



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

December 17, 2014

Mr. C. R. Pierce  
Regulatory Affairs Director  
Southern Nuclear Operating Company, Inc.  
Post Office Box 1295, Bin - 038  
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – ISSUANCE OF  
AMENDMENTS RE: USE OF 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 173 to Renewed Facility Operating License NPF-68 and Amendment No. 155 to Renewed Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2 (VEGP), respectively, in response to your letter dated August 31, 2012, as supplemented by letters dated May 17, July 2 and September 13, 2013, and May 2, July 22, and August 11, 2014.

The amendments revise the licensing basis for the VEGP by adding license conditions that allow for the voluntary implementation of the regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." As stated in 10 CFR 50.69, a licensee may voluntarily comply with 10 CFR 50.69 as an alternative to compliance with the following requirements for certain structures, systems and components after it submits and the NRC approves an application for license amendment: (i) 10 CFR part 21; (ii) a portion of 10 CFR 50.46a(b); (iii) 10 CFR 50.49; (iv) 10 CFR 50.55(e); (v) certain requirements of 10 CFR 50.55a, (vi) 10 CFR 50.65, except for paragraph(a)(4); (vii) 10 CFR 50.72; (viii) 10 CFR 50.73; (ix) Appendix B to 10 CFR part 50, (x) certain containment leakage testing requirements, and (xi) certain requirements of Appendix A to 10 CFR part 100.

C. R. Pierce

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A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,



Robert Martin, Senior Project Manager  
Plant Licensing Branch II-1  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:

1. Amendment No. 173 to NPF-68
2. Amendment No. 155 to NPF-81
3. Safety Evaluation

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 173  
Renewed License No. NPF-68

1. The U. S. Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Renewed Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 31, 2012, May 17, July 2 and September 13, 2013, and May 2, July 22, and August 11, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, by Amendment No. 173, Renewed Facility Operating License No. NPF-68 is hereby amended to authorize implementation of the provisions of Title 10 of the *Code of Federal Regulations*, Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," as set forth in the licensee's application dated August 31, 2012, as supplemented by letters dated May 17, July 2, and September 13, 2013, and May 2, July 22, and August 11, 2014, and evaluated in the NRC staff's safety evaluation dated December 17, 2014.
3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert Pascarelli, Chief  
Plant Licensing Branch II-1  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Date of Issuance: December 17, 2014



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 155  
Renewed License No. NPF-81

1. The U. S. Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Renewed Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated August 31, 2012, May 17, July 2 and September 13, 2013, and May 2, July 22, and August 11, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, by Amendment No. 155, Renewed Facility Operating License No. NPF-81 is hereby amended to authorize implementation of the provisions of Title 10 of the *Code of Federal Regulations*, Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," as set forth in the licensee's application dated August 31, 2012, as supplemented by letters dated May 17, July 2, and September 13, 2013, and May 2, July 22, and August 11, 2014, and evaluated in the NRC staff's safety evaluation dated December 17, 2014.
  
3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Robert Pascarelli, Chief  
Plant Licensing Branch II-1  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Date of Issuance: December 17, 2014

ATTACHMENT

TO LICENSE AMENDMENT NO. 173

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 155

RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages

License

License No. NPF-68, page 5  
Appendix D, page D-2  
License No. NPF-81, page 4  
Appendix D, page D-2

Insert Pages

License

License No. NPF-68, page 5  
Appendix D, page D-2  
License No. NPF-81, page 4  
Appendix D, page D-2

7. Spent fuel pool mitigation measures

- (c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
  2. Dose to onsite responders

(11) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 173, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

- D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 5.

An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1967, issued August 21, 1986, and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in item band c above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

- E. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.
- F. GPC shall comply with the antitrust conditions delineated in Appendix C to this license.



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 830.2, 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	
154	<p>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:</p> <p>(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.</p> <p>(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.</p> <p>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 pressure measurement</p>	
173	<p>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 submittals dated August 31, 2012, May 17, 2013, July 2, 2013, September 13, 2013, May 2, 2014, July 22, 2014 and August 11, 2014.</p> <p>The licensee shall implement the items listed in enclosure 1, Implementation items of SNC letter NL-14-0960, dated July 22, 2014, prior to categorizing systems under the process.</p> <p>NRC prior approval, under 10 CFR 50.90, 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 approach).</p>	As stated in the Additional Condition

successfully demonstrated prior to the time and condition specified below for each;

- a) DELETED
- b) DELETED
- c) SR 3.8.1.20 shall be successfully demonstrated at the first regularly scheduled performance after implementation of this license amendment.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

- (a) Fire fighting response strategy with the following elements:
  - 1. Pre-defined coordinated fire response strategy and guidance
  - 2. Assessment of mutual aid fire fighting assets
  - 3. Designated staging areas for equipment and materials
  - 4. Command and control
  - 5. Training of response personnel
- (b) Operations to mitigate fuel damage considering the following:
  - 1. Protection and use of personnel assets
  - 2. Communications
  - 3. Minimizing fire spread
  - 4. Procedures for implementing integrated fire response strategy
  - 5. Identification of readily-available pre-staged equipment
  - 6. Training on integrated fire response strategy
  - 7. Spent fuel pool mitigation measures
- (c) Actions to minimize release to include consideration of:
  - 1. Water spray scrubbing
  - 2. Dose to onsite responders

(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 155, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

- D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption bare identified in Section 6.2.6 of SSER 8.

Amendment Number	Additional Condition	Implementation Date
155	<p>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 submittals dated August 31, 2012, May 17, 2013, July 2, 2013, September 13, 2013, May 2, 2014, July 22, 2014 and August 11, 2014. .</p> <p>The licensee shall implement the items listed in enclosure 1, Implementation items of SNC letter NL-14-0960, dated July 22, 2014, prior to categorizing systems under the process.</p> <p>NRC prior approval, under 10 CFR 50.90, 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 approach).</p>	As stated in the Additional Condition



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 173 TO RENEWED FACILITY OPERATING LICENSE NPF-68

AND

AMENDMENT NO. 155 TO RENEWED FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated August 31, 2012, as supplemented by letters dated May 17, July 2, and September 13, 2013, and May 2, July 22, and August 11, 2014 (References 1 – 6 and 28), Southern Nuclear Operating Company, Inc. (SNC, the licensee) submitted a license amendment request (LAR) regarding the Vogtle Electric Generating Plant Units 1 and 2 (VEGP). The U. S. 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 review of this issue at the VEGP (Reference 29).

The amendments would revise the licensing basis for the VEGP, by adding license conditions that allow for the voluntary implementation of the regulation in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors." As stated in 10 CFR 50.69, a licensee may voluntarily comply with 10 CFR 50.69 as an alternative to compliance with the following requirements for certain structures, systems and components after it submits and the NRC approves an application for license amendment: (i) 10 CFR part 21; (ii) a portion of 10 CFR 50.46a(b); (iii) 10 CFR 50.49; (iv) 10 CFR 50.55(e); (v) certain requirements of 10 CFR 50.55a; (vi) 10 CFR 50.65, except for paragraph(a)(4); (vii) 10 CFR 50.72; (viii) 10 CFR 50.73; (ix) Appendix B to 10 CFR part 50; (x) certain containment leakage testing requirements; and (xi) certain requirements of Appendix A to 10 CFR part 100.

