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SOUTHERN COMPANY

August 31, 2012

Docket Nos.: 50-424
50-425

NL-12-0932

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request

Ladies and Gentlemen:

Pursuant to 10 CFR 50.69(b)(2) and 10 CFR 50.90, Southern Nuclear Operating Company (SNC) hereby requests amendments to the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Renewed License Numbers NPF-68 and NPF 81. The proposed amendments would revise the VEGP licensing basis to implement 10 CFR 50.69, risk-informed categorization and treatment of structures, system, and components (SSCs) for nuclear power plants. The Nuclear Regulatory Commission (NRC), by letter dated June 17, 2011, in response to SNC's letter dated December 6, 2010, granted pilot status for the VEGP 10 CFR 50.69 license amendment request (LAR).

Implementation of 10 CFR 50.69 allows for application of a risk-informed categorization process per 10 CFR 50.69(c) to modify the scope of SSCs subject to special treatment requirements. Alternative treatments per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. Implementation of 10 CFR 50.69 will allow the licensee and the NRC to better focus attention and resources on SSCs that have safety significance, resulting in improved plant safety.

The VEGP 10 CFR 50.69 LAR conforms to the scope requirements of 10 CFR 50.69(b)(2). The categorization process described in the LAR conforms to the guidance in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006. The categorization process also conforms to the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0 dated July 2005. SNC has determined that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

Enclosure 1 provides the basis for the proposed change to the VEGP Units 1 and 2 Operating Licenses. Enclosure 2 provides the VEGP Operating Licenses marked-up pages showing the proposed change. Enclosure 3 provides

the VEGP Operating Licenses clean typed pages showing the proposed change. SNC requests approval of the proposed license amendments by August 31, 2013.

This letter contains no NRC commitments. If you have any questions, please contact Ken McElroy at (205) 992-7369.

Mr. M. J. Ajluni states he is the Nuclear Licensing Director of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and, to the best of his knowledge and belief, the facts set forth in this letter are true.

Sworn to and subscribed before me this 31st *day of* August, *2012.*

Laura L. Crapo
Notary Public

My commission expires: 11-02-2013

MJA/CLT/lac

Respectfully submitted,

Mark J Ajluni

M. J. Ajluni
Nuclear Licensing Director

Enclosures:
1. Basis for Proposed Change
2. Operating Licenses Marked-up Pages
3. Operating Licenses Clean Typed Pages

cc: Southern Nuclear Operating Company
Mr. S. E. Kuczynski, Chairman, President & CEO
Mr. D. G. Bost, Executive Vice President & Chief Nuclear Officer
Mr. T. E. Tynan, Vice President – Vogtle
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Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request

Enclosure 1

Basis for Proposed Change

Table of Contents

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

3.1 Categorization Process Description (10 CFR 50.69(b)(2)(i))

3.1.1 Categorization Process Overview
3.1.2 Categorization Process Key Steps
3.1.3 SNC Conformance to NEI 00-04

3.2 Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))

3.2.1 Internal Events with Internal Flooding Probabilistic Risk Assessment (PRA) Model
3.2.2 Fire PRA Model
3.2.3 Other Techniques

3.3 PRA Review Process Results (10 CFR 50.69(b)(2)(iii))

3.3.1 Internal Events (including Internal Flooding) PRA Model
3.3.2 Fire PRA Model

3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

3.4.1 Rule Requirements
3.4.2 Explanation of Process
3.4.3 Conclusion

4.0 REGULATORY EVALUATION

4.1 Significant Hazards Consideration

4.2 Applicable Regulatory Requirements/Criteria

4.3 Precedent

4.4 Conclusions

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

APPENDIX A: Regulatory Guide 1.201 Conformance Matrix

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 Renewed Facility Operating License (OL) Numbers NPF-68 and NPF-81. Specifically, the VEGP Units 1 and 2 OLs will be amended to add proposed OL license conditions (LCs) which will allow for the voluntary implementation of 10 CFR 50.69.

The proposed amendments would revise the VEGP licensing basis to implement 10 CFR 50.69, risk-informed categorization and treatment of structures, systems, and components (SSCs) for nuclear power plants (Reference 1). The Nuclear Regulatory Commission (NRC), by letter dated June 17, 2011 (Reference 2), in response to Southern Nuclear Operating Company's (SNC) letter dated December 6, 2010, granted pilot status for the VEGP 10 CFR 50.69 license amendment request (LAR). Implementation of 10 CFR 50.69 will allow the licensee and the NRC to better focus attention and resources on SSCs that have safety significance, thereby resulting in improved plant safety.

SNC requests approval of the proposed license amendments by August 31, 2013. Initial 10 CFR 50.69 implementation will consist of SNC issuing the VEGP procedures documenting the 10 CFR 50.69 SSC categorization process within 90 days of issuance of the amendments. The resultant VEGP procedures will reflect the 10 CFR 50.69 processes as approved by the NRC and as documented in the NRC safety evaluation report associated with the subject license amendments.

2.0 DETAILED DESCRIPTION

The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. Implementation of 10 CFR 50.69 allows for application of a risk-informed categorization process per 10 CFR 50.69(c) to modify the scope of SSCs subject to special treatment requirements. Alternative treatments per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. Implementation of 10 CFR 50.69 allows the licensee and NRC to better focus attention and resources on SSCs that have safety significance resulting in improved plant safety.

The VEGP 10 CFR 50.69 LAR conforms to scope requirements of 10 CFR 50.69(b)(2). The categorization process described in the LAR conforms to the guidance in NRC Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006 (Reference 3). The categorization process also conforms to the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0 dated July 2005 (Reference 4). SNC has determined that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

3.0 TECHNICAL EVALUATION

3.1 Categorization Process Description (10 CFR 50.69(b)(2)(i))

3.1.1 Categorization Process Overview

The process that SNC will use for categorization of SSCs into the four risk-informed safety classes (RISC) defined in 10 CFR 50.69, specifically RISC-1, RISC-2, RISC-3, and RISC-4, will be specified in SNC procedures. The SNC categorization process conforms to the categorization process guidance provided in NEI 00-04, as endorsed by RG 1.201. Conformance of the SNC categorization process to RG 1.201 is documented in Appendix A of this enclosure. As indicated in Appendix A, no exceptions are taken to RG 1.201.

Per 10 CFR 50.69, once a system is selected for categorization, all the components associated with this system shall be categorized. A system has active and passive components. Consequently, NEI 00-04 provides guidance on categorizing active and passive components, which includes the involvement of a group of experienced and plant-knowledgeable personnel known as the Integrated Decision-making Panel (IDP).

For active components, the SNC procedures implement the active function categorization process defined in NEI 00-04 with clarifications as outlined in section 3.1.3 of this enclosure. The process includes the various quantitative and qualitative evaluations and sensitivity studies intended to provide reasonable confidence that the initial categorization is valid and that validity of the process is maintained.

Passive components are defined as SSCs having only a pressure retaining function. Active components can also have a passive or pressure retaining function. Therefore, the term "passive component" also refers to the passive function of active components, if applicable. NEI 00-04 states that passive component categorization should be performed using the guidance of ASME Code Case N-660 Revision 0, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," (Reference 5) or subsequent versions of this guidance approved by ASME. RG 1.201 states that alternatives to this code case may be submitted for NRC review and approval as part of a specific 10 CFR 50.69 application. SNC has elected to use an alternate method for categorization of passive components. The alternate method was developed after NEI 00-04 was approved but is based on the EPRI risk-informed in-service inspection methodology EPRI TR-112657 Revision B-A (Reference 6) that is cited as an acceptable approach in NEI 00-04. This alternative method is conservative but still provides sufficiently realistic insights with regard to categorization of passive components.

A comparison was conducted of the SNC process for 10 CFR 50.69 passive component categorization and WCAP-16308-NP-A, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process – Wolf Creek Generating Station," (Reference 7). The review included the main body of the WCAP and Appendices, responses to related NRC requests for additional

information (RAIs), and the NRC's associated safety evaluation report (SER) (Reference 8). The results of this review are presented in the following two tables.

Table 1A compares Table 1 of the NRC SER for WCAP-16308-NP-A, titled "NRC Staff Position on Proposed Changes in ASME Code Case N-660 in TR WCAP-16308," and the SNC process for 10 CFR 50.69 passive component categorization. Table 1B presents a comparison of the guidance contained in Appendix A of WCAP-16308-NP-A, specifically pages A-18 through A-30, to the SNC process for 10 CFR 50.69 passive component categorization. WCAP-16308-NP-A pages A-18 through A-30 provide the ASME Code Case N-660 Revision 0, as approved in 2002, marked up to reflect the NRC approved passive categorization process.

Additionally, some clarifications of the SNC process for 10 CFR 50.69 passive component categorization are provided below:

- Consistent with EPRI TR-112657 Revision B-A, the SNC process requires that all safety functions supported by a system be completely evaluated as part of that system's categorization, while WCAP-16308-NP-A would allow an 'interim' categorization.
- Operator actions in the SNC process, when credited, need to meet the requirements of NRC approved EPRI TR-112657 Revision B-A.
- A spectrum of break sizes needs to be evaluated in the SNC process and the one with the highest consequence rank used.
- The SNC process currently limits the application to Class 2 and 3 components (i.e., Class 1 is always high-safety-significant (HSS) for passive categorization).
- The SNC process requires that all relevant configurations be assessed as part of the categorization process (e.g., Table 3-2 of EPRI TR-112657, Revision B-A).

The overall categorization process guidance includes the involvement of a group of experienced and plant-knowledgeable professionals known as the IDP. The IDP is responsible for ensuring that an integrated decision-making process is employed that systematically considers the quantitative and qualitative information available regarding the various modes of plant operation and initiating events, including PRA quantitative risk results and insights (e.g., core damage frequency (CDF), large early release frequency (LERF), and importance measures), deterministic engineering insights (e.g., defense-in-depth, safety margins, and containment integrity); and other pertinent information (e.g., industry and plant-specific operational and performance experience, feedback, and corrective actions program) in the categorization of SSCs. The IDP is responsible for approving the final categorizations.

SNC procedures define the duties and responsibilities of the IDP members, the IDP evaluation, and the IDP overall assessment process as defined in NEI-00-04 without exception. When holding IDP meetings, the SNC procedure requires a quorum for the IDP. A quorum consists of at least five qualified persons, collectively having site specific expertise in the following functional areas: plant

operations (senior reactor operator (SRO) qualified), safety analysis, design engineering, systems engineering, and PRA. The training and qualification of each IDP member is documented.

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
{I-2.0} [I-2.0]	Objection requiring qualification	The owner shall define the boundaries included in the scope of the RISC evaluation process **subject to the constraints in paragraph 50.59(c)(1)(v) that the categorization must be performed for entire systems**. Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class.1 items shall be classified High Safety Significant (HSS) and the provisions of the RISC evaluation shall not apply.	Conforms to NRC Mark-up. SNC limits the application to Class 2 and 3 only. This is conservative.
{I-3.0} [I-3.0]	Objection requiring qualification	Additionally, information considered relevant to the classification shall be collected for each piping segment e.g., information regarding design basis accidents, at-power risk, shutdown risk,	Conforms to NRC Mark-up. SNC procedure requires all initiating events and operating modes to be evaluated. This is consistent with the

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
		containment isolation, flooding, fires, seismic conditions, etc.). **Consistent with 50.69(c)(1)(ii), the classification must address all initiating events and plant operating modes.**	EPRI TR-112657 SER.
{ } [I-3.1.1(a)(4)]	Objection requiring qualification	*Entire proposed section should be deleted*	Conforms to NRC Mark-up. The applicable SNC procedure requires that breaks from small to large are postulated and the break size with the highest consequence rank be used. This is consistent with the EPRI TR-112657 SER.
{I-3.1.2} [I-3.1.2]	Objection requiring qualification	...in accordance with (a) through (d) below. In assessing the appropriate consequence category, risk information for all initiating events, including fire and seismic, should be considered. **To capture the risk importance from**	Clarification to NRC Mark-up. Consistent with the EPRI TR-112657 SER, the applicable SNC procedure requires all initiating events, operating modes and external

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
		initiating events for which no quantitative PRA is available, any piping segment supporting a safe shutdown pathway would be classified as HSS.	hazards to be evaluated for each individual system application. Experience has shown that these evaluations, which complete the risk assessment process, identify portions of the respective system that are not high from an at-power perspective, but can be important from a shutdown perspective or their support of other systems (e.g., Emergency Core Cooling Systems during internal events or external hazards).
{I-3.1.2(b)(3)} [I-3.1.2(b)]	No objection	In lieu of Table I-2, quantitative indices may be used to assign consequence categories in accordance with Table I-5.	Clarification to NRC Mark-up. Per the applicable SNC procedure, quantitative indices may be used to assign consequence categories in accordance with Table 5 in lieu of Table 2 provided the quantitative basis of Table 2 (e.g., one full train

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
			unavailability approximately 10^{-2}, exposure time) is consistent with the failure scenario being evaluated. Differences in the consequence rank between the use of Table 2 and 5 shall be reviewed, justified and documented or the higher consequence rank assigned. This is further clarification of the technical intent of the methodology and is consistent with or more conservative than the NRC position.

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
{I-3.1.3(a)(3)} [I-3.2.2(b)(1)]	Objection requiring qualification	Even when taking credit for plant features and operator actions, failure of the piping segment will not ~~directly~~ fail ~~another~~ high safety-significant function.	Clarification to NRC Mark-up. Consistent with the EPRI TR-112657 SER, the applicable SNC procedure uses the term "basic safety function" (e.g., reactivity control, core cooling, heat sink, RCS inventory) instead of "high safety significant function". This term is more complete and/or conservative.
{I-3.1.3(a)(4)} [I-3.2.2(b)(2)]	Objection requiring qualification	Failure of the piping segment will not result in failure of ~~another~~ high safety significant piping segment, e.g. through indirect effects	Clarification to NRC Mark-up. Consistent with the EPRI TR-112657 SER, the applicable SNC procedure uses the term "basic safety function" (e.g. reactivity control, core cooling, heat sink, RCS inventory) instead of "high safety significant function". This term is more complete and/or conservative.

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
{I-3.1.3(a)(5)} [I-3.2.2(b)(3)]	Objection requiring qualification	Consideration changed and moved to new section I-3.2.2(b)(3), ~~Even when taking credit for plant features and operator actions~~, failure of the piping segment will not prevent ~~or adversely affect~~ the plant'~~s capability to reach or maintain~~ reaching or maintaining safe shutdown conditions.	Clarification to NRC Mark-up. Operator actions can be credited provided they are evaluated in accordance with EPRI TR-112657 Revision B-A. This is consistent with N-660 Revision 0 and the EPRI TR-112657 SER.
{I-3.1.3(b)(3)} [I-3.2.2(b)(6)]	Objection requiring qualification	~~Even when taking credit for plant features and operator actions, f~~ **F**ailure of the piping segment will not result in releases of radioactive material that would result in the implementation of off-site emergency response and protective actions.	Clarification to NRC Mark-up. Operator actions can be credited provided they are evaluated in accordance with EPRI TR-112657 Revision B-A. This is consistent with N-660 Revision 0 and the EPRI TR-112657 SER.
{I-3.2.2(b)} [I-3.2.2(b)(5)]	No objection	The plant condition monitoring would identify any known active degradation mechanisms in the pipe segment prior to	Clarification to NRC Mark-up. While condition monitoring programs

Table 1A: 10 CFR 50.69 Passive Component Categorization Process Comparison of NRC Staff Positions on WCAP-16308-NP-A Methods (Table 1 of NRC SER for WCAP-16308-NP-A) And the SNC Passive Component Categorization Process			
{N-660 Revision 0 Section} [WCAP-16308 Section]	**NRC Position**	**NRC Mark-up of WCAP-16308 Proposed Changes to ASME Code Case N-660 Revision 0 (additions in bold, deletions in strikeout, comments in italics)**	**SNC Passive Component Categorization Process**
		its failure in test or an actual demand event (e.g., flow accelerated corrosion program).	certainly exist, "monitoring" is not an element of the categorization process and need not be addressed in categorization guidance. This wording was originally added to various draft revisions of N-660 to allow for the postulation of small or medium break sizes instead of large breaks. The applicable SNC procedure requires that the full spectrum of break sizes be evaluated and the one with the highest consequence be used.

Table 1B: 10 CFR 50.69 Passive Component Categorization Process Comparison of the NRC Approved Passive Categorization Process (Mark-up of ASME Code Case N-660 Revision 0) And the SNC Passive Component Categorization Process			
WCAP-16308-NP-A Page	**Section**	**Topic**	**SNC Passive Component Categorization Process**
A-18	1100 Scope	WCAP-16308-NP-A allows process to be applied to Class 1 components	SNC limits the application to Class 2 and 3 only. This is conservative.
A-18	1320 Required Disciplines	WCAP-16308-NP-A has requirements for IDP	SNC has a standalone procedure for the IDP.
A-19	9000 Glossary	WCAP-16308-NP-A has a Glossary	The applicable SNC procedure has similar information.
A-21	I-2.0 Scope Identification	WCAP-16308-NP-A has wording as follows: The owner shall define the boundaries included in the scope of the RISC evaluation process subject to the constraints in paragraph 50.59(c)(1)(v) that the categorization must be performed for entire systems. Items optionally classified to Class 1 and Class 1 items connected to the reactor coolant pressure boundary, as defined in paragraphs 10 CFR 50.55a (c)(2)(i) and (c)(2)(ii), are within the scope of the RISC evaluation process. All other Class.1 items shall be classified High Safety Significant	SNC limits the application to Class 2 and 3 only. This is conservative.

Table 1B: 10 CFR 50.69 Passive Component Categorization Process Comparison of the NRC Approved Passive Categorization Process (Mark-up of ASME Code Case N-660 Revision 0) And the SNC Passive Component Categorization Process			
WCAP-16308-NP-A Page	**Section**	**Topic**	**SNC Passive Component Categorization Process**
		(HSS) and the provisions of the RISC evaluation shall not apply	
A-22	I-3.1.1(a)	WCAP-16308-NP-A allows leak before break to support the small break assumption	SNC does not which is conservative.
A-22	I-3.1.1(b) and (e)	WCAP-16308-NP-A allows operator actions to be credited but no criteria or requirements are identified	Operator actions can only be credited provided they are evaluated in accordance with EPRI TR-112657 Revision B-A. This is consistent with the EPRI TR-112657 SER.
A-22	I-3.1.1	WCAP-16308-NP-A does not require that success criteria diagrams be developed for relevant initiating events	Based upon lessons learned from application of the methodology, SNC's applicable procedure requires that success criteria diagrams be developed for relevant initiating events.
A-22	I-3.1.1	WCAP-16308-NP-A does not require that each of the relevant operating configurations identified in EPRI TR-112657, Rev B-A, (Chapter 3.3.3) be evaluated	Based upon lessons learned from application of the methodology, SNC's applicable procedure requires that this be done. This provides for a more comprehensive and complete assessment.

Table 1B: 10 CFR 50.69 Passive Component Categorization Process Comparison of the NRC Approved Passive Categorization Process (Mark-up of ASME Code Case N-660 Revision 0) And the SNC Passive Component Categorization Process			
WCAP-16308-NP-A Page	**Section**	**Topic**	**SNC Passive Component Categorization Process**
A-22	I-3.1.2	WCAP-16308-NP-A wording: To capture risk importance from initiating events for which no quantitative PRA is available, any piping segment supporting a safe shutdown pathway would be classified as HSS.	SNC applicable procedure has requirements consistent with the EPRI TR-112657 Revision B-A SER to evaluate the impact of shutdown events and external hazards.
A-22	I-3.1.2(a)	WCAP-16308-NP-A has no requirement to investigate differences in the consequence rank in the use of Table I-1 and I-5.	SNC does which provides for a more comprehensive and complete assessment.
A-23	I-3.1.2(b)	WCAP-16308-NP-A has no requirement to investigate differences in the consequence rank in the use of Table I-2 and I-5.	SNC does which provides for a more comprehensive and complete assessment.
A-23	I-3.1.2(b)	WCAP-16308-NP-A has no requirement that for defense in depth purposes, all postulated failures leading to "zero defense" (i.e., no backup trains) shall be assigned a high consequence rank.	SNC does which provides for a more robust consequence ranking.
A-23	I-3.1.2(c)	WCAP-16308-NP-A has no requirement to investigate differences in the consequence	SNC does which provides for a more comprehensive and complete assessment.

Table 1B: 10 CFR 50.69 Passive Component Categorization Process Comparison of the NRC Approved Passive Categorization Process (Mark-up of ASME Code Case N-660 Revision 0) And the SNC Passive Component Categorization Process			
WCAP-16308-NP-A Page	**Section**	**Topic**	**SNC Passive Component Categorization Process**
		rank in the use of Table I-3 and I-5.	
A-24	I-3.2.2(b)	WCAP-16308-NP-A allows operator actions to be credited but no criteria or requirements are identified.	Operator actions can only be credited provided they are evaluated in accordance with EPRI TR-112657 Revision B-A. This is consistent with the EPRI TR-112657 SER.
A-24/A-25	I-3.2.2(b) and (c)	"Additional Considerations" – wording is generally consistent except for editorial changes. WCAP-16308-NP-A also contains the following: The plant condition monitoring program would identify any known active degradation mechanisms in the pipe segment prior to its failure in test or an actual demand event (e.g., flow accelerated corrosion program).	While condition monitoring programs certainly exist, "monitoring" is not an element of the categorization process and need not be addressed in categorization guidance. This wording was originally added to various draft revisions of N-660 to allow for the postulation of small or medium break sizes instead of large breaks. The applicable SNC procedure requires that the full spectrum of break sizes be evaluated and the one with the highest consequence rank be used.
A-30	Table I-5	Use of the equal sign is inconsistent with EPRI TR-112657 Revision B-A.	SNC applicable procedure consistent with the EPRI TR-112657 Revision B-A SER
N/A	N/A	WCAP-16308-NP-A does not contain a table identifying relevant operating configurations for postulating when a piping segment fails.	The SNC applicable procedure has such a table. This provides for a more comprehensive and complete assessment.

3.1.2 Categorization Process Key Steps

The key steps of the SNC 10 CFR 50.69 categorization process for active and passive components, including IDP deliberation, are provided below.

- The SNC 50.69 categorization process utilizes a combination of deterministic engineering (e.g., defense-in-depth, safety margins, and containment integrity), quantitative PRA, and qualitative risk insights to determine if a SSC performs one or more HSS functions and identifies those functions. The term "HSS function" is synonymous with the term "safety-significant function", as defined in 10 CFR 50.69.

- The process is applied only to those plant systems or structures (herein referred to simply as system) that VEGP chooses to categorize.

- Once a plant system is selected for categorization, all components within that system are categorized into one of the following categories:

    - RISC-1: safety-related SSCs that perform HSS functions
    - RISC-2: nonsafety-related SSCs that perform HSS functions
    - RISC-3: safety-related SSCs that perform low safety-significant (LSS) functions
    - RISC-4: nonsafety-related SSCs that perform LSS functions

- The identification of components within a system is based on the VEGP Plant Data Management System (PDMS) which is a design-controlled database that reflects the current plant configuration.

- The functions performed by the system are identified, including but not necessarily limited to, design basis functions, maintenance rule functions, and functions credited for mitigation and prevention of severe accidents.

- Plant and industry operating experience with the system being categorized is obtained and evaluated for applicability and insights.

- System functions are qualitatively categorized using the set of deterministic questions identified in NEI 00-04.

- For each component in the system, the function(s) that the component supports are identified.

- Component functional importance is determined through an integrated, systematic process that considers all of the following factors:

    - Internal Events (including Internal Flooding) At-Power – the VEGP Internal Events (including Internal Flooding) PRA models severe accident scenarios resulting from internal initiating events, including internal flooding, occurring at full power operation. Importance measures related to CDF and LERF are used to identify SSCs that are HSS with respect to internal events.

- Fire Hazards - the VEGP Fire PRA models severe accident scenarios resulting from fire-initiated events, occurring at-power operation. Importance measures related to CDF and LERF are used to identify SSCs that are HSS with respect to fire events.

- Seismic hazards - Seismic hazards are assessed using the VEGP Seismic Margins Analysis (SMA) developed for the Individual Plant Examination of External Events (IPEEE) (References 9 and 10). As stated in NEI 00-04, components that are credited as part of the seismic safe shutdown path are considered HSS with respect to seismic risk.

- Other External hazards – Other external hazards, such as high winds, external flooding, and accidents from transportation routes and nearby industrial facilities, are assessed using the results of the VEGP IPEEE. The other external hazards evaluation was updated in 2011 using guidance provided in ASME/ANS RA-Sa-2009 (Reference 11). The evaluation included the impact of construction activities at Vogtle Units 3 and 4. The evaluation concluded that the other external hazards screened out. Therefore, components are LSS with respect to risk from other external events.

