7	93.5
S_NSCW-TOWER	NUCLEAR SERVICE COOLING WATER TOWER	44.0	44.0
S_CRDM-ATWT	ATWT DUE TO CRDM FAIL TO DROP	30.2	30.1
S_CB-CHLR-NSCW-FLOOD	SEISMIC FAILURE OF CB ESF (ENGINEERING SAFEGUARDS FUNCTIONS) CHILLERS CAUSE NSCW FLOOD ON CB 260	22.1	22.1
S_SEISMIC-SG	SEISMIC FAILURE OF SG (STEAM GENERATOR)	22.1	22.1
S_CONTAINMENT	CONTAINMENT	22.1	22.1
S_RV-INT-ATWT	SEISMIC INDUCED FAILURE OF RX VESSEL INTERNALS	15.8	15.8
S_RX-TRIP-BKRS-SEIS	REACTOR TRIP BREAKERS	15.4	15.4
S_AUX	AUX BUILDING	11.7	11.8
S_SEISMIC-RVR	SEISMIC FAILURE OF RX VESSEL	6.5	6.6
S_SEISMIC-LLOCA	SEISMIC INDUCED LLOCA (RCP)	4.9	4.9
SDS-F	RCP SHUTDOWN SEAL FAILS TO ACTIVATE AND SEAL FOR 24 HRS.	4.9	4.9
S_SEISMIC-MLOCA	SEISMIC INDUCED MLOCA (RCP)	4.8	4.8
1RCPORV0456A-K	PORV PV0456A RANDOMLY FAILS TO CLOSE	4.6	5.0
1RCPORV0455A-K	PORV PV0455A RANDOMLY FAILS TO CLOSE	4.6	5.0
S_SEISMIC-SLOCA	SEISMIC INDUCED SLOCA (RCP)	4.6	4.6
S_1DCBS-MCC-ALL	ALL 125 VDC MCC	2.2	2.2
1DGDGU1SU2---XCC	DG1A, DG1B, AND A UNIT 2 DG FAIL TO RUN DUE TO CCF (24 HR MISSION TIME)	2.1	2.4
S_CST	CST 1&2	2.0	2.2

Group	Group Description	RAW-Orig. Truncation	RAW-Lower Truncation
1ACCBLSABTRN-FCC	COMMON CAUSE FAILURE OF TRAINS A AND B CB (CIRCUIT BREAKER) TO OPEN DURING LOAD SHED	2.0	2.3
1DGDGG4001002XCC	DG1A AND DG1B FAIL TO RUN BY CCF (24 HR MISSION TIME)	2.0	2.3
1ACCB02050301DCC	RAT A & B SUPPLY CIRCUIT BREAKERS FAIL TO OPEN BY COMMON CAUSE	2.0	2.3
S_1DCBS-SGR-CB180	125 VDC SWITCHGEAR CB180	1.9	1.9

Table 11a-4: Comparison of SSCs near the RAW Criterion

Group	Group Description	RAW-Orig. Truncation	RAW-Lower Truncation
1ACCB02050301DCC	RAT A & B SUPPLY CIRCUIT BREAKERS FAIL TO OPEN BY COMMON CAUSE	2.01	2.31
S_1DCBS-SGR-CB180	125 VDC SWITCHGEAR CB180	1.93	1.93
S_1DG	DIESEL GENERATOR	1.75	1.75
S_DG-BLDG	DIESEL BUILDINGS	1.75	1.75
S_1DCBC-CB180	BATTERY CHARGER CB180	1.72	1.72
S_1DGPN-ENG	DG ENG CTL PNL	1.69	1.69
S_CONTROL-BLDG	CONTROL BUILDING	1.68	2.43
S_1DGHE-LUBEOIL	DG LUBE OIL HX	1.67	1.68
S_1SWFN-NSCW-FANS	NSCW TOWER FANS	1.67	1.68
S_1DGPN-GEN	DGGEN CTL PNL	1.67	1.68