In its submittal, SNC describes the risk-informed categorization process that SNC proposes to use at the VEGP to categorize SSCs into high safety significant and low safety significant components. This categorization would permit the licensee to remove SSCs of low safety significance from the scope of the special treatment requirements identified in 10 CFR 50.69 and to revise requirements for SSCs of greater safety significance. SNC's categorization process is based on the guidance in Nuclear Energy Institute (NEI) 00-04, "10 CFR 50.69 SSC

Enclosure 3

Categorization Guideline" (Reference 8), endorsed by NRC Regulatory Guide (RG) 1.201, Revision 1, "Guidelines For Categorizing Structures, Systems, And Components In Nuclear Power Plants According To Their Safety Significance, For Trial Use" (Reference 7).

The licensee developed plant-specific processes and approaches by applying the categorization process to three systems: the Chemical and Volume Control System, the Containment Spray System, and the Radiation Monitoring System.

The licensee proposed to amend Appendix D, Additional Conditions, Renewed Facility Operating License No. NPF-68 and No. NPF-81 to add a license condition noting the licensee is approved by the NRC to implement 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures Systems and Components for Nuclear Power Reactors." The license condition references the attachment to letter dated July 22, 2014 (Reference 6), which identifies seven modifications (i.e., implementation items) that the licensee has made or will make to its categorization risk-assessment models, approach, or methods before categorizing additional systems.

The supplemental letters dated May 17 and July 2, 2013, provided additional information clarifying the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 2, 2014 (79 FR 52067).

## 2.0 REGULATORY EVALUATION

On November 22, 2004, the NRC added to its regulations 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" dealing with the risk-informed categorization and treatment of SSCs for nuclear power plants (69 FR 68008). This section permits power reactor licensees and license applicants to implement an alternative regulatory framework with respect to "special treatment." Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design basis functions. Implementation of 10 CFR 50.69 requires that licensees first categorize safety-related and non-safety-related SSCs according to their safety significance. SSCs are classified into high safety significant (HSS) and low safety significant (LSS) SSCs. Alternative treatments per 10 CFR 50.69(b)(1) and 10 CFR 50.69(d) can then be applied consistent with the categorization of the SSCs.

In May 2006 the NRC issued RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, And Components in Nuclear Power Plants according to Their Safety Significance, For Trial Use" (Reference 7). RG 1.201 endorses a categorization method, with conditions, described in NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," (Reference 8). NEI 00-04 describes in detail a process for determining the safety significance of SSCs and for categorizing them into the four risk-informed safety class (RISC) categories defined in 10 CFR 50.69. This categorization process uses an integrated decision-making process, incorporating both risk and traditional engineering insights. NEI 00-04 guidance allows licensees to implement different approaches, depending on the scope of their probabilistic risk assessment (PRA). It allows the use of non-PRA type evaluations when PRAs have not been performed. These non-PRA type evaluations include fire-induced vulnerability evaluation (FIVE), seismic margin

analysis (SMA), and guidance in Nuclear Management and Resource Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 15), to address shutdown operations. Such non-PRA type evaluations yield conservative results because all SSCs relied upon in the evaluations will be categorized as HSS.

RG 1.201 states that the applicant is expected to document, as a minimum, the technical adequacy of the internal initiating events PRA. Either PRAs or alternative approaches for hazards other than internal initiating events may be used. One acceptable approach to determining the technical adequacy of a PRA is contained in RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 9). Regulatory Guide 1.200, Revision 2 endorses, with clarifications, the American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009 (ASME/ANS 2009 Standard) (Reference 12). The ASME/ANS 2009 Standard addresses internal events and other hazards.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 11), provides guidance on the use of PRA findings and risk insights in support of licensee requests for changes to a plant's licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations.

### 3.0 TECHNICAL EVALUATION

In its submittal, the licensee describes the risk-informed categorization process that it proposes to use at the VEGP to categorize SSCs into HSS and LSS components based on safety significance. This section of the safety evaluation documents the NRC staff's review of whether the licensee's 50.69 categorization process satisfies the requirements of 10 CFR 50.69(c). The review has the following main objectives: (1) verify conformance of the licensee's categorization process with the relevant NRC-endorsed guidance; and (2) validate that the quality of the licensee's PRA is adequate for use in the application.

#### 3.1 Overview of the Categorization Process

As described in the LAR, the licensee's categorization process contains the elements/steps listed below, which are reviewed in detail in the following subsections of this safety evaluation.

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary HSS or LSS based on the seven questions in Section 9 of NEI 00-04 (see Section 3.2). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.
- Component safety significance assessment. Safety significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

- Assessment of defense in depth (DID) and safety margin. Components that are categorized as preliminary LSS are evaluated for their role in providing defense-in-depth and safety margin and, if appropriate, upgraded to HSS.
- Review by the Integrated Decision-making Panel (IDP). The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.
- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of RG 1.174.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.

NEI 00-04 describes alternative methods for some of the elements. For example, seismic events may be included in the categorization using a seismic PRA or a seismic margin analysis. The draft procedures submitted by the licensee as an enclosure to the May 17, 2013, Request for Additional Information (RAI) response contained text from NEI 00-04 that described all the different alternatives implying that any alternative could be used for any system. In RAI No. 2 the NRC staff requested clarification about which alternatives had been applied to the systems that had been categorized, and which alternatives were the licensee requesting authorization to use. In response to RAI No. 2 the licensee clarified that it was only requesting authorization for those methods that had been used in the systems that had been categorized, and will remove the descriptions of unused alternatives from the procedures. These methods for which the licensee is requesting authorization to use are as follows: Internal PRA to assess internal risk; Fire PRA to assess fire risk; Seismic Margin Analysis (SMA) to assess seismic risk; Individual Plant Examination of External Events (IPEEE) screening to assess the risk from other external hazards (high winds, external floods); and Shutdown Safety Plan to assess shutdown risk. Removal of the unused alternatives from the procedures is included as Implementation Item No. 1, which according to the license condition, must be implemented prior to the licensee categorizing systems under the process. Therefore, the NRC staff finds that only methods that have been reviewed and determined to be acceptable by the NRC will be used. The license condition also states that any changes to the approved methods must be submitted to the NRC staff under 10 CFR 50.90 for prior review and approval.