- Low Power/Shutdown Risk – In the absence of low power or shutdown PRA risk models, the low power risk is evaluated using at-power PRA for those cases where the at-power PRA model is valid for evaluating risk for low power conditions. In addition, in those cases where at-power PRA risk model is not appropriate for use at low power conditions, the low power risk will be evaluated by qualitatively assessing the function(s) performed by the system as outlined in NEI 00-04 section 9.2.2 review of risk information item 6. A component is assigned a preliminary risk, which is higher of these two risks where applicable. For shutdown risk, VEGP has performed and maintains a Shutdown Defense-In-Depth Evaluation, in accordance with the NUMARC 91-06 Program (Reference 12). Components that play a primary or alternate role in accomplishing key shutdown safety functions or whose failure would initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.) are identified as HSS with respect to shutdown risk.

- Sensitivity Studies – For PRA-modeled components, several sensitivity studies are performed as outlined in NEI 00-04 to ensure that assumptions in the PRA are not masking the importance of an SSC.

- Passive Component Categorization – For the purpose of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components are evaluated through a process that utilizes the guidance in EPRI TR-112657, Revision B-A, with the following additional constraints:

  - Component failure is assumed and only the consequence evaluation is performed.
  - Additional deterministic considerations (e.g., defense in depth, safety margins) are applied.
  - ASME Class 1 components are, by default, categorized as HSS with respect to passive risk.
  - Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model.

- Active components are assigned a qualitative risk based on the system function(s) they support. If a component supports one or more HSS functions, it is considered to be HSS initially. A component may be assigned a qualitative risk of LSS, even it supports an HSS function, if a credible failure of the component would not preclude the fulfillment of the HSS function (e.g., locked open valve).

- An active component is assigned an overall categorization of HSS if one or more of the following evaluations identify that component as HSS: integrated results of Internal Events (including Internal Flooding) PRA and Fire PRA assessment, seismic hazards, other external hazards, shutdown risk, and qualitative risk based on the system function(s). A passive component is categorized as HSS per processes and criteria discussed herein. In addition, an active component having an HSS passive categorization is categorized as HSS.

- Components that are still LSS are then evaluated for their role in providing defense-in-depth and, if appropriate, upgraded to HSS.

- Overall Risk Sensitivity Study – For PRA-modeled components, an overall risk sensitivity study is used to confirm that the categorization process results in acceptably small increases to CDF and LERF. The overall risk sensitivity study is performed using guidance outlined in NEI 00-04. SNC will use a factor of 3 to increase unreliability and unavailability of RISC-3 LSS components modeled in PRAs.

- For components that are HSS or that are LSS but support HSS functions, the associated critical attributes are identified.

- The above categorization results are presented to the IDP for review and approval. The VEGP IDP is staffed with expert, plant-knowledgeable, and process-trained members whose collective expertise includes, at a minimum,

PRA, safety analysis, plant operations (SRO qualified), design engineering, and system engineering. The IDP will review the categorization results and make the final determination on the safety significance of system functions and components, in accordance with the following requirements:

- The basis for the adequacy of the PRA results; the results of non-modeled hazards; and the system functions and the basis for their categorization are reviewed to ensure adequacy in supporting the process.

- The detailed categorization results, including results of sensitivity studies, and defense-in-depth considerations are reviewed for completeness and adequacy.

- Components that are HSS from one of the following evaluations cannot be categorized as LSS, although the IDP may request that further clarification and/or analysis be performed and brought back to the IDP.
  - Internal Events PRA
  - Non-PRA evaluations of seismic, other external events, or shutdown risk hazards.
  - Passive categorization

- Components that are HSS only from the Fire PRA may be categorized as LSS if the integrated assessment of component risk importances over all PRA models shows the integrated risk importance measures meet the LSS criteria.

- The IDP may change the categorization of a component from LSS to HSS based on its assessment and use of conservative decision-making. Conversely, as outlined in NEI 00-04, the IDP may re-categorize components from HSS to LSS if a credible failure of the component would not preclude the fulfillment of the HSS function and if the component did not meet one of the previously stated criteria precluding LSS categorization.

- Once the safety significance of the components in the selected system has been approved by the IDP, the components will be categorized into one of the four RISC categories as detailed at the beginning of this section 3.1.2.

- Periodic reviews are conducted at least once every two Unit 1 refueling outages to ensure continued validity and performance monitoring for those SSCs that have been categorized.

- The categorization results, supporting bases, IDP meeting minutes, and results of periodic reviews are documented as Quality Assurance records and maintained for the life of the plant.

3.1.3 SNC Conformance to NEI 00-04

In summary, the SNC 10 CFR 50.69 process to be implemented at VEGP addresses all aspects of the guidance provided in NEI 00-04 as endorsed by RG 1.201, with the following clarifications.

Clarification 1: Approach Used to Risk Rank System Functions:

The NEI 00-04 process uses component risk significance (determined using the methodology outlined in sections 5.0 and 6.0) to determine function risk significance (section 7.0). For example, if a component identified by PRA as HSS is mapped to a function, then the function is considered HSS. Subsequently, all the components mapped to this function are preliminary categorized as HSS. NEI-00-04 section 9.2 states that

> "The IDP approval of the categorization of system functions, based on the coarse mapping of components to system functions, would be used to define the safety significance of each SSC as described in Section 10. Thus, if a system function is found to be safety-significant by the IDP, then all components required to support that function would initially be considered safety-significant. If a more detailed categorization of the SSCs associated with a safety-significant function is performed after the initial IDP, then the basis for that re-categorization must be considered in a follow-up IDP session. In this follow-up session, the IDP would be expected to review the basis for the re-categorization and to assess the impact of this re-categorization on the risk importance and defense in depth implications using the same criteria as in the original IDP session for candidate low safety-significant SSCs."

NEI 00-04 section 10.2 provides guidance on performing detailed categorization of the SSCs and the criteria for assignment of low safety significance for an SSC supporting a safety-significant function. Consistent with section 9.2, section 10.2 further states that

> "For SSCs that are re-categorized to a lower classification (e.g., components in a safety-significant function that are determined to be LSS based on the above considerations), the new categorization and its basis should be presented to another session of the IDP to be re-categorized using the same rigor as described in Section 9. If the SSCs being considered for re-categorization to a lower classification are modeled in the PRA, then the risk sensitivity described in Section 5 would need to be completed prior to presentation to the IDP."

The following is a clarification of the way SNC has applied the above mentioned steps when performing a trial categorization of three systems.

SNC evaluates each function using the seven questions outlined in section 9.2.2 of NEI 00-04. If any of the seven questions is answered "yes" for a function, the function is considered HSS. The components mapped to the functions are considered preliminary HSS. This is called "qualitative assessment".

The component safety significance assessment is performed using guidance provided in section 5.0 of NEI 00-04. This portion of the assessment is called "Active Risk Assessment".

The passive risk for applicable components is determined through the process outlined in this LAR. This portion of the assessment is called "Passive Risk Assessment".

A component is assigned a preliminary risk, which is the highest of the following three risks – qualitative assessment, active risk assessment, and passive risk assessment. Then the defense-in-depth and safety margin analysis are performed using the guidance provided in section 6.0 of NEI 00-04. The preliminary risk of a component is adjusted, as needed, to reflect insights gained from defense-in-depth and safety margin analysis. The overall risk sensitivity study is performed per section 8.0 of NEI 00-04.

The categorization results are provided to the IDP for their review per section 9.0 of NEI 00-04. The IDP uses information in section 9.0 and 10.2 to evaluate the categorization results and disposition them appropriately.

The SNC process meets the guidance of NEI 00-04 in the manner described herein. SNC utilizes all the sections of the NEI 00-04 methodology. The sequence in which the sections (sections 5.0, 6.0, 9.2.1, 9.2.2, 9.2.3, and 10.2) are applied varies without compromising the fidelity of the methodology. The SNC process provides a more logical and intuitive top-down approach for the qualitative portion of the assessment. System functions are categorized first, followed by component-to-function mapping, and ending with component categorization. The SNC process improves the overall process efficiency while applying all the elements of the NEI 00-04 methodology.

<u>Clarification 2: Mapping of Truly Passive Components to Each Function:</u>

In section 4.0, under "Identification of System Functions", NEI 00-04 states that the classification of SSCs having only a pressure retaining function (also referred to as passive components) or the passive function of active components should be performed using passive component categorization methodology.

In the same section 4.0, under "Coarse Mapping of Components to Functions", NEI 00-04 states that "The assignment of SSCs to each of the functions is necessary at this step to ensure that every SSC with a tag identifier for the system being considered is represented in at least one of the functions." SNC complies with this step with the following clarification. The insights obtained from the trial categorization of three systems indicate that truly passive components like pipes and thermowells should be assigned to a pressure retention function only. These types of components do not perform an active function. During the trial categorization, SNC observed that when these components were mapped to various functions (as currently stated in NEI 00-04), some of these components were determined to be candidate HSS only because they were in the flow path of safety significant function(s). However, when the IDP review was performed per

section 9.0 and 10.2 of NEI 00-04, their safety significance was ultimately determined using the results of the passive component categorization methodology. The current verbiage (quoted above) in NEI 00-04 does not allow this exclusion.

SNC takes the position that by mapping truly passive components only to the passive function, the methodology becomes more efficient without compromising the fidelity of the process.

Clarification 3: Components in the Flow Path vs. Components Required for a Function(s):

On page 28, NEI 00-04 states that "Define the pathway associated with each function and then define the components associated with that pathway." SNC complies with this step with the following clarification.

The current wordings in NEI 00-04 imply that each and every component in the pathway of a function should be mapped to the function. However, the insights obtained from the trial categorization indicate that there will be instances when select components will be in the function pathway, but these components are not required to perform the function. Consider the following two examples.

- When categorizing a system, functions performed by the system are identified. One of the functions performed by the Chemical and Volume Control System (CVCS) is to maintain primary coolant inventory during normal operations, startup, and shutdown (includes operation in support of accident response when restoring CVCS inventory control). When a pathway is traced on piping and instrumentation diagrams (P&ID) for this function (i.e., maintain primary coolant inventory), the pathway includes components associated with demineralizer beds. However, these components are not required to perform this function other than a passive role of pressure retention to maintain a flow path. When categorizing a system, the risk associated with the pressure retention function is captured in a separate function (e.g., maintain CVCS pressure boundary during normal operations, startup, and shutdown). Therefore, it is not necessary to map the components associated with demineralizer beds to this function (i.e., maintain primary coolant inventory). Note that the components associated with demineralizer beds are required for other functions (such as control primary coolant pH during normal operations, startup, and shutdown; limit primary coolant radioactivity levels during normal operations, startup, and shutdown; etc.); therefore, they are mapped to these functions.

- Virtually all mechanical systems have drain, test, and vent valves. These components are there to support a specific maintenance or test function. However, they will show up in the pathway for other functions performed by a system being categorized. The failure of these components will not impact other system functions. Assuming flow diversion does not affect specific function, it is not necessary to map these components to the other functions identified for the system being categorized.

Therefore, SNC takes the position that only components that are required to perform an active function will be mapped to that function. This makes the methodology more efficient without compromising the fidelity of the process.

3.2 Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))

SNC employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the PRA models for VEGP. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews.

The NRC-endorsed NEI 00-04 methodology also allows for categorization of SSCs to be performed in the absence of quantifiable PRA models for the following other risks - seismic hazards, other external events hazards (e.g., tornados, high winds, transportation, etc.), and low power/shutdown hazards.

The following sections describe the approach used to meet the requirements of 10 CFR 50.69(b)(2)(ii).

3.2.1 Internal Events with Internal Flooding PRA Model

VEGP has two operating units of Westinghouse 4-loop pressurized light water reactors with a core thermal power rating of 3625 MWt. The VEGP baseline PRA models CDF and LERF due to internal events, including internal flooding, at-power. There is one model for VEGP because each unit is a mirror image of the other from a PRA perspective. The model has been significantly upgraded since the IPE PRA model and reflects the as-built, as-operated plant. The baseline PRA is a large fault tree methodology model, using the EPRI R&R Workstation (CAFTA) software and FTREX software for quantification. The baseline CDF and LERF values are 2.25E-05/yr and 7.37E-08/yr, respectively.

3.2.1.1 PRA Maintenance/Updates and Application to 10 CFR 50.69 Categorization Process

The administrative controls applicable to the PRA models used to support the 10 CFR 50.69 program ensure that these models reflect the as-built, as-operated plant. Plant changes, including physical modifications and procedure or operating practice changes, are reviewed prior to implementation to determine if they could impact the PRA models. If so, the process then determines the quantitative significance of the change and, if appropriate, implements the PRA model change concurrently with the plant change. Otherwise, the PRA model change is prioritized for implementation at a routine model update. Such pending changes are considered when evaluating other changes until they are fully implemented into the PRA models. Routine updates are performed as a minimum every two refueling cycles.

Internal Events (including Internal Flooding) PRA Maintenance and Update

The SNC risk management process ensures that the applicable PRA model reflects the as-built and as-operated plant for each of the VEGP units. The process delineates the responsibilities and guidelines for updating the full power internal events and internal fire PRA models at all operating SNC sites, and includes both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential impact areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files.

Review of Plant Changes for Incorporation into the PRA Model – Refueling and Non-Refueling Outage Period

(1) Plant changes (including both physical modifications to the facility and changes to procedures or operating practices) are reviewed as follows:

   a. Refueling Outage Cycle Implementation: Six months prior to each refueling outage, changes slated for implementation during the refueling outage are reviewed to determine if any plant changes planned for the refueling outage could have an impact on the PRA models used to support 10 CFR 50.69 categorization.

   b. Non-Refueling Outage Cycle Implementation: Changes slated for implementation during non-refueling outage periods are monitored concurrently with the planning phase to determine if changes planned for implementation prior to the next refueling outage could have an impact on the PRA models used to support 10 CFR 50.69 program risk calculations.

(2) When considered along with other changes, plant changes are categorized as "Required" if they are considered to have a quantitative significant impact. "Required" plant changes are incorporated in the model to coincide with the time that the changes are implemented at the plant (refueling outage or non-refueling outage period). Plant changes that do not meet the criteria for "Required" are assigned to other categories (i.e., "High Priority", "Moderate Priority", and "Low Priority") and are incorporated at the next regular update, or in a future update, based on the priority category assigned. Accordingly, these changes are placed into a "Pending" file, which may be maintained in a database, spreadsheet, or other appropriate tool. When new changes are being evaluated (as discussed above) these pending changes are also considered with the newly proposed changes for their cumulative effect in order to determine the appropriate schedule and priority of PRA model changes.

(3) PRA updates, including updates to reliability data, failure data, initiating events frequency data, human reliability data, and other PRA inputs, are performed at least once every two Unit 1 refueling cycles.

(4) If PRA model errors are discovered, they are reviewed within the SNC Corrective Action Program to determine the quantitative impact on PRA results. A similar prioritization process to the plant change evaluation process is then applied to determine the appropriate priority and schedule for correcting the error, based on the significance of the quantitative impact.

(5) When a PRA model change is required but cannot be immediately implemented for a significant plant change or model error, either:

a. Alternative analyses to conservatively bound the expected risk impacts of the changes are performed. In such a case, these alternative analyses become part of the 10 CFR 50.69 categorization process until the plant changes are incorporated into the PRA model during the next update.

b. Appropriate administrative restrictions, if any, on the use of the categorization results are put in place until the model changes are completed.

3.2.1.2 Identification and Disposition of 10 CFR 50.69 Application Specific Sources of Uncertainty for Internal Events (including Internal Flooding) PRA Model

The purpose of this section is to identify and disposition the impact of epistemic uncertainty from the internal events (including internal flooding) PRA model on the 10 CFR 50.69 program. The baseline internal events (including internal flooding) PRA model documents assumptions and sources of uncertainty. These have been reviewed during the model peer review. Therefore, in order to disposition the impact of epistemic uncertainty from the internal events (including internal flooding) PRA model on the 10 CFR 50.69 program, the approach taken is to review these documents to identify those items which may be directly relevant to the 10 CFR 50.69 program, to perform sensitivity analyses where appropriate, to discuss the results, and to provide a disposition for the 10 CFR 50.69 program.

Background:

NEI 00-04 outlines a set of sensitivity studies that must be performed when using the PRA models for different risk hazards. NEI 00-04 requires the following sensitivity studies that evaluate the potential impact on the categorization results of several defined areas of uncertainty when using the internal events (including internal flooding) PRA.

- Increase all human error basic events to their 95th percentile value
- Decrease all human error basic events to their 5th percentile value
- Increase all component common cause events to their 95th percentile value
- Decrease all component common cause events to their 5th percentile value
- Set all maintenance unavailability terms to 0.0
- Perform any applicable sensitivity studies identified in the characterization of PRA adequacy

The first five sensitivity studies are self-explanatory. Therefore, they are not discussed further in this section. The discussion in this section is focused on the requirement to perform: "Any applicable sensitivity studies identified in the characterization of PRA adequacy". In order to determine which, if any, such studies are needed, it is first necessary to evaluate the model and application-specific sources of uncertainty.

Methodology for Identifying 10 CFR 50.69 Application Specific Sources of Uncertainty for Internal Events (including Internal Flooding) PRA Model:

The baseline PRA model uncertainty report was developed using guidance provided in NUREG-1855 (Reference 13). As described in NUREG-1855, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness (or scope and level of detail)" uncertainties.

Parametric uncertainty was addressed as part of the VEGP aleatory uncertainty analysis as documented in Reference 14.

Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. The assumptions are defined consistent with the definition provided in NUREG-1855. Plant-specific assumptions made for each of the VEGP internal events PRA technical elements are noted in the individual PRA notebooks. These assumptions were collected from each notebook and were evaluated to determine if they were related to a source of modeling uncertainty. A source of uncertainty having a potential impact on the 10 CFR 50.69 application was retained for further evaluation.

EPRI TR-1016737 ((Reference 15, Table A-3) compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element. An evaluation of each generic source of modeling uncertainty was performed.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA but are only considered for their impact on a specific application.

Disposition of 10 CFR 50.69 Application Specific Sources of Uncertainty for Internal Events (including Internal Flooding) PRA Model:

From the characterization of potential sources of uncertainty in the baseline PRA model and of supplementary issues from EPRI TR-1016737 (Reference 15 Table A-3), the following items were identified as having a potentially important impact on the internal events (including internal flooding) PRA results.

1. Pressure-Induced SGTR
2. Seasonal Impacts on Initiating Events
3. Basis for human error probabilities (HEPs)

The potential impacts of each of the above three sources of uncertainty on the 10 CFR 50.69 application were evaluated to determine if any of them should be retained as part of the standard sensitivity studies performed for the internal events (including internal flooding) PRA model.

Table 2 describes the analyses performed to assess the impact of these sources of model uncertainty on 10 CFR 50.69 application. The results of evaluation indicated that these three epistemic sources of uncertainties would not have an impact on the 10 CFR 50.69 application.

Table 2: VEGP Internal Events (including Internal Flooding) PRA Model Sources of Uncertainty and Impact on 10 CFR 50.69 Program				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**Sensitivity Conclusions**	**Disposition of Conclusions**
Pressure-Induced SGTR: High reactor coolant system (RCS) pressure impacts the potential for induced SGTR. Only the Large loss of coolant accident (LOCA), Reactor vessel rupture, and the Medium LOCA scenarios are treated as low RCS pressure scenarios. Model may over-estimate contribution of pressure-induced SGTR (PI-SGTR) to LERF.	Scenarios with significant reactor coolant pump (RCP) seal leakage, a stuck open pressurizer valve, or a pressurizer power-operated relief valve (PORV) open for feed and bleed are conservatively considered RCS high pressure scenarios	Sensitivity cases were performed by reclassifying several scenarios (RCP seal leak, stuck-open relief valve (SORV), etc.) from high pressure to low pressure to determine impact on large early release frequency (LERF).	Several scenarios (RCP seal leak, SORV, etc.) were reclassified from high pressure to low pressure and LERF re-evaluated. There was no change in LERF. Therefore, 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.	No explicit sensitivity evaluation is needed for 50.69 impact.

Table 2: VEGP Internal Events (including Internal Flooding) PRA Model Sources of Uncertainty and Impact on 10 CFR 50.69 Program				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**Sensitivity Conclusions**	**Disposition of Conclusions**
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 needed in cold weather.	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.

Table 2: VEGP Internal Events (including Internal Flooding) PRA Model Sources of Uncertainty and Impact on 10 CFR 50.69 Program				
Source of Epistemic Uncertainty	**Related Assumptions**	**Sensitivity Case**	**Sensitivity Conclusions**	**Disposition of Conclusions**
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 THERP method is applied for pre-initiator HFEs. The 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 (Reference 4) process).	Since there are no specific HFEs affected by the 10 CFR 50.69 application, no additional explicit sensitivity evaluation is needed for 10 CFR 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.

3.2.2 Fire PRA Model

A state-of-the-art VEGP Fire PRA model was developed using the guidance provided in the ASME/ANS PRA standard (RA-Sa-2009) (Reference 11) in early 2012. A peer review was conducted during the week of February 13, 2012. The results of the peer review are provided in section 3.3.2 of this LAR.

VEGP has two operating units of Westinghouse 4-loop pressurized light water reactors with a core thermal power rating of 3625 MWt. The VEGP baseline Fire PRA models core damage frequency and large early release frequency due to internal fire, at power. There are two models, one for each unit, because each unit is not a mirror image of the other from a Fire PRA perspective. The models reflect the as-built, as-operated plant. Because the models have been developed recently, they have not gone through major upgrades or revision. However, SNC anticipates that they will continue to be refined over the coming years, similar to the internal events (including internal flooding) model. The Fire baseline PRA is a large fault tree methodology model, using the EPRI R&R Workstation (CAFTA) software and FTREX software for quantification. The baseline CDF and LERF values are 5.22E-05/yr and 2.10E-06/yr, respectively for U1 and 5.19E-05/yr and 2.38E-06/yr, respectively for U2.

3.2.2.1 Fire PRA Maintenance/Updates and Application to 10 CFR 50.69 Categorization Process

The Fire PRA models would be maintained and updated in the same manner as the internal events (including internal flooding) PRA model. This process is described in section 3.2.1.1 of this LAR.

3.2.2.2 Identification and Disposition of 10 CFR 50.69 Application Specific Sources of Uncertainty for Fire PRA Model

The purpose of this section is to describe the evaluation of epistemic uncertainty in the Fire PRA (FPRA) model for impact on the implementation of 10 CFR 50.69.

Background:

The Vogtle FPRA model includes various sources of uncertainty that occur because there is inherent randomness in the elements that comprise the FPRA and because the state of knowledge in these elements continues to evolve.

NEI 00-04 outlines standard sensitivity studies that must be performed when using a PRA for different risk hazards. NEI 00-04 requires the following sensitivity studies that exercise key areas of uncertainty when using the Fire PRA.

- Increase all human error basic events to their 95th percentile value
- Decrease all human error basic events to their 5th percentile value
- Increase all component common cause events to their 95th percentile value

- Decrease all component common cause events to their 5th percentile value
- Set all maintenance unavailability terms to 0.0
- Take no credit for manual suppression
- Perform any applicable sensitivity studies identified in the characterization of PRA adequacy

The first five sensitivity studies are self-explanatory. Therefore, they are not discussed further in this section. The discussion in this section is focused on the last two sensitivity studies - "No credit for manual suppression" and "Any applicable sensitivity studies identified in the characterization of PRA adequacy."

Take No Credit for Manual Suppression:

In performing this sensitivity study for the 10 CFR 50.69 categorization process, VEGP will set all manual suppression credit in the fire PRA model to zero (i.e., manual suppression fails) with the exception of scenarios involving main control room abandonment. Manual suppression will be credited for scenarios involving main control room abandonment because if a fire is potentially necessitating control room abandonment, it is not reasonable to assume that manual suppression will not be attempted. Treating this sensitivity in this manner makes it more consistent with the treatment of the human error and common cause failure sensitivities listed above. With this approach, the sensitivity of the categorization considers the impacts of application of reasonable ranges of issue treatments, rather than preemptively removing all credit. In the case of main control room abandonment, plant procedures and operator training make it highly unlikely that manual suppression would not be attempted.

Any Applicable Sensitivity Studies Identified in the Characterization of PRA Adequacy:

The discussion in this section is focused on the requirement to perform: "Any applicable sensitivity studies identified in the characterization of PRA adequacy". In order to determine which, if any, such studies are needed, it is first necessary to evaluate the model and application-specific sources of uncertainty.