FV and RAW results for LERF are provided in Tables 11a-5 and 11a-6. From these tables, it can be seen that both the FV and RAW values were very similar for SSCs with the original and lowered truncations. For FV, the ACCW surge tanks went from 0.49% to 0.50%. This is within the 10% margin that is used to categorize components, so they would have been selected as meeting the FV criterion with or without lowered truncation. For RAW, several of the SSCs moved from below the RAW criterion of 2 to above the criterion. However, as mentioned in the NEI 00-04 methodology, selection of truncation level of five orders of magnitude below the baseline CDF or LERF would have identified these SSCs as potentially HSS. The only bin that does not meet the criteria of five orders of magnitude below the baseline LERF is bin %G14. The bin %G14 truncates at 4E-12 as it would not quantify below this level using advanced computing machine. The truncation of 4E-12 is only slightly higher than the truncation at five orders magnitude lesser than the baseline LERF of 3.3E-07. When factoring in 10% margin that SNC will use when categorizing SSCs, the results of the sensitivity show that importance measures of the SSCs will be captured appropriately.

Table 11a-5: Comparison of LERF Truncation on FV Importance Measures

Group	Group Description	FV-Orig. Truncation	FV-Lowered Truncation
S_SEISMIC-LLOCA	SEISMIC INDUCED LLOCA	20.48%	20.27%
S_SEISMIC-MLOCA	SEISMIC INDUCED MLOCA	20.48%	20.27%
S_SEISMIC-SLOCA	SEISMIC INDUCED SLOCA	19.62%	19.41%
S_1DCBS-SGR-CB180	125 VDC SWITCHGEAR CB180	19.19%	19.12%
S_1ACIV-120-CB180	AC INVERTER CB180	19.11%	19.18%
S_SEISMIC-SG	SEISMIC FAILURE OF SG	14.35%	14.23%
S_1ACBS-120PN-CB180	120 VAC PANEL CB 180	12.35%	12.40%
S_CONTAINMENT	CONTAINMENT	7.56%	7.48%
S_1DCBC-CB180	BATTERY CHARGER CB180	1.93%	1.92%
S_1DCBY-CB180	125 VDC BATTERY CB180	1.73%	1.73%
S_AUX	AUX BUILDING	1.12%	1.11%
SDS-F	RCP SHUTDOWN SEAL FAILS TO ACTIVATE AND SEAL FOR 24 HRS.	0.88%	1.09%
L2TEAR	CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS, CRACKS, OR TEARS	0.68%	0.77%
S_1XCTK-4	ACCW Surge Tank	0.49%	0.50%
S_CRDM-ATWT	ATWT DUE TO CRDM FAIL TO DROP	0.41%	0.44%
S_1CCHE-4	CCW HEAT EXCHANGER	0.34%	0.31%
S_1DCBS-PN-CB180-1E	125 VDC 1E DISTR. PANEL - CB180	0.21%	0.20%
S_1DGHE-LUBEOIL	DG LUBE OIL HX	0.17%	0.17%
S_1FCMO-CCU-6FANS	CONTAINMENT FAN COOLER UNITS-6,3,4,1,5,8	0.15%	0.15%
S_1ACIV-120-AB220-LC-B	VITAL AC INVERTER 1BD1I12	0.12%	0.13%
S_1DGDM-VENT-1-3	DG VENT DAMPER FOR FANS 1&3	0.11%	0.13%
S_1DG	DIESEL GENERATOR	0.10%	0.10%
S_DG-BLDG	DIESEL BUILDINGS	0.10%	0.10%

Table 11a-6: Comparison of LERF Truncation on RAW Importance Measures

Group	Group Description	RAW-Orig. Truncation	RAW-Lower Truncation
S_SEISMIC-SG	SEISMIC FAILURE OF SG	377.3	377.3
S_CONTAINMENT	CONTAINMENT	377.3	377.3
S_AUX	AUX BUILDING	189.6	189.5