In LAR Section 3.1.3, the licensee provided three clarifications regarding the licensee's conformance with NEI 00-04. In RAI No. 28, the NRC staff requested additional information regarding Clarification No. 1, which described the licensee's "approach used to risk rank system functions," which appeared to be a departure from the NRC-endorsed guidance. The categorization process described in NEI 00-04 starts with component categorization based on risk, followed by function categorization, followed by the final assessment of the IDP. When a component's function is categorized as HSS based on risk, the associated system function is considered HSS. System functions not addressed by the risk assessment are categorized based on the seven questions in Section 9 of NEI 00-04. Once a function has been identified as HSS,

then all components supporting the function are assigned as preliminary HSS and the IDP must intervene to assign any of these components to LSS. In contrast to NEI 00-04, the licensee's proposed categorization process performs function categorization first based on the seven questions in Section 9 of NEI 00-04, followed by component categorization based on risk, followed by the final assessment of the IDP. In the licensee's proposed process, a system function categorized as LSS based on the qualitative assessment would remain LSS even if the risk assessment categorizes a component associated with that function as HSS. Therefore the licensee's process would have more components categorized as preliminary LSS than the NEI 00-04 method.

In response to RAI No. 28 (Reference 5) the licensee stated that it would revise its process to ensure that if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS. Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The licensee has included Implementation Item No. 6 to revise its categorization process to comply with the response to RAI No. 28. According to the license condition, Implementation Item No. 6 must be completed before the licensee may categorize additional systems and the proposed change will make the licensee's process consistent with the NEI 00-04 guidance. Therefore, the NRC staff finds the licensee response to RAI No. 28 acceptable.

Clarification No. 2 and No. 3 in LAR Section 3.1.3 pertain to mapping of passive components to functions. Clarification No. 2 indicated that components with only a passive function are mapped only to the passive function, as opposed to also mapping them to the active function as specified in the NEI 00-04 guidance. Similarly, clarification No. 3 indicated that only components required to perform an active function are mapped to that active function. This process is not consistent with NEI 00-04, which directs that every component and component function in the function's pathway is assigned to the function. The licensee's process indicates that a passive component (e.g., a pressure retaining component) whose failure could fail an active HSS function can be assigned LSS if the passive categorization process assigns the passive function to the LSS category. In RAI No. 29, the NRC staff inquired about these clarifications. In the response, the licensee confirmed that the failure of a passive component (e.g., motor operated valve body) that supports an HSS active function may be assigned LSS by the passive categorization methodology if confirmed LSS by the IDP. This can occur because, for example, there are no common cause failures (CCF) among passive components (i.e., multiple and simultaneous pipe ruptures are not expected), so an active function may be HSS due to CCF considerations but the individual pressure retaining components whose individual failures do not fail the function can be LSS. The NRC staff finds that risk assessments generally do not consider the very unlikely simultaneous multiple failures of passive components (except for external hazard events impacts that should be included in the external hazard evaluation) and therefore the proposed method is acceptable.

The next safety evaluation sections summarize the NRC staff's review of each element in the licensee's proposed categorization process.



### 3.2 Qualitative Assessment of System Functions

Once a system is chosen for categorization, all of the system functions and all of the components belonging to that system are identified. The system functions are qualitatively categorized using a set of deterministic questions.<sup>1</sup> The deterministic questions used by the licensee in the qualitative categorization of functions correspond to the seven questions provided in Section 9.2.2 of NEI 00-04. The licensee simplified and logically reversed the questions from the NRC-endorsed guidance, such that any answer of "yes" would mean high safety significant. The licensee submitted its draft procedures in response to RAI No. 1 as part of its May 17, 2013, letter (Reference 2). The seven deterministic questions are found in the licensee's draft procedure NMP-ES-065-003, "10 CFR 50.69 Risk Informed Categorization for Structures, Systems and Components," Step 5.10.1, and are as follows:

1. Does failure of the function directly cause an initiating event?
2. Does failure of the function cause a loss of reactor coolant pressure boundary integrity resulting in leakage beyond normal makeup capability?
3. Does failure of the function result in the failure of a basic safety function?
4. Is the function called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means for the successful performance of operator actions required to mitigate an accident or transient? This also applies to instrumentation and other equipment needed to allow the required actions to be performed.
5. Is the function called out or relied upon in the plant Emergency/Abnormal Operating Procedures or similar guidance as the sole means of achieving actions for assuring long-term containment integrity, monitoring of post-accident conditions, or offsite emergency planning activities? This also applies to instrumentation and other equipment needed to allow the required actions to be performed.
6. Does failure of the function prevent the plant from reaching or maintaining safe shutdown conditions and/or is the function significant to safety during mode changes or shutdown? Assume that the plant would be unable to reach or maintain safe shutdown conditions if the function failure results in the need for actions outside of plant procedures or available backup functions/SSCs.
7. Does failure of the function that acts as a barrier to fission product release during plant operation or during severe accidents result in the implementation of off-site radiological protective actions?

In the licensee's process, if the answer to any one of the above questions is "yes" for a function, then that function is considered HSS. The licensee's simplifications from the NEI 00-04 questions removed some explanatory text but retained the core question. The NRC staff finds the questions and process acceptable because the results are equivalent to application of the NRC-endorsed guidance.

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<sup>1</sup> Upon completion of Implementation Item No. 6, functions initially categorized LSS according to the qualitative categorization will be changed to HSS when any supporting component's function is categorized as HSS based on the risk evaluation.

The NRC observed the licensee's IDP deliberation on November 29, 2011. The NRC observations are documented in "Vogtle Electric Generating Plant, Units 1 and 2, Audit Report For The Process Being Developed To Support A License Amendment Request To Implement Risk Informed Categorization Of Systems, Structures, And Components," (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12061A245). In RAI No. 7, the NRC staff requested additional detail about how the IDP reached its decision on the answers to questions 4 and 5 regarding the determination as to whether a function provides the "sole means" of accomplishing a specific mitigation function. In response to the RAI, the licensee clarified that the procedures will be changed to clarify that "[p]lant Vogtle will take credit for alternate means only if the alternate means are proceduralized and included in the Licensed Operator training." The licensee will implement this change prior to categorizing any additional systems as specified by Implementation Item No. 3 referenced in the license condition allowing implementation of 10 CFR 50.69.

The NRC staff finds the licensee's process for the preliminary categorization of functions acceptable because it is based on guidance in NEI 00-04 with additional clarification regarding when "sole means" must be assigned and therefore provides a clear and logical record of how the questions were addressed.

### 3.3 Component Safety Significance Assessment

The next step in the licensee's process is to assess the safety significance of components using quantitative or qualitative risk information from a PRA or other risk assessment methods. In the NEI 00-04 guidance, component risk significance is assessed separately for five hazard groups:

- Internal event risk
- Fire
- Seismic
- Other external risks (tornadoes, external floods)
- Shutdown risks

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For the other risk hazards - fire, seismic, other external hazards (high winds, external floods, etc.), and shutdown - 50.69(b)(2) allows, and the NEI 00-04 guidance summarizes, the use of PRA, if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods (e.g. Fire Induced Vulnerability Evaluation, Seismic Margin Analysis, IPEEE Screening, and Shutdown Safety Plan).