Methodology for Identifying 10 CFR 50.69 Application Specific Sources of Uncertainty for FPRA:

The development of the Vogtle FPRA was guided by NUREG/CR-6850 (Reference 16). The VEGP FPRA model used either the consensus model described in NUREG/CR-6850 or, in one instance, an alternate consensus model that was found to be adequate to meet the fire PRA standard. The following discussion provides information on one instance in which an alternate consensus approach was used. The approach dealt with assigning severity factors, which, per the peer review team, was based on an unreviewed analysis method. Consequently, an F&O was assigned. The F&O related to an alternate approach stated:

> "The severity factors approach is based on an unreviewed analysis method. Currently in the FPRA there appears to be no credit for

suppression activities when the severity factor is applied. Therefore, the scenario quantification does not limit the severity of the fire from the perspective of suppression credit. This assessment is limited to the treatment of severity factors and non suppression probabilities at the time of the peer review."

The F&O resolution that reflects the alternate consensus approach states:

"The application of the method to the FPRA is consistent with the results of the recent industry review completed subsequent to the Peer Review. This method involves the application of a modified fire factor developed through a review of the individual industry fire events that form the basis for the generic fire frequency. However, the FPRA has applied a conservative value – higher than the value proposed in the method reviewed."

VEGP has used guidance provided in NUREG -1855 (Reference 13) to address uncertainties associated with Fire PRA for 10 CFR 50.69 application. As stated in Section 1.5 of NUREG -1855:

"Although the guidance does not currently address all sources of uncertainty, the guidance provided on the process for their identification and characterization and for how to factor the results into the decision making is generic and is independent of the specific source. Consequently, the process is applicable for other sources such as internal fire, external events, and low power and shutdown."

NUREG-1855 (Reference 13) describes an approach for addressing sources of model uncertainty and related assumptions. It defines "a source of model uncertainty is one that is related to an issue in which no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event)." NUREG-1855 defines a consensus model as "a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group. In addition, widely accepted PRA practices may be regarded as consensus models. Examples of the latter include the use of the constant probability of failure on demand model for standby components and the Poisson model for initiating events. For risk-informed regulatory decisions, the consensus model approach is one that NRC has utilized or accepted for the specific risk-informed application for which it is proposed."

<u>Disposition of 10 CFR 50.69 Application Specific Sources of Uncertainty for FPRA:</u>

The potential sources of model uncertainty in the Fire PRA model were characterized for the 16 tasks identified by NUREG/CR-6850 (Reference 16). This framework was used to organize the assessment of baseline FPRA

epistemic uncertainty and evaluate the impact of this uncertainty on 10 CFR 50.69.

Table 3 outlines sources of uncertainties by task and their disposition. As noted in the table, the VEGP FPRA was developed using methods outlined in the NUREG/CR-6850 (Reference 16), which is considered a consensus methodology. Furthermore, the use of unreviewed analysis method has been determined to be consensus approach. Therefore, consistent with NUREG-1855 (Reference 13), FPRA modeling does not introduce epistemic uncertainties to be addressed in the 10 CFR 50.69 application.

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
1	Analysis boundary and partitioning	This task establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on the discussion of sources of uncertainly, it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
2	Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized though use of the PWROG Generic Multiple Spurious Operation (MSO) list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
3	Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. The overall process is essentially the same as that used to perform the analyses to demonstrate compliance with 10 CFR 50.48.	Based on the discussion of sources of uncertainty it is concluded that the methodology for the Cable Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
4	Qualitative Screening	Qualitative screening was performed; however, some structures (locations) were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the fire PRA (based on industry guidance and criteria) were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip. However, such locations would not contain any features (equipment or cables identified in the prior two tasks) and consequently are expected to have low risk contribution.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a CCDP commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
5	Fire-Induced Risk Model	The internal events PRA model was updated to add fire specific initiating event structure as well as additional system logic. The methodology processes used are consistent with that used for the internal events PRA model development as was subjected to industry Peer Review. The developed model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed all significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
		PRA model includes structure to translate them into the appropriate induced initiator.	not have an impact on 10 CFR 50.69 application.
6	Fire Ignition Frequency	Fire ignition frequency is an area with inherent uncertainty. Part of this uncertainty arises due to the counting and related partitioning methodology. However, the resulting frequency is not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the industry generic frequency values used for the FPRA. This is because there is no specific treatment for variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates. The applied fire frequency values are believed to be over-estimated.	Industry generic frequency values were used when developing VEGP FPRA. Based on the discussion of sources of uncertainty, it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
7	Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	The Vogtle FPRA development did not screen out any fire initiating events based on low CDF/LERF contribution. Screening of individual fire ignition sources occurred only if it involved a discrete component and the consequences of the associated fire did not involve failure of any other plant component or feature. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
			Quantitative Screening task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
8	Scoping Fire Modeling	The framework of NUREG/CR-6850 includes two tasks related to fire scenario development. These two tasks are 8 and 11. The discussion of uncertainty for both tasks is provided in the discussion for Task 11.	See Task 11 discussion.
9	Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
			not have an impact on 10 CFR 50.69 application.
10	Circuit Failure Mode Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability is assigned using industry guidance such as that published in NUREG/CR-6850. The uncertainty associated with the applied conditional failure probabilities poses competing considerations. On the one hand, a failure probability for spurious operation could be applied based solely on cable scope without consideration of less direct fire affects (e.g., a 0.3 failure likelihood applied to the spurious operation of an MOV without consideration of the fire-induced generation of spurious signal to close or open the MOV). The analysis has biased the treatment such that it is assumed the spurious signal will always drive the valve in the unsafe direction. In addition, for those valves that might have multiple desired functions – consideration of spurious closure and consideration of failure to open on demand, the non-spurious failure state is treated with a logical TRUE rather the complement of the spurious probability. For those valves that only have an active function, the potential for a spurious signal to drive the valve in the desired direction is ignored. The treatment results in skewing of the results	Uncertainty in the circuit failure mode likelihood analysis could lead to assumed failures of related components and related system functions. This would generate conservative results and that would typically be acceptable for most applications. Furthermore, consensus modeling approach is used for Circuit Failure Mode Likelihood Analysis. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
		such that the resulting risk is over-estimated.	
11	Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and the response of plant staff (detection, fire control, and fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of	Consensus modeling approach is used for the Detailed Fire Modeling. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
		that guidance retains that conservatism. The resulting risk estimates are also conservative.	
12	Post-Fire Human Reliability Analysis	The human error probabilities used in the FPRA were adjusted to consider the additional challenges that may be present given a fire. The human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The human error probabilities were obtained using the EPRI HRAC and included the consideration of degradation or loss of necessary cues due to fire. The impact of any remaining uncertainties is expected to be small. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
13	Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified fire PRA model. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
			does not have an impact on 10 CFR 50.69 application.
14	Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit. However, the selected truncation was confirmed to be consistent with the requirements of the PRA Standard.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would require sensitivity treatment Therefore, this task does not have an impact on 10 CFR 50.69 application.
15	Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.
16	Fire PRA	This task does not introduce any new uncertainties	This task does not introduce any new

Table 3: Fire PRA Uncertainty and Sensitivity Matrix			
Task Number	**Description**	**Sources of Uncertainty**	**Disposition**
	Documentation	to the fire risk.	uncertainties to the fire risk as it outlines documentation requirements. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Fire PRA Documentation task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this task does not have an impact on 10 CFR 50.69 application.

3.2.3 Other Techniques

The NRC endorsed NEI-00-04 methodology allows for categorization of SSCs to be performed in the absence of quantifiable PRA models for other risk hazards, specifically seismic hazards, other external events hazards (e.g., tornados, high winds, transportation, etc.), and shutdown hazards. The following information provides a historic perspective of how these risks were analyzed in the past and how the results will be used when categorizing SSCs per NEI 00-04.

External hazards were evaluated in the VEGP Individual Plant Examination of External Events (IPEEE) submitted in response to the NRC IPEEE Generic Letter 88-20, Supplement 4 (Reference 9). The IPEEE was a one-time review of external hazard risk and was limited in its purpose to the identification of potential plant vulnerabilities and the understanding of associated severe accident risks. The results of the VEGP IPEEE study are documented in the VEGP IPEEE main report.

In the VEGP IPEEE, the seismic risk evaluation was performed in accordance with the EPRI Seismic Margins Analysis (SMA) methodology (References 17, 18, and 19). The SMA Review Level Earthquake (RLE) for VEGP was a 0.3g Peak Ground Acceleration (PGA) NUREG/CR-0098 (Reference 20) spectrum. VEGP structures and equipment were designed for a safe shutdown earthquake (SSE) defined by a Regulatory Guide 1.60 spectrum tied to a PGA of 0.2 g. However, due to conservatism applied to the demand and/or evaluation techniques, most of the seismic Category I structures and equipment were designed and qualified for a 0.3g PGA capacity.

Because the SMA approach was used, there are no CDF and LERF models or values available from the seismic analysis in the VEGP IPEEE. One of the insights from the VEGP SMA was that VEGP is one of the most seismically rugged nuclear power plants. A conclusion from the SMA was that VEGP has a high-confidence-low-probability-of-failure (HCLPF) capacity of at least 0.3g PGA.

The VEGP IPEEE analysis of High winds, Floods, and Other (HFO) external hazards was accomplished by using a progressive screening approach described in NUREG-1407 (Reference 21). The VEGP IPEEE concluded that the existing VEGP design was in conformance with the 1975 Standard Review Plan (SRP), NUREG 75-087 (Reference 22), criteria, in all reviewed areas and no potential vulnerabilities were identified. HFO events were screened out by compliance with the SRP. As such these hazards were determined to be negligible contributors to the overall plant risk.

As stated earlier, the NEI 00-04 methodology allows for categorization of SSCs to be performed in the absence of quantifiable PRA models for the following other risk hazards - seismic hazards, other external events hazards (e.g., tornados, high winds, transportation, etc.), and low power/shutdown hazards. The following information provides how each of these risks will be evaluated when categorizing SSCs per NEI 00-04.

3.2.3.1 Updated Seismic Margin Analysis SSEL

Currently, a state-of-the-art VEGP Seismic PRA model is being developed that will address the requirements of the ASME/ANS PRA standard (Reference 11). It is anticipated that the model will be available for use to categorize SSCs in early 2015. As such, the use of a seismic PRA model is not proposed as part of the VEGP 10 CFR 50.69 process at this time.

In the absence of a seismic PRA model that meets the seismic PRA standard, the NEI 00-04 methodology allows the use of the seismic margin analyses, which is a screening approach to evaluating seismic hazards. It does not generate core damage values; rather, it functions to identify potential seismic susceptibilities and vulnerabilities.

To respond to the NRC IPEEE Generic Letter 88-20, Supplement 4, VEGP chose to address IPEEE-Seismic by performing an SMA per the EPRI SMA methodology (Reference 17). In NUREG-1407 (Reference 10), the NRC states that the EPRI SMA methodology is an acceptable methodology for resolution of IPEEE-Seismic. Per the EPRI methodology, one preferred and one alternate path capable of achieving and maintaining a safe shutdown condition for at least 72 hours following a seismic margin earthquake (SME) were selected for each unit.

The SMA included the development of seismic Safe Shutdown Equipment Lists (SSELs) and the performance of seismic capability walkdowns and evaluations of the SSEL components. In 2011 and 2012, SNC reviewed the IPEEE SSELs against the current plant configuration, updated SSELs accordingly, and performed seismic capability walkdowns and evaluations on accessible new and modified SSEL components in accordance with EPRI NP-6041-SL (Reference 17). The primary remaining work consists of evaluating inaccessible new and modified SSEL components and resolving open items.

In the updated seismic SSEL, the review level earthquake (RLE) for VEGP was the same as that was used in IPEEE, which was a 0.3g peak ground acceleration (PGA) NUREG/CR-0098 (reference 20) spectrum. VEGP structures and equipment were designed for a safe shutdown earthquake (SSE) defined by a RG 1.60 (Reference 18) spectrum tied to a PGA of 0.2g. However, due to conservatism applied to the demand and/or evaluation techniques, most of the seismic Category I structures and equipment were designed and qualified for a 0.3g PGA capacity. One of the insights from the VEGP IPEEE SMA was that VEGP is one of the most seismically rugged nuclear power plants. A conclusion from the SMA was that VEGP has a high-confidence-low-probability-of-failure (HCLPF) capacity of at least 0.3g PGA. These insight and conclusion also apply to the SMA containing updated seismic SSEL.

The SMA identified the seismic design basis and severe accident functions of components. A determination was made to find out if a component would be needed during and after the seismic event. A seismic SSEL was developed containing components that would be needed during and after the seismic event. When categorizing SSCs per 10 CFR 50.69, a component that is credited as part

of a seismic-margins-evaluated safe shutdown path will be considered safety-significant from a seismic risk perspective. The attributes, which yielded that conclusion, will be identified. If the component does not participate in the seismic safe shutdown path, then it is considered a candidate low safety-significant with respect to seismic risk.

3.2.3.2 Updated IPEEE Screening (other external events)

When the IPEEE analysis was performed, the ASME/ANS standards did not exist. However, ASME/ANS standards have since been developed for analyzing various hazards. The current ASME/ANS Combined PRA standard (RA-Sa-2009) (Reference 11) has guidance for other external events. As cited in RA-Sa-2009, other external events include all external events other than seismic events. In 2011, SNC evaluated other external risks, including the impact of construction activities at Vogtle Units 3 and 4, using the criteria cited in RA-Sa-2009. The conclusion of this evaluation is that all other external events screened out. Therefore, per RA-Sa-2009, SNC is not required to develop VEGP PRA models for other external events. The results of the screening indicate that all SSCs will be treated as candidate low safety significant with respect to other external events risk.

3.2.3.3 Low Power/Shutdown

Currently, a consensus Low Power (LP) PRA standard is not available. Consequently, VEGP does not have a LP PRA model. A typical LP PRA methodology would rely extensively on the extension of at-power analysis methods to LP conditions. Accordingly, the at-power methodology employed for higher power operations are typically adapted or extended to address LP conditions. The at-power PRA model is generally valid for evaluating LP conditions. When accounting for failure data, VEGP does not differentiate between at-power conditions versus low power conditions. Therefore, in order to account for LP risks, insights can be obtained using the at-power PRA model for those cases where at-power PRA model is valid for evaluating low power conditions. In addition, in those cases where at-power PRA risk model is not appropriate for use at low power conditions, the low power risk will be evaluated by qualitatively assessing the function(s) performed by the system as outlined in NEI 00-04 section 9.2.2 review of risk information item 6. A component is assigned a preliminary risk, which is higher of these two risks where applicable.

Currently, a consensus Shutdown (SD) PRA standard is not available, and consequently, VEGP does not have a SD PRA model. In the absence of SD PRA, NEI 00-04 methodology permits use of qualitative analysis. It is recognized that the categorization process using a qualitative approach is more conservative (i.e., designed to identify more SSCs as safety-significant) than using a plant-specific SD PRA. This is due to the fact that the qualitative approach provides safety function defense-in-depth without regard to the likelihood of demand or reliability of the functions credited. The NEI 00-04 approach identifies all SSCs necessary to support primary shutdown safety systems as safety-significant. This

measure of safety significance assures that the SSCs that were required to maintain low shutdown risk are retained as safety-significant. A typical SD PRA would credit all of the same SSCs in a probabilistic framework, so some may avoid being identified as safety-significant using the PRA, but the shutdown safety management plan approach identifies them as safety-significant regardless of the frequency of challenge or level of functional diversity.

The NEI 00-04 methodology uses a shutdown safety assessment process per NUMARC 91-06. VEGP updated its Shutdown Safety Program developed per NUMARC 91-06 (Reference 12) in early 2011. As a result, VEGP has a qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM). The development of the model is based on the framework for and guidance for DID provided in NUMARC 91-06, which provides guidance for assessing and enhancing safety during shutdown operations. Other documents considered in this model development include INPO 06-08 (Reference 23), SOER 09-01 (Reference 24), and NRC Inspection Manual Chapter 0609, Appendix G (Reference 25). This qualitative model will be used as follows. It is recognized that the use of shutdown DID model introduces an additional deterministic conservatism.

1. A component can be identified as safety-significant for shutdown conditions for either of the following reasons:

    - NUMARC 91-06 specifies that a defense in depth approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and alternative system/train to accomplish the given key safety function. When multiple systems/trains are available to satisfy the key safety function, only SSCs that support the primary and first alternative methods to satisfy the key safety function are considered to be the "primary shutdown safety system" and are thus candidate safety-significant.

    - Its failure would initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.).

2. If the component does not participate in either of these manners, then it is considered a candidate as low safety significance with respect to shutdown safety.

Note that in this assessment, a primary shutdown safety system refers to a system that has the following attributes:

- It has a technical basis for its ability to perform the function.
- It has margin to fulfill the safety function.
- It does not require extensive manual manipulation to fulfill its safety function.

3.3 PRA Review Process Results (10 CFR 50.69(b)(2)(iii))

NEI 00-04 (Reference 4) requires that the internal events (including internal flooding) PRA and Fire PRA models be reviewed to the guidance of Regulatory Guide 1.200 to ensure that PRA models meet Capability Category (CC) II for the supporting requirements of the ASME PRA standard. NEI 00-04 also requires that deviations from CC II relative to the 10 CFR 50.69 program be justified and documented. This section provides PRA review process results for the internal events (including internal flooding) PRA and Fire PRA models.

3.3.1 Internal Events (including Internal Flooding) PRA Model

The VEGP PRA has been subjected to a number of peer reviews and self-assessments, including one performed in accordance with the 2007 version of the PRA Standard (Reference 26) as clarified by RG 1.200, Revision 1 (Reference 27). The information provided in this section demonstrates that the VEGP internal events PRA model (including flooding) meets the requirements of RG 1.200, Revision 2 (Reference 28).

Section 3.3.1.1 summarizes prior peer reviews and self-assessments for the VEGP PRA model.

Section 3.3.1.2 describes the overall results of the RG 1.200 peer review (Capability Category and Findings) performed in 2009.

Section 3.3.1.3 summarizes the resolution of Findings and Observations (F&Os) identified in the RG 1.200 peer review.

In May 2009, the VEGP PRA was reviewed against the 2007 version of the PRA Standard as amended by RG 1.200, Revision 1 (Reference 27) and not the latest version of the PRA Standard (Reference 11) issued in 2009 as clarified by RG 1.200, Revision 2 (Reference 28). Therefore, Section 3.3.1.4 provides a list of additional requirements associated with the latest PRA Standard. This section also describes the VEGP PRA model and model documentation revisions that assure consistency with the latest version of the PRA Standard and RG 1.200.

3.3.1.1 Previous Peer Review and Self Assessment for VEGP PRA Model

In addition to the independent internal and external review during each VEGP PRA model development and update, several assessments of the technical capability were made prior to the Pressurized Water Reactor Owners Group (PWROG) peer review against the ASME PRA Standard and RG 1.200, Revision 1 in May 2009. Listed below are the previous assessments of the VEGP PRA.

- An independent PRA peer review was conducted under the auspices of the Westinghouse Owners Group (WOG) in December 2001, following the Industry PRA Peer Review process (Reference 29). This peer review included an assessment of the PRA model maintenance and update process.

This assessment did not identify any "A" F&Os. All "B" F&Os from the 2001 Industry PRA Peer Review for VEGP PRA were addressed in VEGP PRA model Revision 3.

- During 2005, the VEGP PRA model results were evaluated in the WOG PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process. Results of this cross-comparison are presented in WCAP-16464, Westinghouse Owner's Group Mitigating Systems Performance Index Cross Comparison. The PRA Cross Comparison Candidate Outlier Status was described in section 3.4 of VEGP MSPI base document. Noted in this document was the fact that, after allowing for plant-specific features, there were no MSPI cross-comparison outliers for VEGP PRA.

- In 2006, a gap analysis was performed against the available versions of the ASME PRA Standard (Reference 30) and Regulatory Guide 1.200, Revision 0 (2003 trial version). Documentation (especially system notebooks), issues related to the internal flooding PRA, and treatment of uncertainty correlations were identified as major gaps. The identified gaps were resolved in VEGP PRA model revision 4 in 2009.

- In 2008, the VEGP PRA model (draft Revision 4) was benchmarked with three Westinghouse PWRs (Comanche Peak, Callaway, Wolf Creek) as a part of an MSPI margin study. The benchmarking concluded that there were no significant issues in the VEGP PRA model which would impact MSPI calculations.

3.3.1.2 Industry PRA Peer Review for VEGP Internal Events (including Internal Flooding) PRA Model

The VEGP PRA model for internal events (including internal flooding) at-power was updated to Revision 4 early in 2009 to close the gaps from the 2006 self assessment, to meet the ASME PRA standard supporting requirements (including NRC clarifications as stated in RG 1.200, Revision 1), and to represent the as-built as-operated plant.

In May 2009, the VEGP PRA internal events model Revision 4 (including internal flooding) was reviewed against the requirements of the 2007 version of the PRA Standard (Reference 26), as amended by RG 1.200, Revision 1 (Reference 27). A summary of this peer review is provided below:

1. The 2007 version of ASME PRA Standard contains a total of 327 numbered supporting requirements (SRs) in nine technical elements and the configuration control element. Eleven of the SRs represent deleted requirements (IE-A8, IE-A9, SC-A3, SY-A9, SY-B9, HR-G8, IF-A2, IF-B4, IF-D2, IF-E2, and QU-D2) and 20 were determined by the peer review team to be not applicable to the VEGP PRA. Thus, a total of 296 SRs were applicable.

2. Among 296 applicable SRs, 99% of SRs met Capability Category II or higher as follows:

Table 4: Summary of VEGP Internal Event (including Internal Flooding) PRA Capability Categories		
Capability Category Met	**Number of SRs**	**% of Total Applicable SRs**
CC-I/II/III (or SR Met)	210	70.9%
CC I	0	0%
CC II	38	12.8%
CC III	7	2.4%
CC I/II	14	4.7%
CC II/III	24	8.1%
SR Not Met	3	1.0%
SR (CC-I/II/III) Met	296	100

3. Three SRs were judged to be not met. These are HR-G6, QU-D3, and LE-G5.

   - SR HR-G6 was not met because the reasonableness check of Human Reliability Analyses (HRA) was done for the previous revision of the PRA model and not the latest revision.

   - SR QU-D3 was not met because the SR requires the PRA results to be compared with those from similar plants. The VEGP PRA report cites the MSPI benchmark report as evidence of meeting this requirement, but the peer reviewers viewed this as an outdated comparison.

   - SR LE-G5 was characterized as "Not Met" because the limitation of the LERF calculations that could impact risk-informed applications was not identified.

4. The peer review generated 11 Findings. These Findings and their resolutions are described in section 3.3.1.3. Resolution of Findings HR-G6-01, QU-D3-01, and LE-G5-01 resulted in SRs HR-G6, QU-D3, and LE-G5 being met to a Capability Category I/II/III. Thus, the VEGP internal events PRA (including flood) model meets the requirements of RG 1.200.

3.3.1.3 Resolution of Findings from VEGP PRA Peer Review

Table 5 documents the VEGP Internal Events (including Internal Flooding) PRA model peer review F&Os and their resolutions. As shown in Table 5, the three "Not Met" SRs have since been resolved.

Table 5 reflects two types of F&O - those that are already resolved and those, as indicated in the table, that would be resolved prior to implementation of the 10 CFR 50.69 program. As indicated in Table 5, the unresolved F&O do not

involve any Not Met SRs and mostly are related to enhancing documentation and are not expected to impact 10 CFR 50.69 categorization results.

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
IE-A4-01	IE-A4	CC-II Met	The SR requires a systematic evaluation of each system to assess the possibility of an initiating event occurring due to failure of the system. The reviewers could not find documentation of such a systematic review. Additional notes made by review team in response to SNC's comments: When the reviewers asked for the Initiating Events (IE) notebook (NB), they were told that Chapter 2 of the main report is the IE NB. Chapter 2 does not contain any evidence that a systematic evaluation of each system was performed. Nor does Chapter 2 contain a system failure modes and effects analysis (FMEA) as required by the Standard which would have been an acceptable alternate. The fact that a systematic evaluation was performed during the Individual Plant Examination (IPE), in of itself, is not sufficient. The evaluation performed for the IPE should have been reviewed and a statement to that extent should have been presented in the Chapter 2. In absence of such evidence, the review comment stays. As noted elsewhere in the report, it is very important to have good documentation.	This F&O will be resolved prior to the implementation of the 10 CFR 50.69 Program when the Vogtle internal events PRA notebook documentation is enhanced to describe the systematic evaluation performed for identifying special initiating events. A systematic evaluation for initiating events was performed during the Vogtle IPE and the baseline Vogtle internal events PRA model credits this process. The internal event model documentation provides a block diagram describing the IE identification task..