Group	Group Description	RAW-Orig. Truncation	RAW-Lower Truncation
S_CONTROL-BLDG	CONTROL BUILDING	54.9	55.5
S_SEISMIC-LLOCA	SEISMIC INDUCED LLOCA	7.9	7.9
S_SEISMIC-MLOCA	SEISMIC INDUCED MLOCA	7.9	7.9
L2TEAR	CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS, CRACKS, OR TEARS	6.8	8.0
S_SEISMIC-SLOCA	SEISMIC INDUCED SLOCA	6.2	6.2
S_1DCBS-SGR-CB180	125 VDC SWITCHGEAR CB180	2.8	2.8
S_1ACIV-120-CB180	AC INVERTER CB180	2.7	2.7
S_1ACBS-120PN-CB180	120 VAC PANEL CB 180	2.6	2.6
ADMN-PEN-NI	ADMINISTRATIVELY CONTROLLED PENETRATIONS NOT ISOLATED	2.1	3.8
SDS-F	RCP SHUTDOWN SEAL FAILS TO ACTIVATE AND SEAL FOR 24 HRS.	1.6	2.1
S_1DCBS-PN-CB180-1E	125 VDC 1E DISTR. PANEL - CB180	1.5	1.9
1CIAV2626-27BKCC	AOV HV-2626B ANDAOV HV-2627B FAIL TO OPERATE (HARDWARE)	1.5	2.1
1CIAVHV28-29BKCC	AOV HV-2628B AND AOV HV-2629B FAIL TO OPERATE (HARDWARE)	1.5	2.1
1CIAVHV780781KCC	AOV HV-0780 AND AOV HV-0781 FAILTO OPERATE (HARDWARE)	1.5	2.1
S_CRDM-ATWT	ATWT DUE TO CRDM FAIL TO DROP	1.2	1.2
S_1DCBY-CB180	125 VDC BATTERY CB180	1.1	1.1
S_SEISMIC_RCP_ALL	SEISMIC FAILURE OF ALL RCPS	1.1	1.5
S_1DCBC-CB180	BATTERY CHARGER CB180	1.1	1.1
S_1ACBD-MCB	MAIN CONTROL BOARD	1.1	1.4
S_RX-TRIP-BKRS-SEIS	Reactor Trip Breakers	1.1	1.1
S_RV-INT-ATWT	SEISMIC INDUCED FAILURE OF RX VESSEL INTERNALS	1.1	1.1
S_1ACBS-480SGR-CB	480V SWITCHGEAR CB	1.0	1.0

The results of the sensitivity analysis show that lowering the truncation to meet the five orders of magnitude criterion had little impact on the results of the importance measures analysis. SNC plans to use five orders of magnitude lower truncation criterion to the extent allowed by the software/hardware capabilities. In addition, SNC will continue to use 10% margin for FV and RAW thresholds when categorizing SSCs. When using lower truncation and 10% margin to

categorize SSCs, importance measures of the SSCs will be captured appropriately, which, in turn, will be used to categorize SSCs.

SNC Response to RAI 11b

Use of SNC selected screening criteria in conjunction with implementation of guidance provided in NEI 00-04 ensures that the importance measure of the SSCs is captured appropriately. The justification for the VEGP SPRA screening level is:

- Although an SSC may have a fragility above the screening value, these fragilities are usually conservative. Once the SSC fragility is evaluated to be above the screening value, no further refinement of the fragility is performed although conservatisms may still be present. For example, when the VEGP SPRA components were re-evaluated between the initial 2.5g screening and 3g advanced screening, the recalculated median capacity of most components was at least 3g, with many much higher. Thus, the screening is conservative.

- Many of the screened components would not lead directly to core damage or large early release; other failures would also be needed to result in core damage or large early release. While the screening level might be 2 percent, the combination of failures needed to result in core damage or large early release would result in lower percentage contribution.

- When SSCs were judged to be important to CDF and LERF, they were included in the logic model even though their fragility was greater than the screening level. For instance, structures were included, as were components associated with the diesel generators, switchgear, and the instrumentation and control power.

In addition, a review of SSCs with fragilities greater than the screening level would be performed to identify any seismic "singletons." That is, any SSC (or correlated group of SSCs) that could lead directly to core damage or large early release would be identified during the categorization process, to be consistent with NEI 00-04.