As clarified in response to RAI No. 2, the licensee's categorization process uses PRA to assess risks from internal events (including internal flooding) and from fire. For the other three risk hazard groups, the licensee's process uses non-PRA methods for the risk characterization, as follows:

- SMA to assess seismic risk
- IPEEE Screening to assess the risk from other external hazards (high winds, external floods)
- Shutdown Safety Plan to assess shutdown risk

The methods used by the licensee are consistent with the methods included in the NEI 00-04 guidance and therefore acceptable to the NRC staff. The application of these methods is reviewed in the following safety evaluation subsections: PRA in subsections 3.3.1 and 3.3.2 and the non-PRA methods in subsection 3.3.3.

### 3.3.1 Capability and Quality of the PRA to Support the Categorization Process

The licensee's PRA is comprised of (1) an internal events PRA that calculates CDF and LERF from internal events, including internal flooding, at full power and (2) a fire PRA. Section 50.69(c)(1)(i) requires that the PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC. The licensee has had peer reviews of its internal events and fire PRAs. Section 10 CFR 50.69(b)(2)(iii) requires the results of the PRA review process conducted to meet 10 CFR 50.59(c)(1)(i) be submitted as part of the application. The licensee has submitted this information and therefore the licensee has satisfied the requirements that the PRA be subjected to a peer review process and that the results of that process be submitted in the application.

Section 50.69(e) requires periodic updates to the licensee's PRA and SSC categorization. The licensee's periodic update program is discussed in Section 3.8 of this safety evaluation. The NRC staff finds that changes over time to PRA methods and to the SSC reliabilities are inevitable and such changes are recognized by 10 CFR 50.69(e)'s requirement for periodic updates. Therefore, the NRC staff review of the PRA quality and level of detail reported in this safety evaluation (SE) is directed toward determining whether past PRA reviews have identified systemic weaknesses or errors in the current PRA that should be fixed before additional systems are categorized because they otherwise could have a substantive impact on the PRA results.

#### Internal Events PRA

The NRC staff reviewed the results of the peer review process for the internal events PRA presented in LAR Section 3.3.1. The licensee's internal events PRA model was subject to a number of industry peer reviews and self-assessments. The most recent peer review was performed in May 2009, which, as clarified in response to RAI No. 3, was a full scope peer review against Revision 1 of RG 1.200 (Reference 10). The licensee submitted a list of all facts and observations (F&Os) from the 2009 peer review and provided a disposition of each item for this application in Table 5 of the LAR. LAR Table 6, as revised in the attachment to Reference 5, identifies differences between Revisions 1 and 2 of RG 1.200 and integrates these differences into the findings from the previous peer reviews. Therefore, the NRC staff finds that the licensee's internal events PRA has been assessed against the currently applicable revision (Revision 2) of RG 1.200.

The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed the potential impact of the findings on the categorization. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraphs.

In RAI No. 12, the NRC staff asked for additional information regarding the licensee's resolution of F&O IE-A5 (and related, but mislabeled, IE-A4-01) regarding the lack of documentation proving that "a systematic evaluation of each system to assess the possibility of an initiating event occurring due to failure of the system" was performed. The NRC staff asked the licensee to describe the structured approach and to show that the support systems were included and common cause failures were accounted for in the initiating event analysis. The licensee responded that the initiating events have been reviewed during PRA updates with a systematic evaluation that examined new information from the nuclear industry and the NRC; current plant design and operations; and plant-specific events. After identifying candidate initiating events, they were screened and grouped based on site-specific support analysis and the current plant design and the new generic information. Plant-specific events were reviewed to determine any reactor trip events or precursor that could result in any unique initiating events that may not have been identified previously. After compiling the list, a final check was performed by a three-member site review team. Additionally, the licensee specified that a review of failure effects of all supporting systems has been performed in order to identify a list of special initiating events. The NRC staff finds the licensee's evaluation capable of identifying potential initiating events and therefore acceptable.

In RAI No. 13, the NRC staff asked for additional information regarding the licensee's resolution of F&O IF-C2a-01, which stated that "successful mitigation of ALL flood events is assumed to occur 30 minutes into any flood scenario." The licensee responded that currently "[t]he flooding scenario frequency of two scenarios initiated by pipe rupture was reduced by a factor of 0.1 (screening value) by crediting a human action. The remaining flooding scenarios initiated by a pipe rupture assumed no credit for flooding isolation." Human actions were credited following rupture in an auxiliary component cooling water (ACCW) pipe and rupture in the discharge pipe of the chemical and volume control system (CVCS). The licensee reported using a screening value of 0.1 for failure of a human action to start a standby charging pump to mitigate the flooding induced loss of normal charging. The NRC staff finds the use of a screening value in the two scenarios acceptable because the operators would attempt to mitigate a flood scenario, a value of 0.1 for failure to succeed is reasonable, and the scenarios are not dominant scenarios.

In RAI No. 14, the NRC staff requested additional information regarding the licensee's resolution of F&O SY-B3-01, which stated that some systems "may be lacking common cause failure (CCF) grouping." The licensee's resolution was that these systems are non-risk significant and therefore would not impact categorization results. The NRC staff asked the licensee to describe the criteria used to classify these systems as non-risk-significant and the basis for concluding that they would remain non-risk-significant had CCF been accounted for. The licensee responded that this F&O is related to (and limited to) not modeling common cause failure to run for the pumps of the following two non-safety related supporting systems: turbine plant cooling water (TPCW) and turbine plant closed cooling water (TPCCW). The licensee used engineering judgment and sensitivity analyses to show minimal impact on the PRA results. The NRC staff finds the response acceptable because the licensee has identified and evaluated the missing common cause failures and determined that they have a minimal impact on the PRA results.

A number of F&Os from the 2009 peer review were not resolved at the time of the submittal. All the unresolved F&Os pertain to PRA documentation that was identified as confusing, missing or incomplete. In RAI No. 11, the NRC staff asked the licensee to summarize the process used to ensure that the documentation of the PRA is of sufficient clarity and quality to support the

long-term, continuous use of the PRA as it is necessary for the implementation of 10 CFR 50.69. The licensee responded that the recommendations from the peer review were included in the updated documentation. Additionally, during the course of the PRA update, additional documentation items were improved as the documentation process of preparation, review, and approval was completed. The NRC staff finds the resolution acceptable because the licensee has improved documentation as a result of the peer review, and the periodic review process required by 10 CFR 50.69(e) should identify remaining weaknesses in documentation through the periodic use of the PRA models.

In RAI No. 30, the NRC staff requested the status of three issues: (1) crediting offsite recovery within four hours without considering the impact of earlier consequential failures that could lead to unrecoverable scenarios; (2) crediting battery DC power for a longer time than the design; and (3) use of a human error probability estimation method without apparently fully exercising the attributes used to apply the method to time sensitive activities. The licensee evaluated each of these issues (Reference 5) and determined that the impact on the PRA results, and therefore on the SSC categorization, is expected to be negligible. However the licensee did not change the PRA to resolve these issues but proposed Implementation Item No. 7 to complete the analysis and incorporate any changes to the PRA prior to any additional categorization activities. The NRC staff finds the resolution acceptable because any changes to the PRA arising from addressing these issues will be incorporated into the PRA thereby confirming the expected negligible impact before additional systems are categorized and will become part of the periodic re-evaluation for the few systems already categorized if the impact is not confirmed negligible.