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
IE-D1-01	IE-D1	CC-I/II/III Met	The lack of a central place for all the information related to initiating events made it difficult for the review team to review this topic. Most plants have all this information stored in a separate IE NB. The review team recommends that VEGP do the same. Additional notes in response to SNC's comments: The review team disagrees with SNC's comments. The Standard requires that the work be documented in a manner that facilitates PRA applications, upgrades, and peer review. The review team does not believe that the work was documented a manner to facilitate peer review. One could almost make a case for 'not met' categorization for this element as the documentation is the weakest link in this whole effort. The F&O stays as written.	This F&O will be resolved when SNC develops a separate IE notebook prior to the implementation of 10 CFR 50.69 program.

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
AS-A11-01	AS-A11	CC-I/II/III Met	Dependencies are not preserved for consequential anticipated transient without SCRAM (ATWS) for the small loss of coolant accident (SLOCA) initiator and the steam generator tube rupture (SGTR) initiator. The existing ATWS trees, based on a LOFW initiator, were developed for transients that do not include a loss of reactor coolant system (RCS) inventory or operator actions to mitigate a SGTR. Note: The review team decided to leave the F&O as is after reviewing SNC's comments.	This F&O will be resolved prior to the implementation of 10 CFR 50.69 when SNC modifies the PRA Calculation to note that the ATWS event tree was developed to include consideration on loss of RCS inventory (LOCA) and SGTR. This is accomplished as documented in the internal events PRA calculation (Reference 14) and by incorporating logic gates into the system logic models to address primary integrity and system failures caused by SLOCA- or SGTR-initiated ATWS events.

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
SY-B3-01	SY-B3	CC-I/II/III Met	The treatment of main or frontline system and supporting or mitigating system Common Cause Failure (CCF) event groupings do not appear to consistent in the current PRA documents. It appears that for systems considered non-risk significant, reviews for CCF groups may not have been undertaken since the IPE modeling. The updated standards require a systematic treatment of all systems, not just the main systems contributing to core damage. New CCF groups may be required or updated documentation as to why these groups are not required is needed. Note: The review team decided to leave the F&O as is after reviewing SNC's comments.	This F&O will be resolved prior to the implementation of 10 CFR 50.69 when SNC develops updated CCF documentation reflecting that all potential CCF groups have been considered and documented. The systems that may be lacking CCF grouping are non-risk significant systems; therefore, resolution of this F&O is not expected to impact the 10 CFR 50.69 categorization results.
HR-G6-01	HR-G6	CC-I/II/III Not Met	Check of consistency and review for reasonableness is missing in the Revision 4 updated HRA draft and the prior revision document information related to these items is not appropriate to use in light of the updates performed and changes to the results. Section 8 includes a table of human failure events (HFEs) and human error probabilities (HEPs) but does not include HEP reasonableness check, as is documented in Section 8.3 of the November 2005 HRA update for Revision 3.	This F&O is resolved. Reasonableness check for all HRAs for Revision 4 model was re-performed. All HRAs have been determined to be reasonable or have been appropriately revised. The reasonableness check is documented in Section 8.2.2 of the internal events PRA calculation (Reference 14).

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
DA-C2-01	DA-C2	CC-I/II/III Met	Generic data alone was used for the probability that a power-operated relief valve (PORV) is blocked -- refer to Table 6.3.9. Since PORV availability is a critical plant feature with respect to ATWS pressure control, the use of generic data for this parameter is deemed a weakness.	This F&O will be resolved prior to the implementation of 10 CFR 50.69 program when SNC revises Table 6.3-9 of the PRA Calculation (Reference 14) to remove the entry for PORV leakage (POLRC) since a plant-specific value has been used.
IF-C2a-01	IF-C2a	CC-I/II/III Met	Because of a lack of well documented analysis, a lot of information had to be obtained by talking to the analyst who performed the analysis, which is the basis for the F&O. Original F&O: From a more detailed review of the IPE flood calculations (which are the main input to defining the flood events and consequences), it is noted that successful operator mitigation of ALL flood events is assumed to occur 30 minutes into any flood scenario and fully terminate the flood flow, and it appears to be based on assumptions only, as no detailed discussion of the actual ability of operators to perform such actions is given. This appears to be in direct conflict with the HFEs included (but not modeled in the PRA model) in the flooding report (assumed perfect response vs. HFE calculation). Also, the report lists hundreds of pages of a detailed analysis approach using screening criteria, flow calculations, etc. and only	This F&O will be resolved prior to the implementation of 10 CFR 50.69 program to clarify documentation related to flooding analysis. The VEGP internal events flooding analysis conservatively assumes equipment in a room is damaged due to flood when a pipe break occurs in that room. The analysis does not credit operator actions for flood isolation/mitigation (that is, screening HEP values used were equal to 1.0). Screening HEP values in human induced flooding events do not make use of the results of the design related calculations which assumes a 30 minutes flow termination time. As a result resolution of this F&O has no impact on the 10 CFR 50.69 categorization.

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			by locating very specific statements. There appear to be conflicts of the inputs to the flooding PRA and the subsequent discussions of operator mitigation as well as using the information from the IPE calculations for propagation assessments. This is more than an editorial finding and impacts the entire basis of using the older calculated results in the current analysis. Additional notes made in response to SNC's comments: The lengthy flooding methodology outlined in the report is not used in the current VEGP flooding results as mentioned in the original F&O. The previous IPE flooding analysis is used as inputs to the flooding targets and propagations for a bounding case estimation, and the more thorough analysis outlined in the report is not undertaken but is in place for use (as was explained by the SNC analyst in charge of the flooding project during the peer review and very briefly mentioned in the flooding report). Each and every IPE flooding calculation reviewed during the peer review contained the assumption	

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			that the flood water flow was successfully isolated at 30 minutes, and all calculations for flood volumes, propagations, etc. were done with the amount of water generated in this 30 minutes with considerations for system characteristics. If any IPE flooding calculations are done, which do not contain this assumption, they were not seen during the peer review. No change to the F&O is warranted. The problem with this scenario of using the IPE flooding calculations for inputs to the described methodology is the following: If the flooding analysis was performed in accordance to the methodology outlined in the current report, new flooding volumes and propagation assessments would be required that did not take into account successful isolation at 30 minutes (as the IPE calculations do) since another operator isolation assessment is outlined in the flooding report methodology for normal HFE calculations for flow isolation.	
QU-D3-01	QU-D3	CC-I/II/III Not Met	Reviewer asked the VEGP staff to provide evidence of comparison of the VEGP results to those from similar plants. The VEGP staff presented the benchmark report for MSPI as	This F&O is resolved. A new comparison study was performed by comparing VEGP PRA results with two PWR

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			evidence of comparison. Reviewers concluded that report is not sufficient evidence for demonstrating compliance to this SR.	PRAs (Callaway and Wolf Creek), which are considered relatively similar to VEGP. In addition to the comparison of PRA reports, a plant visit to Callaway was performed to identify more details of Callaway systems and PRA modeling. The comparison showed that all three plants have loss of offsite power (LOSP)/station blackout (SBO) as the most dominant contributors which indicated that the VEGP PRA results are not an outlier, as compared to similar PWRs. Differences in dominant CDF contributors were investigated, and it was found that those differences are due to differences in details of system configuration/operation and physical barriers for internal flooding and in the sources for generic initiating event frequency data (VEGP PRA used the latest generic initiating frequency and failure data along with VEGP specific experience data for its data update).

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
QU-F5-01	QU-F5	CC-I/II/III Met	In Chapter 10, there is insufficient documentation for the quantification process, which would impact application (only EOOS). Reviews conclude that the documentation currently in Chapter 10 is not sufficient to meet this SR fully.	This F&O will be resolved prior to the implementation of 10 CFR 50.69 program when SNC revises Chapter 10 to provide additional documentation of the quantification process, including additional limitations when using the PRA model for EOOS and other applications, and verifies that all quantification supporting requirements are met and documented. Section 10.3 of the internal events PRA calculation (Reference 14) discusses a few limitations in the quantification process that would impact applications, including the use of an average configuration, average test and maintenance unavailabilities, and point estimate mean data values.

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
LE-G5-01	LE-G5	CC-I/II/III Not Met	Limitations in the LERF analysis that would impact applications are not identified. The LERF analysis documentation is incomplete because limitations in the LERF analysis that would impact applications, as required by SR LE-G5, are not identified.	This F&O is resolved. A comparison of Vogtle LERF scenarios with those in Table 4.5.9.3 of the ASME PRA standard revealed that the Vogtle PRA included more potential LERF scenarios than as required for a large dry containment plant in ASME PRA standard. The LERF scenarios modeled in VEGP PRA include containment bypass core damage scenarios (steam generator tube rupture and Interfacing systems LOCA), thermally or pressure induced steam generator tube rupture after core damage, containment isolation failure with core damage, and various early containment failure modes.

Table 5: VEGP Internal Events (including Internal Flooding) Resolution of the Peer Review Findings and Observations (F&Os)				
F&O Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
MU-B4-01	MU-B4	CC-I/II/III Met	The VEGP plant procedures do not specifically call for a peer review after a PRA upgrade has been completed. But the plant has had this peer review and other peer reviews in the past. This change is required by the SR.	This F&O is resolved. Procedure(s) outlining requirements dealing with PRA configuration control, as referenced in ASME/ANS RA-Sa-2009, Section 1-5 (Reference 11), have been developed to comply with requirements of 10 CFR 50.69. The revision includes a reference for peer review update after a PRA upgrade has been completed. The PRA model update process is discussed in Section 3.2.1.1 of this licensing submittal.

3.3.1.4 Comparison of RG 1.200, Revision 1 and Revision 2 Internal Events (including Internal Flooding) PRA Requirements

The VEGP PRA model was reviewed against the 2007 version of the PRA Standard (Reference 26) as amended by RG 1.200, Revision 1 (Reference 27). The RG 1.200 Revision 2 (Reference 28) was issued in March 2009. So it would be prudent to review VEGP PRA to the guidance of RG 1.200, Revision 2.

To ensure compliance with any new or changed RG 1.200 requirements, it is necessary to first identify the differences between the RG 1.200 Revision 1 and Revision 2 Capability Category I/II, II/III, and I/II/III requirements. A summary of the differences in these requirements is provided in the following Table 6 along with a response for each of the differences. Note that differences that were considered typographical, editorial, or provided additional descriptions of the SRs were not considered technically significant and were excluded from Table 6. Among these is the exclusion of adjectives, such as the term "key", because no model document can contain all assumptions, only those that are considered significant enough to merit mention.

The review concluded that the VEGP internal events (including internal flooding) PRA model meets RG 1.200, Revision 2.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
IE-C10: CC-I/II/III: ... An example of an acceptable generic data sources is NUREG/CR-5750 [Note 1].	IE-C12: CC-I/II/III: ... An example of an acceptable generic data sources is NUREG/CR-6928 [Note 1].	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR cites a more recent example of an acceptable generic data source.	NUREG/CR-6928 is used as the source for generic data priors in Revision 4 of the VEGP internal events PRA.
SY-B15: CC-I/II/III: ... (h) harsh environments induced by containment venting, **or** failure that may occur prior to the onset of core damage.	SY-B14: CC-I/II/III: ... (h) harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR explicitly requires consideration of containment venting ducts and failure of the containment boundary prior to core damage.	As noted in Table 9.2-1 of the internal events PRA calculation (Reference 14), failure of the containment boundary due to venting is not an applicable to the VEGP large, dry, subatmospheric containment.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
DA-C1: CC-I/II/III: … Examples of parameter estimates and associated sources include: *(a)* component failure rates and probabilities: NUREG/CR-4639 [Note (1)], NUREG/CR-4550 [Note (2)], NUREG-1715 [Note 7]	DA-C1: CC-I/II/III: … Examples of parameter estimates and associated sources include *(a)* component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20]	Reference NUREG-1715 was added by RG 1.200 Revision 1; References NUREG-1715 and NUREG/CR-6928 were included in the 2009 version of the PRA Standard. The updated SR cites more recent examples of acceptable generic data sources.	NUREG/CR-6928 is used as the source for generic data priors in Revision 4 of the VEGP internal events PRA.
QU-A2a: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF …	QU-A2: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF (and LERF) …	The LERF requirement was added by RG 1.200 Revision 2. The updated SR explicitly requires consideration of LERF.	Section 10.3.2 of the internal events PRA calculation (Reference 14) presents estimates for individual LERF sequence cutsets.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
QU-A2b: CC-I: ESTIMATE the point estimate CDF from internal events. CC-II: ESTIMATE the mean CDF from internal events, accounting for the "state-of-knowledge" correlation between event probabilities [Note (1)]. CC-III: CALCULATE the mean CDF from internal events by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.	QU-A3: CC-I: ESTIMATE the point estimate CDF (and LERF). CC-II: ESTIMATE the mean CDF (and LERF), accounting for the state-of-knowledge correlation between event probabilities [Note (1)]. CC-III: CALCULATE the mean CDF (and LERF) by propagating the uncertainty distributions, ensuring that the state-of-knowledge correlation between event probabilities is taken into account.	The phrase, "from internal events", was deleted from the 2009 version of the PRA Standard. The LERF requirement was added by RG 1.200 Revision 2. The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements.	The peer review based on the 2007 version of the PRA Standard (Reference 26) addressed these LERF requirements. Section 10.3.2 of the internal events PRA calculation (Reference 14) presents the mean CDF LERF results.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
QU-B6: CC-I/II/III: ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees.	QU-B6: CC-I/II/III: ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF or LERF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees.	The LERF requirement was added by RG 1.200 Revision 2. The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements.	The peer review based on the 2007 version of the PRA Standard (Reference 26) addressed these LERF requirements. The Level 2 PRA event trees presented in Section 9.2 of the internal events PRA calculation (Reference 14) explicitly account for system successes.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
QU-E3: CC-I: ESTIMATE the uncertainty interval of the CDF results. **Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).** CC-II: ESTIMATE the uncertainty interval of the CDF results. **ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** CC-III: **PROPAGATE parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)**....(no change)	QU-E3: CC-I: ESTIMATE the uncertainty interval of the CDF (and LERF) results. **Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).** CC-II: ESTIMATE the uncertainty interval of the CDF (and LERF) results. **ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** CC-III: **PROPAGATE parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)**....(no change)	The LERF requirement was added by RG 1.200 Revision 2. The SR explicitly requires consideration of LERF. However, per the Note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements.	The peer review based on the 2007 version of the PRA Standard (Reference 26) addressed these LERF requirements. Section 10.4 of the internal events PRA calculation (Reference 14) presents the uncertainty intervals for both CDF and LERF, with consideration of the state-of-knowledge correlation.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
QU-E4: CC-I: PROVIDE an assessment of the impact of the model uncertainties and assumptions on the results of the PRA. CC-II: EVALUATE the sensitivity of the results to model uncertainties and key assumptions using sensitivity analyses [Note (1)]. CC-III: EVALUATE the sensitivity of the results to uncertain model boundary conditions and other assumptions using sensitivity analyses except where such sources of uncertainty have been adequately treated in the quantitative uncertainty analysis [Note (1)].	QU-E4: CC-I/II/III: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).	Separate requirements for CC-I, II, and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard. The reference to Note 1 was deleted by RG 1.200 Revision 2. The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.	No action, CC-II met for 2007 version of the PRA Standard (Reference 14).

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
LE-F2: CC-I: PROVIDE a qualitative assessment of the key sources of uncertainty. Examples: *(a)* Identify bounding assumptions. *(b)* Identify conservative treatment of phenomena. CC-II: PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the significant contributors to LERF. CC-III: PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies.	LE-F3: CC-I/II/III: IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, in a manner consistent with the applicable requirements of Tables 2-2.7-2(d) and 2-2.7-2(e).	Separate requirements for CC-I, II, and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard. The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.	No action, CC-II met for 2007 version of the PRA Standard (Reference 14).

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
IF-F2: CC-I/II/III: DOCUMENT the process used to identify ... flood areas, ... For example, this documentation typically includes ... *(b)* flood areas used in the analysis and the reason for eliminating areas from further analysis	IFPP-B2: CC-I/II/III: DOCUMENT the process used to identify flood areas. For example, this documentation typically includes *(a)* flood areas used in the analysis and the reason for eliminating areas from further analysis *(b)* any walkdowns performed in support of the plant partitioning	The requirement to document walkdowns performed in support of plant partitioning was added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Section 5 and Appendix A of the internal flooding PRA (Reference 13) document the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
IF-B1: CC-I/II/III: For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems)	IFSO-A1: CC-I/II/III: For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems, and reactor coolant system) ...	The requirement to include the fire protection system in Item (a) as a potential flooding source was added by RG 1.200 Revision 1. This requirement was addressed in the peer review, which used the 2007 version of the PRA Standard amended by RG 1.200 Revision 1. The requirement to include the reactor coolant system in Item (a) as a potential flooding source was added to the 2009 version of the PRA Standard. Thus, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	Potential flood sources identified in Section 5 of the internal flooding PRA reviewed as part of 2009 peer review against 2007 version of the PRA standard amended by RG 1.200, Revision 1 (Reference 27) include RCS-connected systems - chemical and volume control system (CVCS), containment spray (CS), residual heat removal (RHR), reactor coolant system drain tank (RCSDT), safety injection (SI), and reactor water makeup system (RMWS). As outlined in the Plant Vogtle Internal Flooding notebook, the Containment Building (and RCS components therein) is not included in the scope of the internal flooding analysis.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
IF-F2: CC-I/II/III: DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: *(a)* flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined ... *(f)* screening criteria used in the analysis ... *(j)* calculations or other analyses used to support or refine the flooding evaluation	IFSO-B2: CC-I/II/III: DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: *(a)* flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined *(b)* screening criteria used in the analysis *(c)* calculations or other analyses used to support or refine the flooding evaluation *(d)* any walkdowns performed in support of the identification or screening of flood sources	The requirement to document walkdowns performed in support of the identification or screening of flood sources was added to 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	The internal flooding PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
IF-F2: CC-I/II/III: DOCUMENT the process used to identify applicable flood scenarios. For example, this documentation typically includes: ... *(c)* propagation pathways ... ... *(d)* accident mitigating features and barriers credited ... ... *(e)* assumptions or calculations used in the determination of ... flood-induced effects on equipment operability ... *(f)* screening criteria used in the analysis *(g)* flooding scenarios considered, screened, and retained	IFSN-B2: CC-I/II/III: DOCUMENT the process used to identify applicable flood scenarios. For example, this documentation typically includes *(a)* propagation pathways ... *(b)* accident mitigating features and barriers credited ... *(c)* assumptions or calculations used in the determination of ... flood-induced effects on equipment operability *(d)* screening criteria used in the analysis *(e)* flooding scenarios considered, screened, and retained	The requirement to document walkdowns performed in support of the identification or screening of flood scenarios was added to 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood scenarios. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	The internal flooding PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
(h) description of how the internal event analysis models were modified ... ... *(j)* calculations or other analyses used to support or refine the flooding evaluation ...	*(f)* description of how the internal event analysis models were modified ... *(g)* calculations or other analyses used to support or refine the flooding evaluation *(h)* any walkdowns performed in support of the identification or screening of flood scenarios		
IF-F2: CC-I/II/III: DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes: ... *(j)* calculations or other analyses used to support or refine the flooding evaluation ...	IFQU-B2: CC-I/II/III: DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes: *(a)* calculations or other analyses used to support or refine the flooding evaluation	The requirement to document walkdowns performed in support of internal flood accident sequence quantification was added in 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood related features considered in flood sequence quantification. Since documentation of walkdowns was not in the 2007	The internal flooding PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

Table 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III			
SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	**SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2**	**Description of Change**	**Resolution**
(f) screening criteria used in the analysis ... *(i)* flooding scenarios considered, screened, and retained ... *(k)* results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D	*(b)* screening criteria used in the analysis *(c)* flooding scenarios considered, screened, and retained *(d)* results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D *(e)* any walkdowns performed in support of internal flood accident sequence quantification	version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard.	

3.3.1.5 General Conclusions Regarding PRA Capability

The information provided in this section demonstrates that the VEGP at-power internal events (including internal flooding) PRA model conforms to the standard at CC-II which satisfies the guidance of RG 1.200, Revision 2 (Reference 28). In addition, the VEGP PRA model complies with all requirements for technical adequacy of the baseline PRA as defined in NEI 00-04.

The VEGP internal events (including flooding) PRA model technical capability evaluations described above provide a robust basis for concluding that the PRA model is suitable for use in supporting the implementation of 10 CFR 50.69.

3.3.2 Fire PRA Model

The VEGP Fire PRA (FPRA) was reviewed in 2012 by the PWROG against Combined ASME/ANS RA-SA-2009 Standards (Reference 11), RG 1.200, Revision 2 (Reference 28), and NEI 07-12 (Reference 31). A self assessment of the VEGP FPRA against the ASME/ANS RA-SA-2009 was performed by SNC in-house staff prior to the PWROG peer review. The overall results of the PWROG peer review (Capability Category and Findings) are described in section 3.3.2.1.

Section 3.3.2.2 summarizes the resolution of F&Os identified in the PWROG peer review.

The PWROG peer review exit meeting concluded that the Vogtle Unit 1 FPRA was complete, that the CDF was within an acceptable range, and that the supporting documents were complete. Some refinements to address F&Os and to address documentation issues were identified. The PWROG peer review team stated that:

- FPRA overall quality was very good
- FPRA team was qualified and capable
- FPRA was found to be of high technical quality
- The ignition frequency assumptions, sources, conditions and corresponding actions were explicitly addressed. The depth and completeness of the uncertainty related discussion in the documents was outstanding
- Documentation was extensive
- HRA was very detailed and evaluation of fire procedures was addressed
- Documentation of Internal Events F&Os resolution against fire PRA requirements was good

The information provided in section 3.3.2 demonstrates that the VEGP FPRA model meets the requirements of RG 1.200, Revision 2.

3.3.2.1 Previous Peer Review and Self Assessment for VEGP FPRA Model

Before performing the PWROG peer review of FPRA against the Combined ASME/ANS RA-SA-2009 Standards (Reference 11), RG 1.200, Revision 2 (Reference 28), and NEI 07-12 (Reference 31) in 2012, a self-assessment of the VEGP FPRA model against the ASME/ANS RA-SA-2009 was performed by SNC in-house staff. The scope of the self-assessment included a review of the following FPRA tasks:

TASK 1 & 6:	Vogtle Fire PRA Plant Partitioning and Fire Ignition Frequency
TASK 2:	Component and Cable Selection
Task 3 & 9:	Cable Selection and Detailed Circuit Failure Analysis
TASK 5:	Fire Model Development
TASK 10:	Circuit Failure Mode and Likelihood Analysis
TASK 8 & 11:	Fire Scenario Selection
TASK 12:	Human Reliability Analysis
TASK 13:	Seismic-Fire Interactions Assessment
TASK 14 & 15:	Fire Risk Quantification

The results of the self assessment concluded that when compared against the SRs of the ASME/ANS RA-SA-2009 FPRA standard, all the SRs were classified as Met. This favorable outcome provided an assurance that the FPRA was ready for submittal to the PWROG peer review team.

3.3.2.2 Industry PRA Peer Review for VEGP Fire PRA Model

The ASME/ANS RA-SA-2009 version of the PRA Standard (Reference 11) contains a total of 173 numbered SRs in 13 technical elements. The configuration control element has 10 additional SRs. Thus, a total of 183 SRs were assessed.

Among 183 SRs, 25 were determined to be Not Applicable, resulting in a total of 158 SRs that were assessed. 13 of the 25 not applicable SRs were associated with the QLS and QNS Technical Elements, which were assessed as Not Reviewed. After the PWROG Peer Review, a focused-scope peer review was conducted for the QLS and QNS elements that were marked as Not Reviewed by the peer review team. The focused-scope peer review dispositioned the 7 SRs for QLS as Met and the 6 SRs for QNS as NA (not applicable). The focused-scoped peer review did not identify any new Findings.

Of the 158 total applicable SRs, approximately 80% met Capability Category II or higher, as shown in Table 7. It is noted that the results of the focused scope peer review are not included in the Table 7 statistics.

Table 7: Summary of VEGP Fire Events Capability Categories			
Capability Category Met	**Number of SRs**	**% of Total SRs**	**% of Total Applicable SRs**
Met	95	51.9%	60.1%
Not Met	31	16.9%	19.6%
CC I	5	2.7%	3.2%
CC II	7	3.8%	4.4%
CCIII	7	3.8%	4.4%
CC I/II	5	2.7%	3.2%
CC II/III	8	4.4%	5.1%
NA	25	13.7%	-
NR	0	-	-
Total	183	100.0%	100%

The peer review generated 50 Findings out of which 31 SRs were judged to be not met. These Findings and their resolutions are described in Section 3.3.2.3. It is noted that several Findings resulted in more than one SR being judged as not met. The resolution of the Findings results in two SRs, SC-C2-02 and PRM-B13-01, being not met. Resolution of these two Findings relates to enhancing documentation and has no impact on any technical element of the analysis. These SRs will be resolved prior to implementation of the VEGP 10 CFR 50.69 Program. All SRs (including the two not met when resolved) are being met at CC II or better, with the exception of FSS-E3 and PP-B5. In the case of FSS-E3 and PP-B5, the treatment of these items in the FPRA satisfies the requirements of CC I, which is judged to be sufficient for this application. Thus, the VEGP Fire PRA meets the requirements of RG 1.200, Revision 2.

3.3.2.3 Resolution of Findings from RG 1.200 Fire PRA Peer Review

Table 8 shows the details of the 50 Findings and the associated resolutions developed after the peer review.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
PP-A1-01	PP-A1, PP-C2	NOT MET	This SR states: INCLUDE within the global analysis boundary all fire areas, fire compartments, or locations within the licensee-controlled area where a fire could adversely affect any equipment or cable item to be credited in the Fire PRA plant response model including those locations of a sister unit that contain shared equipment credited in the Fire PRA. A review of PRA-BC-V-12-004, Version 0, identified several structures within the Protected Area that were identified within the global analysis boundary, but were then screened from further analysis, but the justification for screening appears to be inadequate or incorrect. Table 2.1-1 of PRA-BC-V-12-004, Version 0, lists the plant structures that were considered when determining the scope of the Fire PRA. This table also "screens" some of the structures based on varying criteria. Since a screening is performed without performing any quantitative evaluations, the screening needs to meet the requirements of Section 4.2-4 (QLS) of the ASME/ANS Standard. For a number of the structures screened, this criterion appears to not be met - and the structures should have been retained for further evaluation in the analysis	This F&O has been resolved, and a focused-scope Peer Review for the QLS element found all associated SRs are MET with no FINDINGS. The existing analysis includes a number of plant site locations where a postulated fire is either not reasonably expected to occur (natural draft cooling tower) or where a fire is not anticipated to cause or require a plant shutdown (demineralizer water). These locations were originally screened from the analysis. All plant locations within licensee-controlled area were reassessed and the scope of locations included in the analysis was expanded. Those locations that were screened were addressed in a focused-scope Peer Review for the QLS element which found all associated SRs MET with no FINDINGS. The overall FPRA analysis and related documentation was updated to reflect the changes in the global analysis boundary.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			A number of structures are screened stating: "A walkdown and a review of equipment layout and electrical one-line drawings confirmed it contains no equipment that would impact plant operations and is not susceptible to fire." This does not address the potential for cables that have the potential to impact plant operations to be in the facility. Any PAU that has cables whose failure could cause spurious operation of any equipment, system, function, or Operator action credited in the Fire PRA must be retained in the analysis. IF there are any cables in the facility that are unknown, the facility cannot be verified to not have cables that meet this criterion, and must be retained for further analysis." Re-evaluate the facilities listed in Table 2.1-1 and ensure any that are "screened" are documented to meet the criteria specified in Section 4-2.4 (QLS) of the ASME/ANS standard. Any that do not meet the criteria specified should be retained in the analysis.	
PP-A1-02	PP-A1	NOT MET	This SR states: INCLUDE within the global analysis boundary all fire areas, fire compartments, or locations within the licensee-controlled area where a fire could adversely affect any equipment or cable item to be credited in the Fire PRA plant response model including those locations of a sister unit that contain shared equipment credited in the Fire PRA.	This F&O has been resolved. The F&O details include items that overlap with and duplicate F&O PP-A1-01. The discussion of the resolution of this F&O focuses only on those details not already addressed and resolved as described in the resolution for PP-A1-01.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			A review of section 2.1 of PRA-BC-V-12-004, Version 0, indicates that the development of the fire PRA global analysis boundary may have missed several key fire areas. Section 2.1.2 describes the process that was used to develop the global analysis boundary, and Table 2.1-1 is supposed to be a summary of all fire areas that were considered for the global analysis boundary, and justification for what was excluded. A review of this table indicates that some potentially significant fire areas were not identified as being within scope of the global analysis boundary. Based on the fact that there are several major fire areas that are not identified as being with the Global Boundary, it is not apparent that all potential risk significant fire areas were identified as part of the global boundary analysis definition. Develop a complete list of all structures, above ground and below ground, within the Owner Controlled Area for the Vogtle site. Once the list is developed, ensure that ALL fire areas are addressed within the analysis. If appropriate, identified fire areas can be qualitatively screened - but only if the criteria in Section 4.2-4 is fully met.	The scope of the plant locations included in the analysis was reviewed and confirmed to have included the locations identified in the F&O. All of the plant Fire Areas as well as the specific locations identified in the F&O are included in the analysis scope – none of these locations were screened. Those locations that were screened were addressed in a focused-scope Peer Review for the QLS element which found all associated SRs MET with no FINDINGS.
PP-B2-01	PP-B2,	NOT MET	Calculation PRA-BC-V-12-004 was reviewed for	This F&O has been resolved.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
	PP-B3		treatment of non-rated barriers credited in defining PAUs. Table 2.1 addresses the credited non-rated barriers and documents the results of a walkdown of the barriers. This table appears to be incomplete. For example, the floor/ceiling between zone 49 and zone 149 in fire area 1-AB-LI-B is not addressed (note that others may exist, this example was the first example one looked at). This barrier is not a rated per drawing AXYDR801. In addition, no barriers for Unit 2 are included, and per discussions no walkdowns of Unit 2 were performed. Inspection of Unit 1 non-rated barriers for justification of Unit 2 non-rated barriers is likely not sufficient. Additionally, the walkdown evaluated spatial separations that were credited in definition of PAUs and concluded they were adequate given there were no ignition sources in the spatial separation or no propagation path in the zone of influence of postulated transients. This walkdown did not consider spatial separations in unit 2. The analysis performed during the walkdown appears to have supports that the barriers would "substantially contain the damaging effects of fires", but the documentation of the walkdowns is lacking detail to call it a justification. For example, in some cases the description is "Closed Wall". Additional	Additional walkdowns were performed to confirm the adequacy of the credited non-rated barriers for both units. The criteria that was used required barrier construction using non-combustible materials (concrete or masonry block) with no visible openings or confirmation that there were no combustible targets within the fire zone of influence when projected beyond any opening within a barrier. In all instances, the results of the effort confirmed that the existing treatment was consistent with the characteristics and configuration of the feature. The documentation was expanded and enhanced to include additional technical information to justify the basis for all credited non-rated fire barriers.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			details would add to credibility to the justification of the barriers." Inappropriate credit of non-rated barriers could lead to inappropriate definition of PAUs. Review all non-rated barriers. As a minimum, review all Fire Areas that have multiple fire zones and confirm that the zones are not separated by non-rated barriers. Suggest that additional detail be added to table 2.1 to solidify the justification of using the barriers. Walkdown unit 2 spatial separations and document.	
PP-B5-01	PP-B5	NOT MET	The first assumption in PRA-BC-V-12-004 states that active fire barriers were credited as part of plant partitioning. No information is provided that indicates where these barriers are used and does not justify the active fire barriers except to say that they are utilized in the Fire Hazard Analysis (FHA). Failure to identify the active fire barriers may hamper model maintenance and no justification is provided. Provide a list of the active barriers credited and the justification for credit.	This F&O has been resolved. The design of the plant includes the use of fire dampers and active fire doors. All of the active features credited in the FPRA were confirmed to be the same as those used and controlled by the plant Fire Protection Program. The analysis documentation was updated to add this technical information. The FPRA does not credit any other active features. The associated SR is met at CC I which is judged to be adequate for this application and not crediting other active features is

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
				conservative for the calculation of fire risk contributions. The standard would require the analysis to credit additional active features that are not used and controlled by the plant Fire Protection Program to met CC II/III.
PP-B7-01	PP-B7	MET CC I/II/III	Attachment D of the Plant Partitioning document shows that walkdowns were performed but does not really indicate any information associated with the fire barriers. Table 2.2-1 shows that walkdowns were performed on the Unit 1 non-rated barriers. Based on discussions with the plant team, the rated barriers are controlled under the Fire Protection Visual Inspection procedures 29144-1 and 29144-C. These procedures should be referenced as part of Section 2.2 in PRA-BC-V-12-004. Rated fire barriers are discussed in Attachment D and not clearly addressed in the current document. Add reference to the FP visual inspections to PRA-BC-V-12-004.	This F&O has been resolved. Documentation was expanded to extract details from the Fire Hazards Analysis and replicate in the FPRA documentation. Other documentation enhancements were also incorporated as noted in the F&O.
ES-A1-01	ES-A1	MET CC I/II/III	This SR States: IDENTIFY equipment whose failure, including spurious operation (see ES-A4), caused by an initiating fire would contribute to or otherwise cause an automatic trip, a manual trip per procedure direction, or would invoke a limiting condition of operation (LCO) that would necessitate a shutdown where	This F&O has been resolved. The FPRA assumes at a minimum an induced plant trip for all locations within the global analysis boundary. This treatment bounds the instances in which a manual shutdown may be required given a Tech Spec LCO.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			(a) shutdown is likely to be required before the fire is extinguished, (b) a potentially significant effect on safe shutdown capability is caused by the affected equipment, or (c) the shutdown will be modeled as a plant trip rather than a slow, controlled shutdown of the plant based on the current modeling practice in the Internal Events PRA. A Tech Spec LCO review could not be found. The SR requires a systematic review of Tech Specs to identify any equipment whose failure could invoke a LCO that would require a shutdown (no time frame specified) WHERE other conditions/plant impacts could also exist. No documentation of any LCO review could be found. Perform, and document a systematic review of Technical Specification, and any other governing documents that could require a plant shutdown, and ensure that any equipment that meets the criteria specified in this SR are added into the scope of the Fire PRA. .	An operator reviewed each plant location to determine if a fire at the location would result in a plant trip or manual shutdown. These locations were included in the analysis. A focused-scope Peer Review was performed for the QLS element which found all associated SRs MET with no FINDINGS. Therefore, it was confirmed that a fire at a location in the global analysis boundary that may lead to a reactor trip given a Tech Spec LCO was retained in the analysis. Based on this existing treatment, no technical change in the FPRA is needed to resolve this F&O. The analysis documentation will be updated to clarify this element of the analysis.
ES-C2-01	ES-C2	NOT MET	Each category of this SR starts with: IDENTIFY instrumentation associated with each operator action to be addressed, based on the following: a) fire-	This F&O has been resolved. The Peer Review found that when the

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			induced failure of any single instrument whereby one of the modes of failure to be considered is spurious operation of the instrument. This was found to not have occurred for several credited operator actions. All of the Operator actions from the internal events PRA model do identify at least one instrument/cue that triggers the Operator to perform the action. However, only 1 of the 6 Fire specific Operator actions identified at least one instrument/cue that would trigger the Operator to perform the action. Since this SR states that "each operator action be addressed," and only one of the new operator actions meets this requirement, potentially significant instruments may not have been identified. Identify at least one instrument/cue for each credited Operator action, and ensure that the instrument is included in the scope of the Fire PRA.	documentation for 5 of the 6 new operator actions that were added to the FPRA was reviewed it did not identify associated instrumentation. A review of all credited operator actions was performed to confirm that all associated instrumentation was identified and treated appropriately in the FPRA. That review determined that no technical change to the FPRA model was required. An update to the documentation has been completed to add the information that was inadvertently omitted.
ES-D1-01	ES-D1	MET CC I/II/III	This SR is associated with the documentation of the Equipment selected and the process for identifying the equipment. Table 4.1-3 of the Component and Cable Selection report uses Type Codes that are not reflective of current Vogtle Type Codes and a correlation	This F&O has been resolved. The details in the F&O refer to the use of screening type codes in Table 4.1-3 of the analysis report. The report creates a source of confusion as the type code that is referred to is not the type code used in the basic events

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			between the "N1" categorized Type Codes and the Vogtle Type Codes does not exist. Table 4.1-3 should be updated/revised to reflect the current Internal Events PRA Type Code naming scheme, or the type codes could be removed and only component types and failure modes used in the table.	naming convention. The report will be updated to clarify this usage.
CS-C2-01	CS-C2	MET CC I/II/III	The list of components, cables, and function codes (Attachment 3 of PRA-BC-V-12-005) did not contain all components modeled in the Fire PRA and contained a number of function codes not used in the Fire PRA model. To facilitate review, upgrade, and application, the list of function codes, components, cables, etc. should be limited to only those actually used in the Fire PRA model. Example: Condensate storage tank level transmitters 1LT5101 and 2LT5101 are included in the Fire PRA model but were not listed in Attachment 3 of PRA-BC-V-12-005 (Data Query Circuit Report: Function Code Sort.) These transmitters are in listed in the ARC database along with the required power, instrumentation, and indication cables along with their cable routing. Missing information makes the document	This F&O has been resolved. The version of the database output that was included in the report did not include the use of a filter. As a result, records that are retained in the database for historical purposes were inadvertently included. The report has been updated to correct this issue. Accordingly, the circuit analysis packages from Attachment 3 now correspond to the functional states modeled in the FPRA.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			incomplete; extraneous information makes it difficult to confirm which equipment was modeled in the Fire PRA and unnecessarily complicates review and update. Consistently mark records as significant to FPRA based on the function codes used in the Fire PRA model and produce a report containing only FPRA-significant information.	
CS-C2-02	CS-C2	MET CC I/II/III	A summary of the fire zone nomenclature (e.g. used in cable routing) and table associating fire zones with physical analysis units and referring to appropriate plant drawings and site maps would simplify review. Information is available but scattered, complicating review. Condense the information from the FSAR Chapter 9A (Fire Hazards Analysis) into a table. Add nomenclature description and appropriate plant drawings and site maps.	This F&O will be resolved in a future documentation update This F&O refers to a documentation enhancement. The resolution of this F&O has no impact on any technical element of the analysis.
PRM-A1-01	PRM-A1	MET CC I/II/III	This SR states: CONSTRUCT the Fire PRA plant response model so that it is capable of determining fire-initiated conditional core damage probabilities (CCDPs) and conditional large early release probabilities (CLERPs) for various fire scenarios. Although most of the model appears to be	This F&O has been resolved. The entire scope of mutually exclusive gates was reviewed and no other instances beyond the three occurrences identified in the F&O were found (MUTEX46, MUTEX-ISINJ-HE1, and

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			appropriate, there were several Mutex gates that do not appear to be technically valid and should be revised/removed to ensure that the Fire-induced CCDPs and CLERP are modeled appropriately. The following 3 Mutex gates appear to be invalid, and should be revised/removed from the Fire PRA Plant Response Model. The other Mutex gates modeled in the Fire PRM appear to be valid for the Fire PRA. MUTEX46 - This configuration is allowed by Tech Specs and is not mutually exclusive so the logic should be removed (note that this is also true for the Internal Events model and logic should be removed from that model as well). MUTEX-ISINJ-HE1 - This logic appears to be valid for an Internal Events scenario, but may not be valid for a Fire-induced initiator since the timing of the events for Internal Events is easy to follow, but the timing of events for the Fire is not as straight forward, and the combination of events may be phenomenologically possible. Suggest adding the %ISINJ tag into the logic to retain the logic for Internal Events and eliminate it from the Fire-induced scenarios.	MUTEX529). All of these instances have been corrected. The results of these changes did not alter the reported FPRA results for CDF or LERF.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			MUTEX529 - Discussions with SNOC personnel indicate that this combination is not expected to show up. Since it is unclear what the technical basis is behind this logic, it is recommended that this logic be removed from the model." Remove/revise the 3 MUTEX gates as discussed above.	
PRM-B13-01	PRM-B13	NOT MET	This SR states: For any item identified per PRM-B12, PERFORM the data analysis portion of the Fire PRA plant response model in accordance with HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D and their SRs in Section 2 with the following clarifications: a) All the SRs under HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D in Section 2 are to be addressed in the context of both random events as well as fire events causing damage to equipment and associated cabling, and b) DEVELOP a defined basis to support the claim of no applicability of any of these requirements in Section 2. Based on the PRM document and the fault tree model, a significant number of new basic events associated with equipment failures were added into the Fire PRM. Each of the newly added basic events currently have a random failure probability of "0" assigned to them. This convention means they never randomly fail. If the intent of these added	This F&O will be resolved prior to implementation of the 10 CFR 50.69 program in a future model update. This issue has no technical impact on the quantification. This F&O refers to the naming convention used in the FPRA for fire specific FLAG events. The naming convention used was the same as that used for random failure basic events instead of the convention used for FLAGs. This creates a source of confusion as basic events appear in the FPRA model with a 'zero' probability. These events were added to the model for potential fire impact on MSO and operator actions from instrumentation impacts. These events are treated for fire induced failure only and are set to "TRUE" or the circuit failure likelihood probability when potentially impacted by a postulated fire. This treatment is consistent

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			failures was to function as "Tag/flag" events for use in the fire PRA only, which is an acceptable approach, then the analysis should use the Vogtle Tag/Flag naming convention instead of a basic event naming scheme that links to pre-defined type codes. If the intent of these added failures is to function as basic events, they need to be revisited, and data assigned to them in accordance with the requirements of this SR. If the intent of these added failures was to function as "Tag/flag" events for use in the fire PRA only, then they should use the Vogtle Tag/Flag naming convention instead of a basic event naming scheme that links to pre-defined type codes. If the intent of these added failures is to function as basic events, they need to be revisited, and data assigned to them in accordance with the requirements of this SR. In either case, the documentation needs to be updated to identify all the added failures, and how they are intended to be used in the model.	with SR SY-A15. Documentation has been updated to clarify treatment of events added for fire induced failure concerns. This issue has no technical impact on the quantification.
PRM-C1-01	PRM-C1	NOT MET	This SR is associated with the quality of the documentation of the PRM model. Although some portions of the PRM model documentation was easy to follow and understand, there was a lot of extraneous/no longer used	This F&O has been resolved. The information that was provided in the analysis documentation included a listing of the basic events from the CAFTA database. This printout included extraneous events that are not

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			modeling included in the report that resulted in confusion as to what was and was not included in the current Fire PRA model. Additionally a listing of modified/added basic events was not provided and could not be easily obtained from the basic event file. Create a copy of the CAFTA BE/TC/GT databases, purge these copies, then re-generate the supporting tables, and include a complete listing of modified/added basic events.	used in the model but exist in the database. For the fire PRA, the database was purged to clearly identify basic events used in the fire PRA. Additionally, the documentation was updated to include a table for modified basic events, new basic events, and removed basic events.
FSS-A1-01	FSS-A1	MET CC I/II/III	Review of the ignition sources within the physical analysis units suggests that some ignition sources have been screened out from the scenario selection process without proper justification and documentation. Question FQ-A1 was submitted during the peer review week associated with this issue. The answer to the question listed specific reasons for screening cabinets that are not fully documented and justified. The answer for example includes a statement -8 sources were not used given potential for fire spread. No specific fire modeling justification for no fire spread was readily available. Fire scenarios that may inappropriately be excluded from the scenario selection process and quantification.	This F&O has been resolved. The documentation that was provided to the Peer Review team did not explicitly disposition each of the individual fire ignition sources that were identified during the fire frequency development. A review of all identified fire ignition sources was performed and a disposition was provided for each item. The results of this effort did not result in any additional fire initiating events needing to be added to the analysis. The ignition sources that were not explicitly treated were excluded because a postulated fire would have no consequential impact beyond itself (loss of only the fire source) or had been subsumed by another fire initiating event. The documentation

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			Account for each fixed ignition source identified so that a proper disposition is available and the risk contribution from each is accounted for.. Including the risk contribution that is properly screened out.	has been enhanced to provide this additional technical detail.
FSS-A2-01	FSS-A2	MET CC I/II/III	Instrument tubing has not been considered as a potential target in the Fire PRA. Indications to support human failure events are credited in the analysis, which brings into the scope of the fire PRA the instrument tubing. Instrument tubing should be included as targets and evaluated accordingly. Fire impacts to indications may not be fully accounted and reflected in the quantification process. Identify locations and damage thresholds for the instrument tubing and include them in the fire scenario development and quantification process..	This F&O has been resolved. The issue of instrumentation tubing is a potential concern for those instances where long lengths of tubing are used. In such cases, the failure to consider fire impact to the tubing could result in non-conservative results. Supplemental walkdowns were performed to specifically examine instrumentation installations. In all instances, the length of tubing was minimal and the treatment of the instrument itself and related cabling was sufficient to bound the tubing exposure. The documentation has been enhanced to provide this additional detail.
FSS-A3-01	FSS-A3, FSS-E4	NOT MET	Review of the FRANC database identified scenarios where events dispositioned as Y3 in the Equipment Selection Report are excluded events. The Y3 disposition is used for components or groups of components for which the cable routing is not known. By excluding these events, credit is being taken for these components, that is, it is assumed the cables for the component(s) are not impacted by the scenario. Some examples are scenarios 1530	This F&O has been resolved. The use of assumed routing was applied on an as-needed basis. As applied, the treatment is more appropriately described as crediting by exclusion. That is to say, that while the routing of the circuits was not explicitly determined, selected locations were identified where there was high confidence that no related equipment

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			A, 1532 A, 1048-DC B and 2068-MQ B1 where events 1FW, 1IA, 1TPCCW and othersY3 events are excluded. This is not meant to be a complete listing. No basis is provided for crediting components where cable routing is not known. Review the excluded events table for components dispositioned as Y3 in the Equipment Selection database and provide basis for exclusion. Document the basis or remove the credit for the component. If assumed routing is used as a basis for exclusion, ensure supporting requirement CS-A11 is met.	or circuits were present. This approach and treatment is consistent with Note 8 of SR CS-A10. The documentation that was Peer Reviewed provided a general discussion of the approach and methodology but did not provide specific details for each instance where this approach was used. The documentation was enhanced to explicitly address and justify those instances where this approach was used. No technical change to the FPRA was required. The treatment remains conservative and consistent with the Standard in the context that equipment and cables that are not explicitly traced are treated as failed in all plant locations unless there is reasonable confidence that they are not present.
FSS-A5-01	FSS-A5	NOT MET	Transient fires are not consistently postulated as described in the FSS notebook throughout the unscreened physical analysis units. As examples, no transients have been postulated in the containment, cable spreading room or switchgear rooms. The walkdowns conducted during the peer review in the cable spreading room and switchgear rooms indicate that there are transient fire scenarios that should be postulated as there are trays or conduits within the zone of influence of a transient fire. This a systematic issue not limited to the areas walked down.	This F&O has been resolved. In concert with the additional plant walkdowns that were performed to support the resolution of other F&Os, the overall treatment of transient fires was also re-verified for both units. Additional postulated transient fire initiating events were defined and added to the analysis. The incorporation of these additional fire initiating events has a minimal effect on the CDF and LERF results. In all cases, the consequence of the postulated transient fire is

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			The risk contribution of the different physical analysis units can be underestimated. Consistently postulate transient fires throughout the unscreened physical analysis units.	bounded by another existing fire initiating event. However, the transient initiating event frequency is several orders of magnitude lower.
FSS-A5-02	FSS-A5	NOT MET	This SR is associated with assigning the correct ignition frequency(s) to evaluate the risk of the PAUs. A review of the PAU-level ignition frequencies documented in Task 6 does not always match the sum of the PAU scenario level ignition frequencies. Some of the summations of ignition frequencies for PAUs overcount the total ignition frequency for the PAU (e.g. 1044-D2 has a total ignition frequency in FRANC of 7.21E-4/yr but the summation for the PAU from Task 6 is only 6.81E-04/yr), while others undercount the total ignition frequencies (e.g. 1140A-S1 has a total ignition frequency of 4.78E-04/yr in FRANC but the summation for the PAU from Task 6 is 2.18E-03 [1.59E-03/yr from fixed sources]). No bases for these discrepancies were provided. Ensure that the total ignition frequency for each PAU is partitioned appropriately. Justify excluding the ignition source contribution for any ignition sources that are "screened" from having an impact.	This F&O has been resolved. The observed differences in the ignition frequency has been addressed and corrected as necessary in conjunction with the resolution of F&Os FSS-A1-01 and FSS-A5-01.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
FSS-B1-01	FSS-B1	NOT MET	The FSS notebook does not describe abandonment criteria for leaving the control room due to operability. Abandonment is only based on fire generated conditions (i.e. habitability). Consider as the context of this scenario a relatively small fire in a control board affecting enough controls (e.g. resulting in a high CCDP) forcing operators to leave the control room before fire generated conditions force abandonment. Abandonment based on plant operability is not considered or evaluated. Include in the Fire PRA modeling of control room abandonment due to operability. The cable end points and their mapping to basic events can provide insights on determining abandonment. ..	This F&O has been resolved. The treatment of control room abandonment in the FPRA and the related operator interview results were re-assessed. The operator interviews indicated that they would generally be reluctant to abandon the main control room. The existing FPRA treated this reluctance by not crediting possible recoveries using designed plant features supporting shutdown from outside the main control room. The FPRA quantifies the fire consequence for equipment inside and outside the control room. Each fire scenario was reviewed to determine the impact on operation from the control room. Fire scenarios resulting in high consequence are control room abandonment scenarios which were treated with a 1.0 CCDP given the potential of hot shorts resulting in failure of equipment relied upon when shutting down outside the control room. Other scenarios with a significant CCDP include fires at the electric power control board which may result in a station blackout condition. Other fire scenarios inside and outside the control room result in at most the loss of a single