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA that must model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200 provides guidance for determining the technical adequacy of the PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 using a peer review process. SNC has followed the guidance and submitted the results of the peer review. The NRC staff has reviewed the peer review results and the licensee's resolution of the results and finds that the quality and level of detail of the PRA is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201. Therefore, the NRC staff concludes that the categorization process satisfies the internal events PRA requirement.

#### Fire PRA

The NRC staff reviewed the results of the peer review process for the fire PRA presented in LAR Section 3.3.2. The licensee's fire PRA was subject to a full scope industry peer review in 2012 against RG 1.200, Rev. 2 (Reference 9) and NEI 07-12 (Reference 25), followed, as clarified in the licensee's response to RAI No. 4, by a focused peer review. The NRC staff reviewed the licensee's resolution of all the peer review findings and considered the potential impact of the findings on the categorization and the clarity of the reported resolution. The NRC staff requested additional information to clarify the licensee's disposition for some of the findings as described in the following paragraphs.

In RAI No. 9, the NRC staff asked the licensee to summarize the review process and the qualifications of the personnel that reviewed the resolution to the F&Os to determine the post-resolution category of each supporting requirement. The licensee responded that the qualifications of the personnel that reviewed the resolutions are similar to those qualifications specified for the peer reviewers in the PRA standard (Reference 12), specifically, personnel knowledgeable of the PRA standard requirements and experienced in performing the applicable PRA activities. The NRC staff finds the licensee's response acceptable.

The full scope peer review team did not assess 25 supporting requirements (SRs) deemed to be "Not Applicable." In RAI No. 9, the NRC staff asked the licensee to summarize the 25 SRs deemed not applicable and provide the criteria used to make this determination. The licensee provided a justification for each of the 25 SRs deemed not applicable. In all cases, the analysis scenarios or assumptions specified in the SR were not used by the licensee in its fire PRA analysis. Therefore, the NRC staff agrees with the licensee's conclusions that these SRs are not applicable.

The licensee submitted a list of all F&Os from the fire PRA peer reviews and provided a disposition of each item in Table 8 of the LAR. The NRC staff reviewed the licensee's resolution of the peer review F&Os. The NRC staff asked for additional information regarding the licensee's resolution of a number of specific F&Os. These RAIs are discussed below.

In RAI No. 10, the NRC staff asked the licensee to clarify the process used for addressing the errors in the fire PRA identified through peer reviews. In furtherance of its goal of reviewing the quality of the PRA, the NRC staff needs to ensure that the errors identified are not the result of systematic errors that would put into question the quality of and undermine the confidence in the PRA. The licensee responded that an assessment was performed and demonstrated that the errors identified by the peer review team were of isolated nature. The licensee added that it performed a complete review of the PRA based on the insights from peer review. The steps taken during the review included understanding the technical basis of the F&O, identifying and resolving the specific error presented in the F&O, and reviewing the PRA model subject to the identified error for correctness. The NRC staff finds this response acceptable because the licensee took the necessary steps to ensure that the errors identified were not of systematic nature and eliminated any significant errors from the model.

In RAI No. 15, the NRC staff requested clarification on the licensee's resolution of finding PP-A1-01 which identified that the supporting requirements PP-A1 and PP-C2 were not met. The peer review team identified several structures within the protected area that were identified within the global analysis boundary, but then screened from further analysis, and the justification for screening appeared to be inadequate or incorrect. In response, the licensee provided a list of these locations and explained for each location whether the structure was added to the model or, if not, why not. The NRC staff finds the licensee's response acceptable and agrees that the licensee has resolved the issue identified.

In RAI No. 16, the NRC staff requested clarification on the licensee's resolution of finding FSS-A1-01. This finding noted that some ignition sources appeared to have been screened out of the fire analysis but no supporting documentation was located. The resolution provided in the LAR indicated that justification on all screened sources was added to the documentation. The response to RAI No. 16 explained that some ignition sources that had a limited impact on the plant

had been excluded. The NRC staff finds that the most risk significant ignition sources have been included although some less risk significant sources (i.e., those that are expected to be low safety significant) may be excluded. Risk importance measures are relative to the total estimated risk and excluding some small measure of the total risk will have a negligible impact or will increase the relative importance of the remaining sources. The NRC staff finds the resolution acceptable for SSC categorization because any low safety significant scenarios excluded will have a negligible impact on the categorization.

In RAI No. 17, the NRC staff requested clarification on the licensee's resolution to finding FSS-A5-01 which identified that supporting requirement FSS-A5 was not met. FSS-A5-01 stated that transient ignition sources did not appear to be postulated in all possible locations. The NRC staff asked the licensee to summarize how transient locations were selected and how the process is consistent with the process of locating transients under pinch points in NUREG/CR-6850 (Reference 23). The NRC staff also requested that the licensee summarize how hot work fires are located and a frequency is assigned. The licensee responded that transient fires were postulated at locations consistent with the guidance of NUREG/CR-6850. That is, transient fires were postulated at locations where fire PRA targets would be postulated to be damaged by transient ignition sources. Transient fire locations were not limited to only "pinch points" as defined by NUREG/CR-6850. The transient fire locations were selected by performing plant walkdowns or review of plant raceway drawings for locations in which a walkdown was not performed. Area accessibility was not used as a criterion when selecting transient fires. Transient fires in the control room were selected at locations with fire PRA targets consistent with the discussion above. Similarly, hot work fires were postulated at locations where fire PRA targets would be postulated to be damaged by hot work activity. The frequency was determined by the total hot work frequency of the plant area apportioned to the postulated hot work fire based on the postulated target damage. The NRC staff finds the licensee's response acceptable and agrees that the licensee has resolved the identified issue.

In RAI No. 18, the NRC staff inquired about F&O FSS-A5-02 which identified that supporting requirement FSS-A5 was not met. The finding description noted that the sum of the ignition frequencies in some physical area units appeared to differ from the expected value. The licensee clarified that after resolving other related F&Os, the sum of postulated fire ignition frequencies match the expected frequency. The NRC staff finds the licensee's response acceptable and agrees that the licensee has resolved the identified issue.

In F&O FSS-C7-01, the peer review team identified missing dependencies between manual and automatic suppression activities in the multi-compartment analyses and in the hot gas layer scenarios. The original licensee disposition to this finding specified that it corrected dependencies in the multi compartment analyses, but there was no mentioning of the hot gas layer scenarios. In response to RAI No. 20, the licensee clarified that only manual suppression is credited in the hot gas layer fire scenarios, there was no automatic suppression credited and therefore no dependencies between suppression activities. The NRC staff finds the licensee's response acceptable and agrees that the licensee has resolved the identified issue.