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				division and offsite power. Given these failures, the control room still has full operation of the opposite division with a diesel generator. Therefore, the FPRA addressed the concern of loss of control resulting in abandonment. The documentation was updated to further describe the review of potential fire scenarios resulting in a loss of sufficient controls resulting in abandonment.
FSS-B2-01	FSS-B2	NOT MET	The FRANC database distributed during the peer review week only list one transient fire for the control room. No documentation for the screening process of transient fires has been provided. The risk contribution for transient fires in the control room should be considered. The control room for example should have relatively high transient influence factors for occupancy and maybe in other factors. Add transient fire scenarios to the control room so that the contributions from such ignition sources are captured.	This F&O has been resolved. The resolution for this F&O is subsumed by the actions associated with F&O FSS-A5-01.
FSS-B2-02	FSS-B2	NOT MET	This SR is associated with Main Control Room (MCR) abandonment scenarios. The documentation states that all MCR abandonment scenarios have a	This F&O has been resolved. The FPRA treatment for all abandonment cases

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			CCDP of 1.0 (0.91 when accounting for capacity factor), but 7 scenarios were identified that had a CCDP of 1.2809E-03. (ABN-1 through ABN-7). Since these scenarios were not set to 1.0, (0.91 when accounting for capacity factor) as per the methodology described, and additional ex-control room Human Error Probabilities (HEPs) were not reviewed for them, this is undercounting the CDF/LERF impact of MCR abandonment. Revise the CCDPs to 1.0 (0.91 when accounting for capacity factor) for the impacted scenarios.	was reviewed to confirm that a correct CCDP value was applied. The instances identified in the F&O were determined to be the only cases where an incorrect value was used. The analysis and associated documentation have been corrected as necessary.
FSS-C4-01	FSS-C4	NOT MET	The severity factors approach is based on an unreviewed analysis method. Currently in the FPRA there appears to be no credit for suppression activities when the severity factor is applied. Therefore, the scenario quantification does not limit the severity of the fire from the perspective of suppression credit. This assessment is limited to the treatment of severity factors and non suppression probabilities at the time of the peer review.	This F&O has been resolved. The application of the method to the FPRA is consistent with the results of the recent industry review completed subsequent to the Peer Review. This method involves the application of a modified fire factor développed through a review of the individual industry fire events that form the basis for the generic fire frequency. However, the FPRA has applied a conservative value – higher than the value proposed in the method reviewed.
FSS-C4-02	FSS-C4	NOT MET	A review of control room abandonment scenarios indicated that some of the abandonment scenarios (ABN3, ABN4, ABN6, and ABN7) have an additional	This F&O has been resolved. All entries used in the FRANC NSP field were

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			NSP of 0.1 assigned. However, an NSP of 1E-3 is already taken in the Control Room Abandonment calculation. Non-Suppression Probability (NSP) appears to be double counted for these events. Remove the 0.1 factor or provide additional justification to show that double counting is not occurring.	reviewed and confirmed to be appropriate. The review found that the instances identified in the F&O are the only cases where an incorrect value was used. The analysis and the associated documentation have been corrected.
FSS-C7-01	FSS-C7	NOT MET	Suppression systems are credited in the Fire PRA. Specific examples include the credit for manual suppression for hot gas layer scenarios and the credit for fixed suppression systems in the multi compartment screening analysis. Assessment for multiple suppression paths is not provided. The standard requires evaluation of potential conflicts with the credit of multiple suppression paths. Evaluate P&ID's for fire water and document independent paths for the features credited.	This F&O has been resolved. The F&O identifies a number of situations where the independence of multiple suppression paths did not adequately justify the applied failure probabilities in the context of dependencies between those paths. The dependency involves potential failures of the fire water system that could disable water based suppression systems as well as the use of hose streams in support of manual suppression. A review of the overall FPRA confirmed that the only instance where this occurs is in the MCA considerations. The overall treatment of these multiple suppression credits in those treatments was re-assessed and the manual suppression credit was either reduced to account for dependencies as necessary. The net impact of this update did not result in any change in the overall analysis

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
				results.
FSS-D3-01	FSS-D3, FSS-C3	NOT MET	This standard requirement indicates that there needs to be reasonable assurance that the risk contribution is bounded or realistically characterized. There is no clear/consistent treatment of intervening combustibles in the Fire PRA that would ensure that risk contribution is bounded. Currently, the analysis includes the assumption of only two cable trays would be involved in a fire. This assumption has the potential for underestimating the temperature in a PAU resulting in a hot gas layer scenario been missed. Risk contributing scenarios can be missed or excluded from the quantification. Develop and apply a consistent treatment of secondary combustibles that reflect plant specific configurations.	This F&O has been resolved. In concert with the additional plant walkdowns that were performed to support the resolution of other F&Os, the definition of the fire scenarios was re-verified for both units. This re-verification included specific consideration of secondary combustibles that could participate in the postulated fire scenarios. The result of these walkdowns did not identify any cases where the existing fire scenarios were found to be insufficient with respect to bounding the consequences of the postulated fire event.
FSS-D4-01	FSS-D4, FSS-D11, FSS-H4	NOT MET	Key input parameters for the models, including room sizes and heat release rates associated with intervening combustibles are not properly justified. Specifically, PAU sizes are referenced to the combustible calc (FHA) and are based on floor areas. No verification is included for these values. The heat release rates associated with intervening combustibles is not justified for specific scenarios. Finally, the heat release rate for transient fires in a number of PAU's is assumed to be 69 kW, which	This F&O has been resolved. The analysis documentation was enhanced to provide a more complete summary of the various fire modeling parameters. As noted in the resolution discussion of F&O FSS-D3-01, this included confirmation of the appropriateness of the treatment of secondary combustibles (intervening combustibles).

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			appears to be developed from an unreviewed analysis method (no specific reference to reviewed industry documents for this value is provided). The parameters appear to be applied on an as needed basis using different approaches. A table listing all the scenarios with the corresponding parameters for each scenario is not readily available in the documentation to show/support the parameters used for each scenario. Input parameters for fire modeling calculations should be justified, verified, and documented in a manner that allows a reviewer to see what was applied for each scenario, and the basis for selecting the scenario-specific parameters. Justify, verify, and document input parameters for the fire modeling analysis. At the time of the peer review, critical input parameters including PAU sizes and heat release rate intensities were not adequately justified or verified. A consolidated table listing the fire scenarios and their specific parameters with clear concise description of each parameter and the resulting frequency used in the scenario quantification process would be an efficient way to ensure clear, concise validation, documentation and completeness of the input parameters used in the analysis.	With respect to the heat release rate used for transient fires, many of the plant locations are spatially smaller which necessarily restricts the floor area for the placement of any transient ignition source and combustible. This limited floor space translates to a lower credible heat release rate. For the large plant locations, a larger heat release rate was used. The overall treatment was consistent with the latest industry guidance as developed by an EPRI sponsored review effort and distributed to industry. The consensus was that NUREG/CR-6850 allowed for modified heat release rates for different plant locations.

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FSS-D7-01	FSS-D7, FSS-D8	NOT MET	Detection and suppression features are credited in the analysis. In addition, the multi compartment analysis credits fixed suppression systems. Since suppression is credited in the FPRA, a review and documentation of governing procedures should be included in the FSS report. Assumptions associated with detection and suppression activities are not properly documented. This particularly applies to specific features at Vogtle, as the FSS report essentially references generic/reviewed industry methods. No documentation/justification that the generic failure probabilities are consistent with the installation and maintenance of the system, the system is fully operable, and meets applicable codes and standards is provided. Justification for the applicability of the generic failure probability values used in the analysis should be provided. In addition an assessment of the effectiveness of the system for the credited scenarios is required. References to specific plant procedures, pre fire plans, etc are necessary. As an example of the impact of the SR in the analysis, during the peer review walkdowns it was observed that fire protection valves are under replacement requiring fire watch in some physical analysis units. The standard requires justification for the values selected and an assessment of the system's	This F&O has been resolved. The F&O raises a concern with respect to NFPA code compliance and system availability. An effort was undertaken to confirm that credited fire protection features conform to the requirements of the applicable fire protection codes, and that there has not been any outlier experience with respect to availability. In all cases, no conditions were found that would require any technical (numerical) change in the FPRA.

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			effectiveness, including verification that plant-specific unavailability history is not an outlier with respect to industry experience. Document Vogtle specific detection, suppression and manual suppression characteristics. Justify the detection/suppression probability values used in the analysis. Validate and document that the Vogtle plant-specific unavailability is not an outlier with respect to industry experience.	
FSS-E3-01	FSS-E3	MET CC I	Southern Nuclear PRA Calculation No. 0293100011.11, Rev.0, Section 1.3 (Fire Scenario Selection Report) and Vogtle Fire PRA Quantification Report No. 0293100011.14, Rev. 0, Section 7.0 provides a characterization of the uncertainty in various aspects of the analysis. The Monte Carlo analysis in Section 9.0 of Quantification Report No. 0293100011.14 does not reflect all uncertainties because distributions are not identified for all parameters. Upgrade to achieve Category II. Establish uncertainty distributions for all uncertainties and propagate the distributions through the Fire PRA model using a Monte Carlo simulation.	This F&O has been resolved. The specific issues that were identified for this F&O involves uncertainty associated with the original NUREG/CR-6850 fire frequency values and uncertainty associated with the input parameters to the fire modeling analyses. The FPRA treatment of uncertainty included a treatment of all parameters that can be mathematically propagated within the capabilities of available analysis tools and software. In the case of fire frequency, uncertainty analysis was performed based on the updated fire frequency values (EPRI TR-1016735). The specific requirement for sensitivity using NUREG/CR-6850 values is specific to NFPA

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				805 applications as noted in FAQ 08-0048. In the case of fire modeling input parameters, that task was performed in a fashion that tended to skew the results towards a conservative upper bound. As such, the results are already conservative and any uncertainty analysis would tend to result in lower CDF/LERF results. However, there is no consensus basis for establishing the uncertainty characterizations. The associated SR was dispositioned as CC I which is judged to be sufficient given the two concerns noted.
FSS-F1-01	FSS-F1	NOT MET	Southern Nuclear PRA Calculation No. 0293100011.11, Rev.0 (Fire Scenario Selection Report), Section 4.1.7.4, indicates that plant buildings have requirements for fire ratings as identified in Design Criteria DC-1000A (General Design Criteria - Architectural) and that this should constitute compliance with this supporting requirement. However, this supporting requirement necessitates a detailed inspection and listing of buildings with exposed structural steel and whether any high hazard fire sources exist in those buildings. This has not been provided in Calculation No. 0293100011.11. Walkdowns conducted during the peer review week suggests that there are exposed	This F&O has been resolved. A review and walkdown of the plant was performed to identify locations where exposed structural steel was present together with a high hazard fire source. The result of this effort did not identify any instances where both co-existed. Therefore, there was no need to modify the analysis for treatment of possible structural collapse. The analysis documentation was enhanced to provide this information.

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			structural steel elements throughout the plant. A detail inspection following the requirements in the FPRA standard may result in fire scenarios requiring quantification. Completeness of analysis for all buildings in the global analysis boundary, and documentation of this analysis for future use. Compliance would typically involve providing a table of all buildings in the global analysis boundary, identification of whether exposed structural steel exists in each building, and identification of whether there is a postulated fire source/fire scenario that could damage the building structure. If an assumption of building collapse is made in FSS-F2 instead of further evaluating the structural capability of a particular building, then associated scenarios must be evaluated and included in the FSS-F3 risk quantification.	
FSS-G3-01	FSS-G3	MET CC I/II/III	There is no systematic process for identifying multi compartment scenarios. Specifically, a multi compartment matrix is not developed. The lack of a rigorous process for defining multi compartment combinations, multi compartment scenarios can be missed and not included in the quantification. For example, vertical combinations of compartments	This F&O has been resolved. A comprehensive multi-compartment matrix was developed and added to the analysis documentation. This process was also used to support the resolution of other F&Os related to barrier adequacy and active fire barrier features. The results of this enhancement of

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			(compartments above or below the compartment of fire origin) are not considered in the analysis. Multi compartment scenarios may be missed and not quantified. Generate a multi compartment matrix and apply the screening criteria to all combinations. Quantify the unscreened combinations.	documentation did not identify any new fire scenarios that required quantification or further treatment in the FPRA.
FSS-G4-01	FSS-G4, FSS-G5	NOT MET	No justification is provided for the credit allowed for non rated barrier or active fire barriers. The documentation is not clear on how the barrier failure probabilities are assigned to each scenario. Walkdowns conducted during the peer review week where the spatial separation in the aux building (equipment hatch) is next to relatively large pumps. Barrier failure probabilities are important screening factor and/or parameter in the risk equation. Proper justification for its use should be documented and applied. The peer review finds that the application of barrier failures probability in the FPRA is inconsistent with the definition of physical analysis units in covered by the PP element. The spatial separation or active fire barrier probabilities are not consistent. For example, spatial separation is credited in plant partitioning and the multi compartment does not	This F&O has been resolved. As noted in the resolution discussions for F&Os PP-B2-01, PP-B5-01, PP-B7-01, and FSS-G3-001, additional assessment, walkdowns, and documentation enhancements were performed to address the associated concerns. The net result of the efforts did not change the resultant analysis in the context of CDF and LERF results. Various documentation enhancements were also incorporated to include the related information.

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Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			assign barrier failures of 1.0 to those combinations. Document and apply a consistent method for determining using barrier failure probabilities.	
FSS-H1-01	FSS-H1, FSS-H5	NOT MET	The fire modeling documentation appears to not be complete or reviewed at the time of the peer review. For example, information important to the development of scenarios such as the handling of fire wraps which are present in the plant is not included in the documentation so it is not clear whether or not they are credited in the analysis. Additionally, there are references to the Duane Arnold Plant in the documentation, which would have been flagged if the documentation had been reviewed. Peer reviewed should receive complete and reviewed Fire PRA notebooks. With the notebook in the state it is in – it is not clear what is applicable to Vogtle, and what is "leftover" from the Duane Arnold Fire PRA.. Review and resolve comments for the FSS notebooks and ensure that all Vogtle specific information is included in the report.	This F&O has been resolved. Additional reviews of the report were performed and corrections have been incorporated. The process of completing this effort did not identify any substantive change to the technical portions of the analysis or methodology discussions.
FSS-H7-01	FSS-H7	NOT MET	Suppression is credited in the FPRA. Review and documentation of governing procedures should be	This F&O has been resolved.

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			included in the FSS report. Assumptions associated with detection and suppression activities are not properly documented. This particularly apply to specific features at Vogtle, as the FSS report essentially references generic/reviewed industry methods. References to specific plant procedures, pre fire plans, etc are necessary. This is a documentation issue directly related to the SR. Document Vogtle specific detection, suppression and manual suppression characteristics.	As discussed in the resolution for FSS-D7-01, supplemental reviews were performed to confirm the appropriateness of the credit assigned for various fire protection system elements (suppression, detection, barriers, etc). This process and the associated governing plant procedures were added to the analysis documentation.
IGN-A7-01	IGN-A7	MET CC I/II/III	Walkdown documentation does not allow confirmation of all ignition sources being included in the Fire PRA process. Instances were found where items are in the model and not in Appendix C or D of Southern Nuclear PRA Calculation No. PRA-BC-V-12-004 (Plant Partitioning and Fire Ignition Frequency, Version 0, NUREG/CR-6850 Task 1 & 6). Documentation and understanding of the Fire PRA model for future use. Correlate the Fire PRA model and the equipment items listed in the ignition frequency report. Avoid having non-existent equipment items listed in the model with a zero CCDP that are not in the ignition	This F&O has been resolved. During the course of the development of the FPRA, instances were identified where fire ignition sources were not explicitly included in the counting for fire ignition frequency. As the analysis was developing, it was not viewed as a significant issue as it would have tended to under-estimate the denominator of the fraction used in the analysis resulting in very slightly conservative fire frequency values. Given the minor changes in the population of counted components, the exclusion of these items from the fire ignition frequency development has almost no impact on the fire frequency for any individual scenario. Instead, a per component fire frequency was used to generate a value for

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			frequency report.	any 'missing' item during the fire scenario development process. The very slight conservative biasing of the results does not have a significant impact on the FPRA results.
IGN-B1-01	IGN-B1	MET CC I/II/III	Ref: Southern Nuclear PRA Calculation No. PRA-BC-V-12-004 (Plant Partitioning and Fire Ignition Frequency, Version 0, NUREG/CR-6850 Task 1 & 6). Calculation of the ignition frequency for Fire Areas 1-AB-LD-A and 1-AB-LB-A appears to have included fire zone 1035-C6 in the wrong fire area." Incorrect PAU ignition frequency may be used in future Fire PRA calculations or risk-informed decisions. Revise the documentation in Calculation No. PRA-BC-V-12-004 to reflect the correct ignition frequency.	This F&O has been resolved. The relationship between fire areas and zones that are used in the analysis was reviewed. It was confirmed that the error identified in the F&O is isolated. The analysis documentation has been corrected.
IGN-B4-01	IGN-B4	MET CC I/II/III	Section 3.2.1 of Southern Nuclear PRA Calculation No. PRA-BC-V-12-004 (Plant Partitioning and Fire Ignition Frequency, Version 0, NUREG/CR-6850 Task 1 & 6) states that Bayesian update was not performed for Bins 5, 6, 11, 24, and 31. No justification was documented for this in the report. If there is justification for not doing a Bayesian update, due to reclassifying generic events within the	This F&O has been resolved. A Bayesian update of the generic fire frequency values for Bins 5, 6, 11, 24, and 31 has been performed. The posterior mean reflects a very slight reduction. The FPRA and related reports have been updated to reflect these updated results.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			industry, this should be documented. Otherwise these Bins should be Bayesian updated. Technical accuracy and consistency with treatment of all other BIN ignition frequencies. Document basis for not Bayesian updating the ignition frequencies for Bins 5, 6, 11, 24, and 31 in Section 3.2.1 of Southern Nuclear PRA Calculation No. PRA-BC-V-12-004 (Plant Partitioning and Fire Ignition Frequency, Version 0, NUREG/CR-6850 Task 1 & 6).	
HRA-B3-01	HRA-B3	MET CC I	For new operator actions, particularly OACONTROL--AFW and OAISOLSTMTDAFW, which are listed as risk significant, the HRA evaluation does not include a scenario description, a procedure reference, the operator cues, or operator interview insights. In particular, actions to correct spurious actuation would require the operators to determine that the system is not needed before tripping the system. Screening value for the HRAs has been justified by the cause-based decision trees. However, the assessment does not have sufficient analysis to verify that the screening value is appropriate. For risk significant fire actions, perform a detailed HRA evaluation with operator talk-through,	This F&O has been resolved. At the time of the Peer Review, the FPRA used applied both detailed and screening HRAs. The F&O identified a number of instances where screening HEPs were used that were found to be risk significant. The FPRA has been updated to include detailed HRA for all modeled actions. The analysis and associated documentation have been revised to reflect this update.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			procedure references and cues to perform the action and cues that may impact the diagnosis. For non-risk significant actions, provide the procedural guidance and the justification for the screening values used.	
FQ-A1-01	FQ-A1	MET CC I/II/III	This SR states: for each fire scenario selected per the FSS requirements that will be quantified as a contributor to fire-induced plant CDF and/or LERF, TRANSLATE the equipment and cable failures, including specification of the failure modes, defined per the FSS element into basic events in the Fire PRA plant response model including consideration of insights from the circuit failure analysis. Contrary to this requirement, the MCR abandonment scenarios did not translate the equipment and cable failures, but assumed a CCDP for the scenarios." Most scenarios appeared to translate the equipment and cable failures appropriately. The exception to this was the MCR abandonment scenarios which did not translate the equipment and cable failures, but used an assumed CCDP for the scenarios. A review of the information currently available for the MCR scenarios showed that the information required to do this for the MCR abandonment scenarios was readily available and could be generated.	This F&O has been resolved. Due to the potential for spurious actuation given a fire resulting in abandonment conditions, a CCDP of 1.0 was assumed for abandonment scenarios. Due to the low probability of MCR abandonment, the use of CCDP of 1.0 does not result in these scenarios as significant contributors to plant risk. Given the methodology that is used, the explicit mapping of equipment and cable failures to basic events to support CCDP quantifications is not relevant given the treatment method that is used.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			Translate the equipment and cable failures for the MCR abandonment scenarios.	
FQ-B1-01	FQ-B1	NOT MET	This SR has back-references to the Internal Events QU SRs. In particular, QU-B2 and QU-B3 are concerned with truncation limits and showing convergence. Although the quantification document does provide a discussion on truncation and convergence, the method used to obtain/provide evidence of convergence does not technically meet the requirements of these SRs. Although the document does not specifically discuss the process used to obtain the convergence results, discussions with SNOC personnel indicate that the model was only "quantified" one time at a truncation value of 1E-9 to obtain CCDP cutsets. These cutsets were merged together and then the ignition frequencies/Severity factors/non-suppression probability values were applied to the cutsets. Once the factors were added, cutsets were removed below the specified new "truncation" values to show convergence. Although this method would show where convergence is obtained within the single quantified cutset file, it does not show convergence of the accident sequences and associated system models since only a single quantification was performed. Quantification	This F&O has been resolved. The F&O details refer to the lack of documentation for convergence. The FPRA was quantified using FRANC with a CCDP truncation of 1E-9. A formal test for convergence has been performed and shows convergence occurs with a CCDP truncation of 1E-7. Therefore, the use of a CCDP truncation of 1E-9 is acceptable. The analysis documentation has been updated to include the FRANC CCDP truncation study.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			at higher/lower CCDP truncation values will provide different cutsets, that when the IF/SF/NSP values are applied and the cutsets below the new "truncation" values are determined, different results are obtained. Quantify the FRANC model at various CCDP truncation values to verify convergence.	
FQ-B1-02	FQ-B1	NOT MET	This SR has back-references to the Internal Events QU SRs. QU-B1 provides requirements for use of software for quantifying. This F&O does not address standard quantifying software (CAFTA, FTREX, etc.), but rather it focuses on non-standard software which is used in the quantification process. Per discussions with ERIN personnel the following software is used in the quantification process: VEGP Scenario Database - This database is used to generate scenarios. The queries in this database could be construed to be software. CTT Appender - This ERIN in-house software is used for appending cutset files from the various scenarios into a single file. In addition to appending files together, the code adds scenario IDs to each cutset and creates and replaces BEs for altered	This F&O has been resolved. The supplemental software tools that are used in the development of the FPRA include an Access database and a cutset merging utility. The VEGP Scenario Database is an Access database with simple queries that involves somewhat standard data relationships largely identical to that in FRANX. The data tables were verified to ensure the queries were providing the correct data. The verification has been included in the updated quantification report. A benchmarking of CTT Appender was performed by not setting a truncation in CTT Appender. The resulting answer matched the FRANC answer. The verification has been included in the updated quantification report.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			events with a unique basic event so as to avoid loss of the altered event probability. Both of these items are subject to the requirements of QU-B1. There is no documentation that the VEGP Scenario Database and the CTT Appender meet the requirements of QU-B1 and they have been used in the quantification process. Perform software qualification as needed and document how the requirements of QU-B1 are met for the VEGP Scenario Database and CTT Appender.	In both cases, the favorable comparison of results is the basis to conclude that the requirements of QU-B1 are MET.
FQ-B1-03	FQ-B1	NOT MET	This SR has look backs to the Internal Events QU SRs. In particular, QU-B1 is concerned with ensuring that method-specific limitations and features of software used in the quantification process that could impact the results are identified. A review of the documentation did not identify any discussion associated with understanding the method-specific limitations and features of the software selected to support the Fire PRA. A review of the limitations of software being used for this analysis by the review team identified several concerns that the limitations of the software being	This F&O has been resolved. The details for this F&O refer to documentation of software limitations. This F&O is related to, and to some degree duplicates, the concerns noted in FQ-B1-02. The discussion of the resolution of FQ-B1-02 resolves the F&O related to software. The other issue that was noted in this F&O is the use of 0 as a value in the quantification instead of setting the event to FALSE. This latter issue has no known quantification concerns. The

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			used may not be fully understood by the SNOC Fire PRA Team. Identify all of the software packages, including "in-house" software packages (excel spreadsheets, MSAccess queries, etc.) that have been used to directly support the development and/or quantification of the Fire PRA. Review each of the software packages and ensure that they have been demonstrated to generate appropriate results when compared to those from accepter algorithms, and that the limitations of the software packages are know and documented.	updated quantification report includes identified limitations of FRANC.
FQ-B1-04	FQ-B1	NOT MET	This SR has back looks to the Internal Events QU SRs. QU-B9 is associated with the requirement to set logic flags to TRUE or FALSE versus 1.0/0.0 prior to the generation of cutsets. Since the Fire PRA does not set the logic flags to FALSE, but uses a value of 0.0 instead, this SR is not met. Develop a defined basis to support the no applicability of the QU-B9 requirement, or set the flags to FALSE in the PRA model and compress the fault tree prior to using it via linking it with FRANC.	This F&O has been resolved. A sensitivity study was performed to address the use of sequence tracer events (FLAGS) set to 1.0 in the quantification versus setting the events to TRUE. The sensitivity study used the CAFTA cutset editor to set events with a value of 1.0 to 'TRUE', to compress those events, and to minimize the updated results. The net impact of this was a minor reduction of the total CDF result. With respect to the second issue related to the use of 0 versus 'FALSE', there is no known issue with this approach.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			.	
FQ-C1-02	FQ-C1	MET CC I/II/III	A review of HRA dependencies in top scenario found that the first HRA combination includes events OA-ALTAFW----H and OAB_TR-------H associated with aligning an alternate water source for AFW and Feed and Bleed Injection. Based on discussions with the Internal Event HRA lead, the combination is in a sequence that is well beyond the PSA mission time (35 hrs vs. 30 hr time to CSD) and was identified as a sequence that should be removed from the model in a Suggestion Level F&O in the FPIE Peer Review. Setting OA-ALTAFW----H to false reduces the CDF by 5% on the top fire scenario. Therefore, it could have an impact on overall CDF. It also adds questions on the depth of HRA dependence review since this is the top HRA combination in the top CDF sequence. Remove the accident sequences longer than 30 hours from the Fire PRA Quantification Model as suggested in the FPIE Peer Review.	This F&O has been resolved. The process of identifying joint human action dependencies was repeated following the incorporation of changes associated with the resolution of the F&Os. The results of this process did not identify any other instances where dependence between action beyond the mission time occurred. This single instance noted in the F&O has been resolved.
FQ-F1-01	FQ-F1	NOT MET	This SR is associated with documenting the	This F&O has been resolved.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			quantification process and its attributes. Although the process followed appears to be fairly robust, documentation of the process is not as robust, and all attributes of the SR are not fully documented in a manner to ensure that they are satisfied. Add information into the documentation to provide the required level of detail to ensure that all of the attributes of the FQ SRs are met. For example, add in a discussion that explains that every Unit 2 PAU is assumed to result in a Unit 1 reactor trip, and provide a method of identifying how to determine which Unit 2 PAUs actually impact Unit 1risk (e.g. CCDP for normal Unit 1 Rx trip is xxx - so any CCDP above this value impacts Unit 1 risk). Provide a similar discussion for Unit 2. Then provide a table in the report that identifies those Unit 1 PAUs that impact Unit 2, the Unit 2 PAUs that impact Unit 1, and a list of the PAUs that impact both Units. The following SRs appear to be met technically after discussions with SNOC personnel, and walking through an actual quantification, but it is not readily apparent they are met without a lot of extra leg work and discussions. FQ-A2 - there is no discussion of Unit 2 PAUs that impact Unit 1 (or vice-versa), and there is no	The documentation changes and updates associated with the resolution of the other F&Os for this review element addresses and resolves this F&O.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			discussion on dual unit initiators and how they are handled. FQ-B1 - not all programs actually used in the process were identified in the report, and there is no discussion that the limitations of the software used in the process were reviewed and understood. Additionally, the approved software list did not always have the same version of the software package as being approved for use, and some of the software packages were not include on the list at all. FQ-B1 - Some Microsoft Access queries were developed to support the fire PRA, but there is no identification of this in the discussion of software used, and there is no documented "software quality" associated with these queries. FQ-B1 - The discussion on convergence does only provides the results, but does not provide the process followed to actually obtain the results. Since it is obvious that the results being discussed are not a direct output of the FRANC quantification, a discussion of how the actual CDF/LERF values are obtained to show convergence needs to be provided. If the intent of these added failures was to function as "Tag/flag" events for use in the fire PRA only, then	

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			they should use the Vogtle Tag/Flag naming convention instead of a basic event naming scheme that links to pre-defined type codes. If the intent of these added failures is to function as basic events, they need to be revisited, and data assigned to them in accordance with the requirements of this SR. In either case, the documentation needs to be updated to identify all the added failures, and how they are intended to be used in the model.	
FQ-F1-02	FQ-F1	NOT MET	This SR is associated with the documentation of the quantification process. Scenario specific quantification factors were applied to the selected fire scenarios. The only record that was provided that identified where these factors were applied was the "VEGP-1-Scenario.mdb" MSAccess database. No documentation was provided that described the process followed to determine where the various quantification factors could be applied or how the applicable quantification factors were calculated for each scenario. The establishment of the parameters used in the FRANC quantification code is heavily dependent upon the "VEGP-1-Scenario.mdb" MSAccess database. This database contains a lot of information, but there is no direct discussion of how the database works (uses inclusion versus exclusion	This F&O has been resolved. This F&O addresses weaknesses in the documentation of the quantification process. The documentation has been revised to provide additional details.

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			logic), calculates the ignition frequencies for scenarios, identifies/applies quantification factors for scenarios, etc. There is no discussion of how the "excluded events" were determined/validated for each scenario, and how this information was added into the database to verify that all of the events in the "excluded events list" should have been excluded from the associated scenario evaluation. There is no roadmap showing how the queries are linked and run in the "VEGP-1-Scenario.mdb" MSAccess database, nor is there a roadmap explaining how the outputs of this software is imported into the FRANC database. The CTT Appender software is used to append information to the cutsets (e.g. ignition frequencies, non-suppression probabilities, severity factors), but this software and the process followed to convert the cutsets from CCDP/CLERP to CDF/LERF is not discussed in the documentation. Provide the documentation necessary to "verify/replicate" the information and processes followed during the quantification process. For example, document the process followed for	

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			determining where the various quantification factors could be applied, and document the calculation of the applicable quantification factors for each scenario.	
SF-A2-01	SF-A2	MET CC I/II/III	As part of their Seismic Fire Interactions, Vogtle looked at the Gaseous Systems and the dry, pre-Action Sprinkler Systems in the Auxiliary Building. In looking at the evaluation, it was determined that there were also dry, pre-action sprinkler systems as well as several wet-pipe sprinkler systems in the turbine building that were not evaluated. Vogtle needs to include an evaluation of the turbine building in Section 3 of Report 0293100011.13 to ensure that the potential impacts of seismic failures of these systems are considered in the seismic fire interaction evaluation. This F&O was discussed with Vogtle staff and they plan to add an additional table to Section 3 of Report 0293100011.13 to address the turbine building sprinkler systems. A draft of the table was provided and it appears to be sufficient to fully resolve this F&O.	This F&O has been resolved. The analysis was updated to include a review of the same suppression system types in the turbine building. The results of the review did not identify any suppression system seismic/fire interactions of concern.
UNC-A1-01	UNC-A1	MET CC I/II/III	Per requirements of HLR-QU-E in section 2 of the Standard, OAB_TR-------H has a probability of 7.3E-2 and an EF of 5 in the HRA Calculator worksheet. However, in the cutset file, the event is split into two	This F&O has been resolved. A review of the entire scope of HRA results was performed and all instances where an HEP of

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			events to address HRA dependencies. The first event is OAB_TR-------H with a probability of 0.119 and an EF of 1. The second event is recovery 1OVDM-OABTR2 with a probability of 0.61 and an EF of 1. Therefore, the EF does not match with the EF assessed in the HRA Calculator sheet. Based on discussion with the plant team, HRA Calculator assigns a value of 1 for HEPs above 0.1. (Note that this is not consistent with NUREG-1278 and should be corrected.) By breaking the HRA event to perform the HRA dependency analysis causes the EF for this HRA to be lost. The EF for this HRA event is less than the HRA Calculator assigned EF. Other HRAs could have similar issues based on the high HEPs used for the Fire PRA. Review the HRA events in the plant database and assign an EF consistent with the HRA Calculator (and NUREG-1278).	0.10 or higher occurs was identified. For these cases, the HRAC assigned EF was replaced with a value of 3. This EF is consistent with current industry consensus practices. The range of HEPs above 0.10 was also reviewed and it was confirmed that the highest HEP was 0.25.
UNC-A2-01	UNC-A2	MET CC I/II/III	Quantification Report No. 0293100011.14, Rev. 0, Section 8.1 identifies differences between mean values of CDF and LERF when using the new EPRI ignition frequencies and NUREG/CR-6850 ignition frequencies. However, a Monte Carlo simulation should be performed and uncertainty intervals	This F&O has been resolved. As noted in the resolution discussion of FSS-E3-01, the FPRA is based on the updated EPRI fire frequency values as published in NURG/CR-6850, Supplement 1. Uncertainty analysis for

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
			identified for CDF and LERF (along with the mean values) using the new EPRI ignition frequencies and NUREG/CR-6850 ignition frequencies. This requirement is derived from IGN-A10 and IGN-B5 requirements. Technical adequacy and completeness. Perform Monte Carlo simulation using old and new ignition frequency data.	the actual fire frequency data used in the analysis has been performed. There is no requirement to perform Monte Carlo uncertainty analysis on prior data no longer used in the FPRA model.
UNC-A2-02	UNC-A2	MET CC I/II/III	Supporting requirement FQ-F1 and FSS-E4 are not met. This results in supporting requirement UNC-A2, which references them, to be met but with this F&O. Documentation related to the propagation of uncertainty from these supporting requirements to CDF and LERF uncertainty has not been fully achieved. Documentation to support future Fire PRA application. Upgrade the documentation related to uncertainty.	This F&O has been resolved. The resolution of the F&Os linked to FQ-F1 and FSS-E4 effectively also closes this F&O. It is noted that the F&O associated with FSS-E4 is FSS-A3-01. With respect to FSS-A3-01, the treatment remains conservative and the CDF and LERF results retain an upper bound bias because of this bias. As a result, a treatment of uncertainty would reflect a risk reduction and is therefore not explicitly treated.
MU-C1-01	MU-C1	NOT MET	The PRA configuration control process does not include evaluation of the cumulative impact of pending changes on risk applications.	This F&O has been resolved. Although this F&O is linked to FQ, the issue is more closely related to the SNC procedures for

Table 8 - Resolution of the VEGP Fire PRA Peer Review Findings				
Finding & Observation (F&O) Number	**Review Element**	**Capability Category (CC)**	**Finding Description**	**Resolution**
				model maintenance and update. The associated procedure is in the process of being updated. Since one of the SNC plants is transitioning to NFPA 805, that plant will have the lead in generating the update to the fleet procedure for this process. This F&O has an impact on the future update to the FPRA, but at this point, this F&O has no technical impact on the FPRA.

3.3.2.4 General Conclusions Regarding FPRA Capability

The information provided in the Section 3.3.2 of this LAR demonstrates that the VEGP at-power FPRA models conforms to the standard at CC-II, which satisfies the guidance of RG 1.200, Revision 2.

The VEGP FPRA model technical capability evaluations described above provide a robust basis for concluding that the FPRA models are suitable for use in supporting the implementation of 10 CFR 50.69.

3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

3.4.1 Rule Requirements

Per 10 CFR 50.69(b)(2)(iv) an LAR must include the following: "A description of, and basis for acceptability of, the evaluations to be conducted to satisfy § 50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions)."

Per 10 CFR 50.69(c)(1)(iv) the following is required: "Include evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment permitted by implementation of § 50.69(b)(1) and (d)(2) are small."

Per 10 CFR 50.69(b)(1) licensees are allowed to use alternate ways for complying with the requirements outlined in 10 CFR 50.69(b)(1) for RISC-3 and RISC-4 SSCs.

For RISC-3 SSCs, 10 CFR 50.69(d)(2) states that, "The licensee or applicant shall ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions, including seismic conditions and environmental conditions and effects throughout their service life. The treatment of RISC-3 SSCs must be consistent with the categorization process. Inspection and testing, and corrective action shall be provided for RISC-3 SSCs.

(i) Inspection and testing. Periodic inspection and testing activities must be conducted to determine that RISC-3 SSCs will remain capable of performing their safety-related functions under design basis conditions; and

(ii) Corrective action. Conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design basis

> conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition."

The SNC categorization process for 10 CFR 50.69, its proposed implementation, and this submittal, meet the above requirements, as described in the following sections.

3.4.2 Explanation of Process

The process that SNC will use at VEGP for categorization of SSCs into the four risk-informed safety categories (RISC-1, RISC-2, RISC-3, and RISC-4) defined in 10 CFR 50.69 is specified in SNC procedures. The guidance in SNC procedures is consistent with and is intended to implement the categorization process guidance provided in NEI 00-04 as endorsed by RG 1.201, with clarification. The process is risk-informed, in that it combines risk insights with traditional engineering principles of maintaining defense-in-depth and adequate safety margins, and employing an IDP, as described in section 3.1 of this LAR, to ensure that the system functions and operating experience have been appropriately considered in the categorization process. The following discussion explains how the SNC process meets 10 CFR 50.69(b)(2)(iv) requirements.

3.4.2.1 Evaluations for Safety Margin, Change in CDF/LERF, and the Potential Impacts from Known Degradation Mechanisms

In accordance with NEI-00-04 section 8, SNC's 10 CFR 50.69 process includes performance of a risk sensitivity study to confirm that the categorization process results in acceptably small increases to CDF and LERF. The set of SSCs whose unreliability is adjusted in the final sensitivity study to assess the acceptability of the categorization results is determined through the overall process defined in the SNC 10 CFR 50.69 procedures and subordinate instructions, which are consistent with NEI 00-04 sections 2 through 7. The integrated risk sensitivity study conservatively increases the failure rate by a factor of 3 for all candidate RISC-3 PRA modeled SSCs simultaneously to ensure that potential increases in CDF and LERF due to changes in treatment are small.

The SNC 10 CFR 50.69 process defense in depth evaluation is as defined in NEI 00-04. For SSCs that are identified as candidate RISC-3, and also for any redundant identical RISC-3 SSCs within the system, the evaluation assures that key safety functions are still maintained assuming that the candidate RISC-3 SSCs do not perform their function.

As part of the 10 CFR 50.69 implementation, a performance monitoring process will be defined and implemented at VEGP to ensure that potential increases in failure rates of categorized components will be detected and addressed before reaching the rate assumed in the sensitivity study. Performance monitoring of categorized SSCs, and PRA updates that SNC will implement during program

implementation, in accordance with the rule requirements, will continue to capture failure data for RISC-3 SSCs, and will allow for the timely identification of any important new degradation mechanisms that may have a bearing on the categorization.

Paragraph 10 CFR 50.69(d)(2)(ii), Corrective Action, states that conditions that would prevent a RISC-3 SSC from performing its safety-related functions under design basis conditions must be corrected in a timely manner. For significant conditions adverse to quality, measures must be taken to provide reasonable confidence that the cause of the condition is determined and corrective action taken to preclude repetition. The primary intent of this provision is to address the possible effects of potential common cause failures and degradation mechanisms following implementation, as discussed in paragraph 10 CFR 50.69(b)(2)(iv). The VEGP corrective action process will be followed regarding potential conditions adverse to quality (including common cause failures). Per this process, the cause of the condition must be determined and corrective action taken in a timely manner to preclude repetition.

3.4.2.2 Evaluation for Common Cause Interaction Susceptibility

Common cause interactions are addressed in the SNC 10 CFR 50.69 process as follows.

The SNC 10 CFR 50.69 categorization procedure specifies the process as defined in NEI 00-04. This process requires consideration of common cause risk importance measures, both risk achievement worth (RAW) and Fussell-Vesely (F-V). This requires that groups of components with potentially high common cause impacts based on quantitative PRA models are maintained in the RISC-1 or RISC-2 categories (i.e., high safety significant).

SNC's 10 CFR 50.69 categorization process addresses both known degradation mechanisms and common cause interactions for both active and passive functions, and meets the requirements of 10 CFR 50.69(b)(2)(iv). The failure rates for equipment and initiating event frequencies used in the VEGP PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, human errors, etc.).

Common cause treatment in the VEGP base (internal events at power) PRA used to support the 10 CFR 50.69 process meets the requirements in the ASME/ANS PRA Standard. For example, the criteria for treatment of common cause are delineated in the supporting requirements in the ASME/ANS PRA Standard associated with high level requirement HLR-SY-B. The VEGP internal events at power PRA includes a robust evaluation of common cause, equivalent to that specified for Capability Category II of the ASME PRA Standard for the relevant supporting requirements. This has been confirmed via industry peer review of the VEGP PRA, as described in section 3.1 of this Enclosure 1.

The specific NRC regulatory positions on NEI 00-04 as identified in section C of RG1.201 are addressed relative to the SNC 50.69 process proposed for implementation at VEGP via this LAR in Appendix A. As indicated in Appendix A, no exceptions to those regulatory positions are taken.

3.4.3 Conclusion

The SNC 10 CFR 50.69 process implements the guidance in each section of NEI 00-04. It includes: the SSC categorization process defined in sections 2 through 7 and 10; the overall risk sensitivity defined in section 8 that is used to confirm that the categorization process results in acceptably small increases to CDF and LERF; the IDP function, defined in section 9, of reviewing and ensuring that the system functions and operating experience have been appropriately considered in the process; and the processes, defined in sections 11 and 12, to provide reasonable confidence that the validity of the categorization process (including the risk sensitivity study) is maintained. The process includes consideration of the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions, considering internally and externally initiated events at full power and shutdown conditions. For passive components, the SNC 10 CFR 50.69 process utilizes the guidance in EPRI TR-112657, Revision B-A, as detailed in section 3.1.2.

Given the above, the SNC 10 CFR 50.69 process to be implemented at VEGP addresses all aspects of the guidance in NEI 00-04. The process provides reasonable confidence that, for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment permitted by implementation of 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) are small, and that that the requirements of 10 CFR 50.69(c)(1)(iv) are met.

4.0 REGULATORY EVALUATION

4.1 Significant Hazards Consideration

The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. Implementation of 10 CFR 50.69 allows for application of a risk-informed categorization process per 10 CFR 50.69(c) to modify the scope of SSCs subject to special treatment requirements. Alternative treatments per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. Implementation of 10 CFR 50.69 will allow the licensee and the NRC to better focus attention and resources on SSCs that have safety significance, resulting in improved plant safety.

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

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

   Response: No.

   The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. The process provides reasonable confidence that, for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small per 10 CFR 50.69(c)(1)(iv). The proposed OL LCs do not result in or require any physical or operational changes to VEGP SSCs, including SSCs intended for the prevention or mitigation of accidents. Implementation of 10 CFR 50.69 in compliance with 10 CFR 50.69 requirements ensures that RISC-1 and RISC-3 SSCs remain capable of performing their design basis functions, including safety-related functions, under design basis conditions. In addition, the process ensures that RISC-2 SSCs are capable of performing their safety significant functions.

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

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

   Response: No.

   The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. The process provides reasonable confidence that, for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small per 10 CFR 50.69(c)(1)(iv). The proposed OL LCs do not result in or require any physical or operational changes to VEGP SSCs, including SSCs intended for the prevention or mitigation of accidents. Implementation of 10 CFR 50.69 in compliance with 10 CFR 50.69 requirements ensures that RISC-1 and RISC-3 SSCs remain capable of performing their design basis functions, including safety-related functions, under design basis conditions. In addition, the process ensures that RISC-2 SSCs are capable of performing their safety significant functions.

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

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

   Response: No.

   The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. The process provides reasonable confidence that, for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small per 10 CFR 50.69(c)(1)(iv). The only requirements that are relaxed for SSCs, consistent with their categorization, are those related to treatment. The safety margins associated with SSCs design basis functions and design technical requirements remain unchanged. Additionally, it is required that there be reasonable confidence that any potential increases in CDF and LERF be small from assumed changes in reliability resulting from the treatment changes permitted by 10 CFR 50.69. As a result individual SSCs continue to be capable of performing their design basis functions. It is concluded that sufficient safety margins are preserved.

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

Based on the above, SNC concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.2 Applicable Regulatory Requirements/Criteria

The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented in this LAR per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. Implementation of 10 CFR 50.69 will be in compliance with 10 CFR 50.69 requirements to ensure that RISC-1 and RISC-3 SSCs remain capable of performing their design basis functions, including safety-

related functions, under design basis conditions. In addition, the process ensures that RISC-2 SSCs are capable of performing their safety significant functions.

The categorization process described in the LAR conforms to the guidance in NRC RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1 dated May 2006 (Reference 3). The categorization process also conforms to the guidance in NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0 dated July 2005 (Reference 4), as endorsed by RG 1.201. Conformance of the SNC categorization process to RG 1.201 is documented in Appendix A of this enclosure. As indicated in Appendix A, no exceptions are taken to RG 1.201.

4.3 Precedent

The NRC, by letter dated June 17, 2011 (Reference 2), in response to SNC letter dated December 6, 2010, granted pilot status for the VEGP 10 CFR 50.69 LAR which is the initial requested amendment to allow for the voluntary implementation of 10 CFR 50.69. There are no directly applicable precedents.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

The proposed VEGP Units 1 and 2 OL LCs will allow for the voluntary implementation of 10 CFR 50.69. The SNC risk-informed categorization process has been documented in this LAR per the requirements of 10 CFR 50.69(b)(2) and meets the requirements of 10 CFR 50.69(c). The SNC risk-informed categorization process will be used to modify the scope of SSCs subject to special treatment requirements. Alternative treatments permitted per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) can then be applied consistent with the categorization of the SSCs. Implementation of 10 CFR 50.69 will be in compliance with 10 CFR 50.69 requirements to ensure that RISC-1 and RISC-3 SSCs remain capable of performing their design basis functions, including safety-related functions, under design basis conditions. In addition, the process ensures that RISC-2 SSCs are capable of performing their safety significant functions.

The proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in

individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors", November 22, 2004.

2. NRC letter to Mark J. Ajluni, Director, Nuclear Licensing, SNC, June 17, 2011 [ML11171A084].

3. NRC Regulatory Guide 1.201 (for Trial Use), "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.

4. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.

5. ASME, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities", Code Case N-660, Revision 0, Section XI, Division 1, American Society of Mechanical Engineers, 2004.

6. EPRI TR-112657, Revision B-A, Revised EPRI Risk-Informed In-service Inspection Evaluation Procedure, Electric Power Research Institute, 1999.

7. WCAP-16308-NP-A, Revision 0, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program - Categorization Process - Wolf Creek Generating Station," August 2009 (PA-SEE-0027)" [ML092430185 and ML092430186].

8. "Final Safety Evaluation by the Office of Nuclear Reactor Regulation, Topical Report WCAP-16308-NP, Revision 0, "Pressurized Water Reactor Owners Group 10 CFR 50.69 Pilot Program – Categorization Process - Wolf Creek Generating Station", Nuclear Energy Institute Project No. 689," March 26, 2009 [ML090260674].

9. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991

10. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", US Nuclear Regulatory Commission, June 1991.

11. ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant

Applications", Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

12. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management, December," 1991.

13. NUREG-1855, "Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making", US Nuclear Regulatory Commission, March 2009.

14. VEGP PRA-BC-V-07-003, VEGP Level 1 and Level 2 PRA Model Revision 4 – at power, internal events, December 15, 2010).

15. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," Electric Power Research Institute, Final Report, December 2008.

16. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities", US Nuclear Regulatory Commission, September 2005

17. EPRI, NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Electric Power Research Institute, Revision 1, August 1991.

18. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," US Nuclear Regulatory Commission, 1973.

19. Letter from Dr. John Reed, Jack R. Benjamin & Associates, Inc., to Mr. Keith D. Wooten, Southern Company Services, Inc., Vogtle Units 1 and 2 Seismic Evaluation Peer Review, JBA Project No. 191-020, Plant Walkdown Status Report, January 10, 1994.

20. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants", US Nuclear Regulatory Commission, May 1978.

21. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," US Nuclear Regulatory Commission, June 1991

22. NUREG 75-087, "Standard Review Plan for the Review of Safety Analysis Report for Nuclear Power Plants," LWR edition, US Nuclear Regulatory Commission, December 1975.

23. INPO 06-08, Guidelines for the Conduct of Outages an Nuclear Power Plants", Institute of Nuclear Power Operators, December 2006.

24. SOER 09-01, Shutdown Safety, August 31, 2009.

25. NRC Inspection Manual Chapter 0609, Appendix G, "Shutdown Operations Significance Determination Process," 5/25/04.

26. ASME RA-Sc-2007, "ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications", Addenda to ASME RA-S-2002, ASME, New York, NY, August 31, 2007.

27. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 1, US Nuclear Regulatory Commission, January 2007.

28. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities", Revision 2, US Nuclear Regulatory Commission, March 2009.

29. NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Nuclear Energy Institute, 2000.

30. ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications, April 2002 and Addenda to Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-Sa-2003, American Society of Mechanical Engineers, 2003.

31. NEI 07-012, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines, Revision 1, Nuclear Energy Institute, 2010.

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
C.1	Revision 0 of NEI 00-04 references numerous other documents, but the NRC's endorsement of Revision 0 of NEI 00-04 does not constitute an endorsement of those other referenced documents	Conforms	See C.3
C.2	Revision 0 of NEI 00-04 includes examples to supplement the guidance. However, the NRC's endorsement of Revision 0 of NEI 00-04 does not constitute a determination that the examples are applicable for all licensees. A licensee or applicant must ensure that a given example is applicable to its particular circumstances before implementing the guidance as described in that example.	Conforms	SNC understands that the NRC's endorsement of Revision 0 of NEI 00-04 does not constitute a determination that the examples are applicable for all licensees.
C.3	To meet the requirements of §50.69 for categorization of SSCs, licensees may use methods other than those set forth in Revision 0 of NEI 00-04. The NRC staff will determine the acceptability of such other methods by evaluating them against the requirements of §50.69.	Conforms	SNC has used methodology explained in NEI 00-04 (Revision 0) for categorization of active components. SNC has used a method other than code case N-660 (Revision 0) for categorization of passive components. The SNC selected method is based on the EPRI risk informed ISI methodology (EPRI TR 112657 Revision B-A). Tables 1A and 1B in this enclosure provide a comparison of the SNC methodology and WCAP-16308-NP-A (including the NRC SER). The selected method is conservative and yields realistic results. This method was developed after NEI 00-04 was approved.
C.4	When PRAs have not been performed, NEI 00-04 (Rev 0) allows the use of non-PRA-type evaluations (e.g., fire-induced vulnerability evaluation (FIVE), seismic margins analysis (SMA), and NEI guidance in NUMARC 91-06, "Guidelines for Industry Actions to	Conforms	At the present time, SNC has a peer reviewed internal events (including internal flooding) PRA model and Fire PRA model. SNC will use qualitative approaches for other risks - Seismic risk, Other External Events, and Shutdown -

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	Assess Shutdown Management," to address shutdown operations). Such non-PRA-type evaluations will result in more conservative categorization.		until PRA models are developed. SNC has dedicated resources to develop Vogtle seismic PRA model.
C.5	Technical Adequacy Attributes of Analyses Implementing Revision 0 of NEI 00-04 The licensee or applicant is expected to document the technical adequacy of their internal events PRA for 50.69 application per RG 1.200, which endorses NEI 00-02 and ASME Standard RA-S-2002. The RG 1.201 Rev 1.0 further states that "However, the documents mentioned above currently cover only internal events at full power. There is not currently a similarly endorsed standard for the external events, internal fires, and low-power and shutdown PRAs, or for non-PRA-type analyses (e.g., FIVE, SMA, NUMARC 91-06), and Section 3.3 of Revision 0 of NEI 00-04 provides only limited guidance for determining the technical adequacy attributes required for these types of analyses for this specific application. Therefore, for §50.69 submittals that are received before the NRC endorses standards for external events, internal fires, and low-power and shutdown PRAs, as well as non-PRA-type analyses, the NRC staff expects the licensee or applicant to document the bases for why the method employed is technically adequate for this application. Toward that end, as part of the plant-specific application requesting to implement §50.69, the licensee or applicant will provide the bases supporting the technical adequacy of its	Conforms	Refer to section 3.2 of Enclosure 1 to NL-12-0932.

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	external events, internal fires, and low-power and shutdown PRAs, and non-PRA-type analyses for this application."		
C.6	Uncertainty Considerations in Revision 0 of NEI 00-04 The staff notes that the purpose of the sensitivity studies performed as part of the risk categorization process is to address the impact of parameter and model uncertainties on the categorization. The staff understands the phrase "applicable sensitivity studies identified in the characterization of PRA adequacy" (in Tables 5.2 through 5.5 of Revision 0 of NEI 00-04), as meaning those uncertainties not addressed by the other sensitivity studies in Tables 5.2 through 5.5. These uncertainties are typically identified via PRA peer reviews or self-assessments that are associated with the licensee's choice of specific models and assumptions, as discussed in Section 2.2.5.5 of Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."	Conforms	SNC agrees that the phrase "applicable sensitivity studies identified in the characterization of PRA adequacy" (in Tables 5.2 through 5.5 of Revision 0 of NEI 00-04), as meaning those uncertainties not addressed by the other sensitivity studies in Tables 5.2 through 5.5.
C.7	"Common-Cause Failure and Degradation Mechanism Considerations in NEI 00-04 (Rev 0) The NRC staff notes that mechanisms that could lead to large increases in core damage frequency (CDF) and large early release frequency (LERF), which could potentially invalidate the assumptions underlying the categorization process, including the risk sensitivity study, are the emergence of extensive **common-cause failures (CCFs)** impacting multiple systems and	Conforms (via Implementation Process)	The noted issues will be primarily addressed via a process consistent with that described in Section 12.4 of NEI-00-04. As noted in that section, various elements of the risk categorization process, which are defined in SNC 10 CFR 50.69 procedures, ensure that the potential for common cause failures of RISC-3 SSCs is appropriately considered. The categorization process elements that accomplish this include base PRA model

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	significant unmitigated degradation. However, for these types of impacts to occur, the mechanisms that lead to failure, in the absence or relaxation of treatment, would have to be sufficiently rapidly developing or not self-revealing such that there would be few opportunities for early detection and corrective action. Section 12.4 of NEI 00-04 describes an acceptable performance-based approach to address these concerns. Alternatively, those aspects of treatment that are necessary to prevent significant SSC degradation or failure from known mechanisms, to the extent that the results of the risk sensitivity study would be invalidated, could be identified by the licensee or applicant, and such aspects of treatment would be retained. This alternative approach would require an understanding of the degradation and common-cause failure mechanisms and the elements of treatment that are sufficient to prevent them. As an example of how this alternative approach might be implemented, the known existence of certain degradation mechanisms affecting pressure boundary SSC integrity could be used to support retaining the current requirements regarding inspections or examinations or use of the risk-informed ASME Code Cases, as accepted by the NRC's regulatory process. As another example, changing levels of treatment on several similar SSCs that might be sensitive to potential CCF would require consideration of whether the planned monitoring and corrective action program, or other aspects of		requirements, consideration of common cause risk importance measures (RAW and FV), defense-in-depth evaluation, and the integrated risk sensitivity study. In addition to the categorization process itself, the requirements of the rule for RISC-3 treatment, including test and inspection (§50.69(d)(2)(i)), periodic evaluation (§50.69(e)) and corrective action (§50.69(d)(2)(ii)), provide important defenses against the potential for common cause failures going undetected. In accordance with the process described in NEI-00-04 Section 12.4, performance monitoring of RISC-3 SSCs, as required by 10 CFR 50.69(e)(3), will be established to detect and address potential increases in failure rates before the rate assumed in the categorization integrated sensitivity study is reached. Failures of RISC-3 SSCs will be identified and tracked in the VEGP corrective action program. As part of the corrective action program, failures of RISC-3 SSCs will be reviewed periodically to determine the extent of condition (i.e., whether this failure is indicative of a potential common cause failure). As part of assessing data from the corrective action program for impact associated with alternate treatment, failures will be assessed for groups of like component types (e.g., motor operated valves, air operated valves, motor-driven pumps, etc.), regardless of system. The intent of the periodic review is

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	treatment, would be effective to sufficiently minimize the potential for CCFs impacting multiple systems, such that the categorization process (including the risk sensitivity study) remains valid."		twofold: First, to ensure that the failure rate of RISC-3 SSCs in a given time period has not unacceptably increased due to the changes in treatment. The periodic review will validate that the rate of RISC-3 SSC equipment failures has not increased by a factor greater than that used in the integrated risk sensitivity study to confirm acceptability of categorization results. Second, the review of component group failure data will be performed to detect the potential occurrence of inter-system common cause failures, and to allow timely corrective action if necessary, as required by §50.69(d)(2)(ii). Since most RISC-3 components have low failure rates, noted increases to these rates will be most readily detected through grouping of components. If failure rate increases are noted, attention will be focused on common treatment changes to groups of components to ensure that the potential for inter-system common cause failure remains low. This corrective action review will also consider previous component performance history. Criteria will be established for detecting adverse failure trends prior to exceeding the factor used in the categorization sensitivity study. If the number of failures for a group of SSCs exceeds

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
			the criterion, e.g., a factor of three increase over the expected (historical) number of failures, a potential adverse trend will be identified requiring further assessment. The failure criterion will be selected to assure an assessment is initiated prior to exceeding the factor used in the risk sensitivity study. Appropriate actions (which could include changes in treatment or categorization) will then be taken to preclude reaching unacceptable performance. If deemed to be appropriate to address the potential for significant component degradation due to particular known degradation mechanisms, or to address possible concerns regarding ability to adequately and promptly detect such mechanisms, SNC may determine that certain aspects of existing treatment that might otherwise be relaxed under this program should be voluntarily retained. This would be done on a case-by-case basis, with consideration of issues such as those noted by NRC in C.7.
C.8	The NRC staff notes that the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence in the evaluations required by §50.69(c)(1)(iv). All aspects of the guidance are important and interrelated. Sections 2 through 7 and Section 10 of NEI 00-04	Conforms (sections 2 - 7, 9, and 10) Conforms (via Implementation Process) for sections 11 and 12	SNC procedures follow all aspects of NEI 00-04 to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv). As noted in C.3, SNC has elected to use a method other than the Code Case N660 (Rev 0) for categorization of passive components. This method is based on the EPRI RI-ISI methodology (EPRI TR 112657 Rev B-A). It is

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	describe the processes used to determine the set of SSCs, for which unreliability is adjusted in the risk sensitivity study described in Section 8, which is used to confirm that the categorization process results in acceptably small increases to CDF and LERF. Section 9 describes the integrated decision-making panel (IDP) function of reviewing and ensuring that the system functions and operating experience have been appropriately considered in the process. Finally, Sections 11 and 12 describe the processes that provide reasonable confidence that the validity of the categorization process (including the risk sensitivity study) is maintained. Thus, all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv).		SNC's position that the selected method is superior than the Code Case N660 (Rev 0) that is mentioned in NEI 00-04. The selected method is conservative and yields realistic results. This method was developed after NEI 00-04 was approved. This selection has no impact on achieving reasonable confidence.
C.9; Section 1.2	The NRC staff understands that term "important-to-safety" refers to nonsafety-related SSCs that have been determined to be important. These nonsafety-related SSCs will be categorized as either RISC-2 or RISC-4, as determined by their safety significance, in accordance with the §50.69 categorization process. The NRC staff understands that the use of phrase "...blends risk insights, new technical information and operational feedback...." and similar phrases (e.g., the third guiding principle in Section 1.3), as meaning that the integrated decision-making process must systematically consider the quantitative and qualitative information available regarding the various modes of	Conforms	SNC procedures require the IDP to consider the SSC function(s) that caused it to be originally classified as important-to-safety in order for an LSS categorization to be justified. Notwithstanding this additional consideration, these nonsafety-related SSCs will be categorized as either RISC-2 if they are determined to be High Safety Significant (HSS) or RISC-4 if they are determined to be Low Safety Significant (LSS), as shown in Section 6.14 of the NMP procedure. SNC's understanding of phrase "...blends risk insights, new technical information and

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	plant operation and initiating events, including: PRA quantitative risk results and insights (e.g., CDF, LERF, and importance measures); deterministic, traditional engineering factors and insights (e.g., defense-in-depth, safety margins, and containment integrity); and any other pertinent information (e.g., industry and plant-specific operational and performance experience, feedback, and corrective actions program) in the categorization of SSCs."		operational feedback…." and similar phrases are in line with NRC understanding.
C9; Section 1.3	The second guiding principle in Section 1.3 states that deterministic or qualitative information should be used if no PRA information exists related to a particular hazard or operating mode. This principle is not to be understood to mean that deterministic or qualitative information should be used ***only*** when no PRA information exists. The NRC staff believes that the integrated decision-making process must systematically consider the quantitative and qualitative information available regarding the various modes of operation and initiating events, including PRA, quantitative risk results and insights; deterministic, traditional engineering factors and insights, and any other pertinent information in the categorization of SSCs.	Conforms	Qualitative information is reviewed for all components in the categorized system, not just those that are not modeled.
C9; Section 4.0	In Section 4.0 and Section 5.1, NEI 00-04 references ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," as an approach for addressing the pressure-retaining function or passive function of active components. The version of ASME	Conforms	As stated in C.3, SNC has elected to use a method other than code case N660 (Rev 0) for categorization of passive components. This method is based on the EPRI RI-ISI methodology (EPRI TR 112657 Rev B-A). It is SNC's position that the selected method is

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	Code Case N-660 that is acceptable to the NRC staff for use in this application is the version identified in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," subject to any conditions or limitations specified therein. Alternatives to this Code Case may be submitted for NRC review and approval as part of a specific §50.69 application.		superior to code case N660 (Rev 0) that is mentioned in the NEI 00-04. The selected method is conservative but still provides sufficiently realistic insights with regard to categorization. This method was developed after NEI 00-04 was approved.
C9; Section 6.2	In Section 6.2, the NEI 00-04 guidance contains criteria for confirming that an SSC is LSS (or recategorizing it as safety-significant) based on defense-in-depth considerations, which include criteria related to containment bypass, containment isolation, early hydrogen burns, and long-term containment integrity. The containment isolation criteria listed in this section of NEI 00-04 are applicable to containment penetrations. The NRC staff understands the use of the phrase "containment penetration" as including electrical penetrations, air locks, equipment hatches, and piping penetrations (including containment isolation valves). Further, the staff notes that the containment isolation criteria in this section of NEI 00-04 are separate and distinct from those set forth in §50.69(b)(1)(x). The criteria in §50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, but the §50.69(b)(1)(x) criteria are not used to determine the proper RISC category for containment isolation valves or penetrations.	Conforms (via Implementation Process)	SNC believes that this comment refers to alternative treatment. When developing alternative treatments, SNC will ensure that the criteria mentioned in §50.69(b)(1)(x) are used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50, but the §50.69(b)(1)(x) criteria are not used to determine the proper RISC category for containment isolation valves or penetrations.
C9:	The risk sensitivity study addresses the impact of	Conforms (via	The VEGP PRA model includes common cause

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
Section 8	potential increases in the failure rates of all RISC-3 SSCs resulting from the change in treatment. Section 8 of NEI 00-04 includes commentary on the consideration of known degradation mechanisms and common-cause interactions and failures in PRAs, which includes the observation that intersystem common-cause failures are not typically modeled because factors such as design diversity and different service environments ensure they are negligible contributors to risk. The NRC staff notes that because intersystem common-cause failures are typically not included in PRA models and, therefore, are not addressed by the risk sensitivity study. Therefore, the licensee or applicant relies upon the alternative treatment and feedback requirements, including corrective action provisions, established in §50.69 and discussed in Section 12 of NEI 00-04, to ensure that any significant intersystem common-cause failure mechanisms would be identified and corrected so that the assumptions underlying the categorization are not invalidated.	Implementation Process)	failures, in accordance with the criteria stated in the ASME/ANS PRA Standard. As noted in the NRC comments on NEI-00-04, modeling of inter-system common cause failure is not an expectation in the PRA Standard, and is not addressed in the VEGP PRA. The SNC alternative treatment process, which has not yet been defined, will include performance monitoring, categorization feedback, and corrective action steps. These will ensure that if a common cause failure mechanism (inter- or intra-system) were to develop, it would be detected (through the performance monitoring steps to be included in the implementation process), evaluated for categorization impact (in accordance with the process defined in SNC procedures), and promptly addressed through the VEGP corrective action process. Depending on the significance of the failure mechanism, the resulting action might include recategorization, modification to the alternative treatment for the SSCs in question, or additional (e.g., more frequent, or re-focused) performance monitoring.
C9; Section 9.2	Section 9.2 of NEI 00-04 limits the IDP review of risk information to active functions and SSCs. The NRC staff believes that this limitation in review scope is attributable to the reliance of NEI 00-04 on ASME Code Case N-660 to address passive functions, which is performed by an expert panel. The expert panel used in performing ASME Code Case N-660 may be the	Conforms	SNC procedures are developed such that 50.69 IDP is responsible for reviewing results of active and passive component categorization.

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	same panel as the IDP used in the §50.69 categorization process; however, it is not required to be the same panel. As such, the IDP review of risk information should address both active and passive functions and SSCs.		
C9; Section 11.2	The NEI 00-04 guidance allows licensees to implement different approaches, depending on the scope of their PRA (e.g., the approach if a seismic margins analyses is relied upon is different and more limiting than the approach if a seismic PRA is used). As part of the NRC's review and approval of a licensee's or applicant's application requesting to implement §50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, of the implementation of the new approach in their categorization process. The focus of the NRC staff's review and approval will be on the technical adequacy of the methodology and analyses relied upon for this application.	Conforms	SNC understands that SNC will have to submit a license amendment to obtain NRC approval prior to implementing a new approach (e.g., switch from a seismic margins analysis to a seismic PRA) in the categorization process.
C9; Section 12.1	The guidance in Section 12 of NEI 00-04 refers to the need to update the risk information and categorization process if the categorization results are "...more than minimally affected." The NRC staff understands that being "more than minimally affected" would include a situation in which there is indication that an SSC that is	Conforms	SNC's understanding is in line with NRC staff's understanding.

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	categorized as low safety significant would be changed to safety-significant. The NRC staff also recognizes that the licensee or applicant may change the categorization and/or treatment aspects of SSCs so that there is reasonable confidence that the cumulative risk increase from implementing §50.69 is maintained acceptably small.		
C9; Section 12.4	The guidance in Section 12.4 of NEI 00-04 defines CCF as "...the simultaneous failure of more than one SSC to perform its function, due to the same cause...," and Appendix B to NEI 00-04 provides a similar definition, but with "simultaneous" replaced by "during a short period of time." These definitions are derived from their use in a PRA context, where the emphasis is on failure of more than one SSC during a specified mission time. The staff notes that the licensee's or applicant's corrective action program associated with the implementation of §50.69 should address the potential for SSC failures at different times resulting from a common cause, even if they are revealed at different times. In addition, the staff notes that the guidance in Section 12.4 that potential adverse trends need not be evaluated until the number of expected failures for a group of SSCs doubles may not be practical for SSCs with low failure rates assumed in the PRA.	Conforms (via Implementation Process)	SNC has not developed treatment guidance at the present time as the LAR submittal per 10 CFR 50.69 (b)(2)(i) - (iv) does not require licensee to address treatment guidance in the LAR. When SNC develops treatment guidelines, SNC will address this comment.
D	IMPLEMENTATION To evaluate licensee compliance with the requirements of §50.69 for the categorization of SSCs, NRC will use either the methods described and/or endorsed in this	N/A	N/A

APPENDIX A Regulatory Guide (RG) 1.201 Revision 1 Trial Version Conformance Matrix			
RG Section	**RG Position**	**Vogtle Position**	**Comments**
	guide **OR** NRC approved alternative method(s) (that were submitted as part of the licensee's LAR).		

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request

Enclosure 2

Operating Licenses Marked-up Pages

Amendment Number	Additional Condition	Implementation Date
102	The licensee will implement all applicable crane, load path and height, rigging and load testing guidelines of NUREG-0612 and ANSI Standard B30.2, as described in the licensee's letters dated September 4, 1997, May 19 and June 12,1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998	Before and during reracking operations, as appropriate.
154	Upon implementation of the Amendment adopting TSTF-448, Revision 3, the determination of CRE unfiltered air inleakage as required by SR 3.7.10.5, in accordance with TS 5.5.20.c.(i), and the measurement of CFE pressure as required by Specification 5.5.20.d, shall be considered met. Following implementation: (a) The first performance of SR 3.7.10.5, in accordance with Specification 5.5.20.c.(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful tracer gas test, as stated in the June 16, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. (b) The first performance of the periodic assessment of CRE habitability, specification 5.5.20.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful tracer gas test, as stated in the June 16, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. The first performance of the periodic measurement of CRE pressure, specification 5.5.20.d, shall be within 18 months, plus the 138 days allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.	As stated in the Additional Condition

Insert 1 here

Insert 1

Amendment Number	Additional Condition	Implementation Date
	Southern Nuclear Operating Company (SNC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the licensee amendment request dated _____ (and supplements dated _____) and as approved in the safety evaluation report dated _____ (and supplements dated _____). NRC prior approval is required for a change to a categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment).	As stated in the Additional Condition

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-81

Amendment Number	Additional Condition	Implementation Date
78	The licensee shall implement a procedure that will prohibit entry into an extended Emergency Diesel Generator Allowed Outage time (14 days), for scheduled maintenance purposes, if severe weather conditions are expected, as described in the licensee's application dated January 22, 1998, as supplemented by letter dated March 18, 1998, and evaluated in the staff's Safety Evaluation dated May 20, 1998.	Prior to implementation of Amendment No. 78.
80	The UFSAR will be updated to include the heat load that will ensure the temperature limit of 170°F will not be exceeded, as well as the requirement to perform a heat load evaluation before transferring irradiated fuel to either pool, as described in the licensee's letters dated September 4, 1997, May 19 and June 12, 1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998.	To be included in the next appropriate UFSAR update following the installation of the Unit 1 spent fuel racks.
135	Upon implementation of the Amendment adopting TSTF-448, Revision 3, the determination of CRE unfiltered air inleakage as required by SR 3.7.10.5, in accordance with TS 5.5.20.c.(i), and the measurement of CFE pressure as required by Specification 5.5.20.d, shall be considered met. Following implementation: (a) The first performance of SR 3.7.10.5, in accordance with Specification 5.5.20.c.(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful tracer gas test, as stated in the June 16, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. (b) The first performance of the periodic assessment of CRE habitability, specification 5.5.20.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful tracer gas test, as stated in the June 16, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. The first performance of the periodic measurement of CRE pressure, specification 5.5.20.d, shall be within 18 months, plus the 138 days allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.	As stated in the Additional Condition

Insert 2 here

Insert 2

Amendment Number	Additional Condition	Implementation Date
	Southern Nuclear Operating Company (SNC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the licensee amendment request dated _____ (and supplements dated _____) and as approved in the safety evaluation report dated _____ (and supplements dated _____). NRC prior approval is required for a change to a categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment).	As stated in the Additional Condition

Vogtle Electric Generating Plant
Pilot 10 CFR 50.69 License Amendment Request

Enclosure 3

Operating Licenses Clean Typed Pages

Amendment Number	Additional Condition	Implementation Date
102	The licensee will implement all applicable crane, load path and height, rigging and load testing guidelines of NUREG-0612 and ANSI Standard B30.2, as described in the licensee's letters dated September 4, 1997, May 19 and June 12,1998, and evaluated in the staff's Safety Evaluation dated June 29, 1998	Before and during reracking operations, as appropriate.
154	Upon implementation of the Amendment adopting TSTF-448, Revision 3, the determination of CRE unfiltered air inleakage as required by SR 3.7.10.5, in accordance with TS 5.5.20.c.(i), and the measurement of CFE pressure as required by Specification 5.5.20.d, shall be considered met. Following implementation: (a) The first performance of SR 3.7.10.5, in accordance with Specification 5.5.20.c.(i), shall be within the specified frequency of 6 years, plus the 18-month allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful tracer gas test, as stated in the June 16, 2004 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years. (b) The first performance of the periodic assessment of CRE habitability, specification 5.5.20.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful tracer gas test, as stated in the June 16, 2004 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years. The first performance of the periodic measurement of CRE pressure, specification 5.5.20.d, shall be within 18 months, plus the 138 days allowance of SR 3.0.2, as measured from March 23, 2004, the date of the most recent successful pressure measurement test, or within 138 days if not performed previously.	As stated in the Additional Condition
	Southern Nuclear Operating Company (SNC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the licensee amendment request dated _____ (and supplements dated _____) and as approved in the safety evaluation report dated _____ (and supplements dated _____). NRC prior approval is required for a change to a categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment).	As stated in the Additional Condition

Amendment Number	Additional Condition	Implementation Date
	Southern Nuclear Operating Company (SNC) is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) specified in the licensee amendment request dated _____ (and supplements dated _____) and as approved in the safety evaluation report dated _____ (and supplements dated _____). NRC prior approval is required for a change to a categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment).	As stated in the Additional Condition