In F&O FSS-E3, the peer review team noted that parameter uncertainty was not propagated through the PRA as required by the PRA standard. The licensee responded that all parameters that can be propagated have now been propagated which is sufficient to meet capability category (CC) I. In RAI No. 22, the NRC staff asked the licensee to justify the use of CC I for the

categorization process and to address the potential effects of uncertainty on the final safety-significance categories for SSCs. The licensee stated that the propagated uncertainty values are not used in the importance measures to determine safety significance and that the impact of uncertainty is adequately addressed by the fire PRA sensitivity studies defined in Section 5.2 of NEI 00-04 (reviewed in Section 3.3.2 of this safety evaluation), which provide a method to identify components that might become more risk significant when the most uncertain PRA inputs are varied. The NRC staff agrees with the licensee's observations and therefore this issue is resolved.

F&O FSS-G4-01 related to FSS-G4 and FSS-G5 noted that no justification was provided for crediting non-rated or active barriers. The licensee's response stated that justification was developed. In RAI No. 23, the NRC staff asked the licensee to specify what credit was taken and how it was developed. The licensee responded that it used the NRC endorsed guidance in NUREG/CR-6850 (Reference 23). The licensee applied a screening value of 0.1 to non-rated barriers and the failure probabilities from NUREG/CR-6850 Table 11-3 were used for the fire dampers and the fire doors active barriers. The NRC staff finds the licensee's response acceptable because NRC-endorsed guidance was used and agrees that the licensee has resolved the identified issue.

In F&O FQ-B1-04, the peer review team noted that the probabilities of consequently failed basic events were set to 1.0 instead of set to logical TRUE. The licensee's resolution in the LAR discussed a sensitivity study to evaluate the impact of this observation. RAI No. 24 requested the licensee to evaluate the potential impact of this simplification on the importance measures. In response to the RAI, the licensee clarified that the PRA model did set the failure basic event to logical TRUE and corrected the description provided in the LAR with respect to resolution of this F&O. The NRC staff finds the licensee's response acceptable and agrees that the licensee has resolved the identified issue.

F&O FSS-D1 related to SRs FSS-D4, FSS-D11, and FSS-H4, which identified the use of a heat release rate of 69kW for transient fires that was not endorsed by NUREG/CR-6850. In response to RAI No. 21, the licensee stated that it used the method endorsed by the NRC in a letter dated June 21, 2012 (Reference 24). The NRC staff finds the licensee's response acceptable and agrees that the licensee has resolved the identified issue.

In RAI No. 25, the NRC staff asked the licensee to provide the sensitivity analysis of the impact that is part of the method of using the fire ignition frequencies from Supplement 1 of NUREG/CR-6850 (Reference 23). The sensitivity study directs that the risk be recalculated using the fire ignition frequencies provided in Table 6-1 of NUREG/CR-6850 and consider additional defense-in-depth measures if associated guideline values are exceeded. The licensee's analysis showed that when CDF and LERF results are combined, the Unit 1 sensitivity run using fire ignition frequencies provided in Table 6-1 of NUREG/CR-6850 did not identify additional components meeting the NEI 00-04 HSS criteria compared to the base case run. The Unit 2 sensitivity run identified only one additional component meeting the importance measure criteria for HSS components compared to the base case run. The NRC staff finds the use of the NUREG/CR-6850 Supplement 1 fire ignition frequencies instead of the Table 6-1 frequencies acceptable because defense-in-depth is already systematically considered in the categorization process and the sensitivity study results indicate a minimal effect on the importance measures and subsequent categorization.



In RAI No. 26, the NRC staff asked the licensee to perform a sensitivity study to quantify the impact of Control Power Transformers (CPT) credit on SSC categorization. Interim Technical Guidance provided in a Memorandum from Richard P. Correia, Office of Nuclear Regulatory Research, NRC, to Joseph G. Giitter, Office of Nuclear Reactor Regulation, NRC, titled "Interim Technical Guidance on Fire Induced Circuit Failure Mode Likelihood Analysis," dated June 14, 2013 (Reference 26), indicates that the effect of any CPTs reduction to the hot short-induced spurious operation likelihood could not be substantiated. The licensee's sensitivity analysis showed that when CDF and LERF results are combined, the Unit 1 CPT sensitivity run identified fifteen additional components as HSS and the Unit 2 CPT sensitivity run identified nine additional components as HSS compared to the base case runs. From these, four HSS components were unit equivalent, i.e., the corresponding component in Unit 1 and in Unit 2 met the HSS criteria. The licensee concluded that eliminating the CPT credit would identify approximately twenty more components as HSS. This sensitivity study indicates that CPT credit has the potential to substantively impact the results. The interim technical guidance was superseded by NUREG/CR-7150, Vol. 2, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)" (Reference 27) which is supported by the "Supplemental Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis" (ADAMS Accession Nos. ML14086A165 and ML14017A135). In response, the licensee provided Implementation Item No. 4, which states that the fire PRA model will be updated to include the JACQUE-FIRE guidance and this implementation item will be completed before additional SSCs are categorized. The NRC staff finds this response acceptable because changes to the PRA methods are expected to periodically occur and the new guidance will become part of the PRA before additional systems are categorized as well as part of the periodic re-evaluation for the few systems already categorized.

In RAI No. 27, the NRC staff asked the licensee to identify and provide justification for any fire PRA methodology that has not been formally accepted by the NRC staff. The licensee identified the following unendorsed fire PRA methods:

- Not using 0.001 as lowest value for failure of manual suppression (using values less than 0.001, even 0);
- Not using lower failure threshold for sensitive electronics;
- Using electrical cabinet heat release rate and severity factors values and cabinet to cabinet fire propagation methods different than values and methods accepted by the NRC staff; and
- Using an alignment factor value for oil pump fires different than the value accepted by NRC staff.

The licensee provided Implementation Item No. 5 which states that the four methods will be replaced with acceptable methods before additional SSCs are categorized and therefore the NRC staff finds this issue resolved. The periodic review process includes maintenance and update to the PRA and the changes made to the PRA methods are similar to other changes to the PRA and PRA methods that will occur over time. The NRC staff finds the resolution acceptable because any changes to the PRA arising from addressing these issues will be incorporated into the PRA before additional systems are categorized and will become part of the periodic re-evaluation for the few systems already categorized.

Pursuant to 10 CFR 50.69(c)(1)(i), the categorization process must consider results and insights from a plant-specific PRA. The use of a fire PRA to support categorization is endorsed by RG 1.201. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard that is endorsed by the NRC. RG 1.200 provides guidance for determining the technical adequacy of a fire PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 using a peer review process. SNC has followed the guidance and submitted the results of the peer review. The NRC staff has reviewed the peer review results and the licensee's resolution of the results and finds that the quality and level of detail of the fire PRA is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201. Therefore, the NRC staff concludes that the categorization process satisfies the PRA requirement.

### PRA Quality Conclusions

Section 50.69(c)(1)(i) states:

*Consider results and insights from the plant-specific PRA. This PRA must at a minimum model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process, and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.*

The licensee has used a plant-specific PRA that models severe accident scenarios resulting from internal initiating events and fire initiated events occurring at full power operation. The PRAs have been assessed using peer reviews in accordance with RG.1200, Revision 2 that endorses the ASME/ANS RA-Sa-2009, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications." RG 1.174 states that, [t]he scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process."

The 10 CFR 50.69 categorization process includes the use of quantitative PRA results together with qualitative considerations to categorize SSCs. The NRC staff has reviewed the results of the peer reviews to assess whether the PRA is adequate to support the categorization process. Errors and weaknesses with respect to capability category II that were identified as F&Os during the PRA peer reviews have been submitted to the NRC by the licensee along with the licensee's disposition of each F&O. Significant errors and weaknesses have been resolved or will be resolved before categorization of additional systems with the completion of Implementation Items 4, 5, and 7. Changes to the PRA are expected over time to reflect changes to the plant and PRA methods and data. The licensee has established a periodic update and review process to incorporate plant and PRA changes into the categorization and to check that past categorization remains valid. The NRC staff finds the quality and level of detail of the PRA sufficient to support the licensee's transition to the 10 CFR 50.69 categorization process because the licensee has (1) reviewed the PRA using NRC-endorsed guidance and adequately resolved all identified errors and weaknesses and (2) established a periodic update and review process to update the PRA to incorporate changes made to the plant and PRA methods and data. Therefore, the NRC staff concludes that the licensee's categorization process satisfies the PRA requirement.

### 3.3.2 Importance Measures and Sensitivity Studies

The licensee follows the guidance in NEI 00-04, Section 5.0. Per the guidance, the component safety significance assessment using PRA involves use of importance measures and sensitivity studies for both CDF and LERF. First, the Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) importance measures are obtained for each component and each hazard (i.e., the internal events PRA and separately for the fire PRA) and compared to specified criteria. Then, sensitivity studies for each component and each hazard are performed. Last, integrated importance measures over all hazards are calculated per Section 5.6 of NEI 00-04. In response to RAI No. 6, the licensee summarized the development of importance measures and the use of sensitivity studies to develop the risk based safety-significance category for components modeled in the PRA.

#### Importance Measures

Per NEI 00-04, a component is assigned as HSS if any of the following importance measure criteria are exceeded:

- Sum of F-V for all basic events modeling the SSC of interest, including common cause events > 0.005
- Maximum of component basic event RAW values > 2
- Maximum of applicable common cause basic events RAW values > 20

In the internal events PRA, the component's importance measure includes its contribution to initiating events and its contribution to accident mitigation as described in NEI 00-04. For the fire PRA, only the component's contribution to accident mitigation is included, which is consistent with the guidance in Section 5.2 of NEI 00-04 and therefore acceptable.

In response to RAI No. 6, the licensee indicated that it applied a 10 percent margin to the NEI importance measure thresholds for the fire PRA (i.e.,  $F-V > 0.0045$  instead of 0.005 and  $RAW > 1.80$  instead of 2.0).

The NEI 00-04 guidance recommends that a truncation level of five orders of magnitude below the baseline CDF (or LERF) value should be used for calculating the F-V risk importance measures. The guidance also recommends that the truncation level used should be sufficient to identify all functions with  $RAW > 2$ . The licensee's draft procedure NMP-ES-065-001, "10 CFR 50.69 Active Component Risk Significance Insights," (NMP-ES-065-001) (Reference 2) includes this exact guidance and is therefore consistent with the NEI guidance.

In draft procedure NMP-ES-065-001, the licensee also defines and uses "absolute importance measures." Absolute importance measures are defined in Section 12 of NEI 00-04, endorsed by the NRC in RG 1.201, and are used when re-evaluating previously categorized SSCs with an updated PRA.

The NRC Staff finds that the calculation and use of importance measures by the licensee are consistent with the NRC-endorsed guidance in NEI 00-04 and therefore acceptable.

### Sensitivity Studies

Per the guidance in NEI 00-04, components not identified as HSS using the importance measure criteria above are further evaluated with sensitivity studies. To identify HSS components, the importance measures are obtained for each sensitivity study and compared to the screening criteria. The sensitivity studies determine the impact of the most uncertain parameters in the PRA (i.e., human error, common cause failure, and manual suppression failure probabilities) and the impact of the different at-power operating configurations associated with online maintenance. The methodology also requires that the evaluation of the adequacy of the licensee's PRA be evaluated to identify any other issues that should be addressed with a sensitivity study. The sensitivity studies required in the NEI 00-04 guidance are as follows:

- 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
- Any applicable sensitivity studies identified in the characterization of PRA adequacy
- No credit for manual suppression (fire PRA only)

As indicated in the LAR and clarified in response to RAI No. 6, the licensee performs the risk sensitivity studies recommended in the NEI 00-04 guidance for the internal events PRA and the fire PRA. The first five sensitivity studies are well defined and the licensee performs the studies as stated. In LAR Section 3.2.1.2 and Section 3.2.2.2 the licensee summarizes its evaluation to determine if additional "applicable sensitivity studies" should be developed.

LAR Section 3.2.1.2 describes how the licensee searched for additional issues in the internal events PRA that should be evaluated with a sensitivity study. The licensee used the NRC guidance in NUREG-1855 (Reference 21) supplemented with the EPRI document TR-1016737 (Reference 22) to identify sources of uncertainty in the internal event PRAs. All the assumptions documented in the licensee's PRA were also evaluated to identify additional sources of model uncertainty. The licensee identified three sources of uncertainty in its PRA that have a potential impact on the internal events PRA results: (1) pressure-induced Steam Generator Tube Rupture (SGTR); (2) seasonal impacts on initiating events; and (3) basis for human error probabilities (HEPs). The licensee summarized its evaluations to determine if additional sensitivity studies should be developed. The licensee performed a quantitative evaluation on the impact of alternative assumptions of item (1) and a qualitative evaluation for item (2). The results of the evaluations indicated that these uncertainties have no impact on the categorization and that sensitivity studies need not be developed. The licensee dispositioned item (3) as being fully addressed by NEI 00-04's sensitivity studies on human error basic events. Therefore, the NRC staff finds that the licensee searched for, identified, and evaluated sources of uncertainty in its internal PRA consistent with the guidance to identify additional "applicable sensitivity studies."

LAR Section 3.2.2.2 describes how the licensee searched for additional issues in the fire PRA that should be evaluated with a sensitivity study. Each of the sixteen "Tasks" in the NRC-endorsed methodology to perform fire PRA (NUREG/CR-6850 supplemented by frequently asked questions) is summarized in the LAR and the sources of uncertainty in each Task described. The

licensee concluded that there were no sources of uncertainty in any of the Tasks that should be evaluated with additional sensitivity studies to support the categorization of SSCs. Therefore, the NRC staff finds the licensee searched for, identified, and evaluated sources of uncertainty in its fire PRA consistent with the guidance to identify additional "applicable sensitivity studies."

In LAR Section 3.2.2.2, the licensee describes how it addressed the fire PRA sensitivity study requiring taking no credit for manual suppression. The licensee described that it 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 (MCR) abandonment. In the case of a fire in the MCR, plant procedures and operator training make it highly unlikely that manual suppression would not be attempted. The NRC staff finds that the licensee's proposal is reasonable because a fire in the continuously occupied MCR will be detected and fought and PRA should be as realistic as possible.

The NRC staff finds the licensee's use of sensitivity studies acceptable because they include all of the studies recommended in the NRC-endorsed guidance. The single deviation from accepted guidance (i.e., not assuming failure of suppression in the MCR) is reasonable and acceptable, as explained above.

#### Integrated Importance Measures

Section 5.6 of NEI 00-04 provides the equations used to develop the integrated importance measures (F-V and RAW). The integrated importance measures are the weighted average of the importance measures from the internal events and the fire PRAs. These integrated values are compared against the criteria for HSS components of  $F-V > 0.005$ ,  $RAW > 2.0$  for individual basic events and  $RAW > 20$  for common cause basic events. The same evaluation is performed for LERF, if available.

Integrated importance measures are less conclusive than internal events importance measures because the fire PRA evaluations have larger uncertainties than the internal events evaluations. A component categorized as HSS based solely on its internal event PRA importance measures cannot be categorized as LSS based on its integrated importance measures. However, a component categorized as HSS based solely on the fire PRA importance measure may be categorized as LSS based on its integrated risk importance measure. The licensee includes this guidance in its categorization process as described in LAR Section 3.1.2. The NRC staff finds this acceptable because the differing treatment appropriately reflects the greater uncertainty in the fire PRA by allowing the total (i.e., integrated) risk to be appropriately used and this approach is consistent with NEI 00-04 guidance.

#### 3.3.3 Non-PRA Methods

As described in the LAR, the licensee's categorization process uses the following non-PRA methods:

- Seismic Margin Analysis (SMA) to assess seismic risk;
- Screening during the IPEEE to assess risk from other external hazards (high winds, external floods);

- Shutdown Safety Plan to assess shutdown risk.

These methods are discussed below.

### Seismic Risk

To assess seismic risk for the 10 CFR 50.69 categorization process, the licensee proposes to use the SMA method. SMA is a screening method that does not quantify core damage frequency. The licensee used the SMA method during the IPEEE (Reference 13) using the EPRI SMA methodology (Reference 14). The SMA method includes the development of the seismic Safe Shutdown Equipment List (SSEL) which contains all the components that would be needed during and after a seismic event. The SSEL identifies one preferred and one alternate path capable of achieving and maintaining safe shutdown conditions for at least 72 hours following an earthquake. The licensee indicated in the LAR that it had updated the IPEEE SSELs in 2011 and 2012 to reflect the current plant configuration.

Consistent with NEI 00-04, the licensee's 10 CFR 50.69 categorization process considers all components in the SSEL as HSS based on seismic risk. All components not listed in the SSEL are considered preliminary LSS with respect to seismic risk.

The method proposed by the licensee to assess seismic risk is consistent with the NRC-endorsed methods in NEI 00-04 and therefore the NRC staff finds it acceptable for use in the licensee's 10 CFR 50.69 categorization process.

### Other External Hazards (high winds, external floods)

As indicated in the LAR, external hazards were initially evaluated by the licensee during the IPEEE. This hazard category includes all non-seismic external hazards such as high winds, floods, and other hazards. The IPEEE external hazard analysis used a progressive screening approach and concluded that all hazards are negligible contributors to overall plant risk. Further, the licensee indicated that it had re-evaluated other external hazards in 2011, including the impact of construction activities at Vogtle, Units 3 and 4, using the criteria in ASME/ANS RA-Sa-2009 (Reference 12) and screened out all other external events. Therefore, the licensee proposes to treat all SSCs as preliminary LSS with respect to other external events risk. The NRC staff finds the licensee's approach consistent with the NEI 00-04 guidance and therefore acceptable for use in the licensee's 10 CFR 50.69 categorization process.

### Shutdown Risk

Consistent with the NEI 00-04 guidance endorsed by the NRC, the licensee proposes to use the shutdown safety assessment process based on NUMARC 91-06 (Reference 15). NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment - primary/secondary. The licensee indicated that it had updated its Shutdown Safety Program in early 2011.

As described in the LAR, components are categorized with respect to shutdown risk using a non-PRA shutdown assessment based on the following attributes.

- NUMARC 91-06 specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function. As clarified in NEI 00-04, 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 categorized HSS.
- NEI 00-04 also clarifies that components should be categorized HSS when failure of the component would initiate a shutdown event (e.g., loss of shutdown cooling, drain down, etc.).

A component that meets either of these conditions is categorized HSS, otherwise it is considered preliminary LSS with respect to shutdown safety. This approach is consistent with the NRC-endorsed NEI 00-04 guidance and therefore the NRC staff finds it acceptable for use by the licensee in the 10 CFR 50.69 categorization process.

#### 3.3.4 Component Safety Significance Assessment for Passive Components

Passive components are not modeled in the PRA and therefore a different assessment method is necessary to assess the safety significance of these components. Passive components are those components having only a pressure retaining function. This process also addresses the passive function of active components such as the pressure/liquid retention of the body of a motor operated valve.

In the LAR, the licensee proposed using a new method for passive component categorization, which used certain elements from the following three methods:

- Revision 0 of ASME Code Case N-660 (N-660) (Reference 16)
- WCAP-16308-NP-A (Reference 17)
- EPRI-TR112657 (EPRI) (Reference 18)

The first two methods were endorsed by the NRC to accomplish categorization of passive SSC functions to support the implementation of 10 CFR 50.69, and the third method is an NRC-endorsed methodology to risk-inform inservice inspection that is unrelated to 10 CFR 50.69.

As documented in RAI No. 5, the NRC staff questioned the licensee's approach of combining different parts of NRC-approved methods. Each endorsed method incorporates all its elements into a single process whereby some non-conservative elements are acceptable based on other conservative elements, and the safety implications of the collective evaluation is judged against the use of the results. Combining disparate elements of previously approved methods into a new method does not provide any basis for acceptability. In the response to RAI No. 5, the licensee elected to use the categorization method as approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 19). The ANO-2 method is a risk-informed safety classification and

treatment program for repair/replacement activities for Class 2 and 3 pressure retaining items and their associated supports (exclusive of Class CC and MC items), using a modification of the ASME Code Case N-660 (Reference 16) (ANO-2 methodology). In Enclosure 2 of the letter dated May 17, 2013 (Reference 2), the licensee provided an updated passive categorization procedure "10 CFR 50.69 Passive Component Categorization," NMP-ES-065-002, Revision 1, which incorporates the ANO-2 methodology. The change from the proposed categorization method to the ANO-2 method is included as Implementation Item No. 2.

The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequence which measures the safety significance of the pipe given that it ruptures is conservative compared to including the rupture frequency in the categorization and the categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore the NRC staff finds that the use of the repair/replacement methodology is acceptable and appropriate for passive component categorization that is subsequently used to adopt alternative treatment requirements.

The NRC staff finds this method acceptable for use in the licensee's 10 CFR 50.69 categorization process to categorize passive components. This method applies to Class 2 and 3 pressure retaining items or their associated supports (exclusive of Class CC and MC items).

### 3.3.5 Summary

The NRC staff reviewed the PRA and the non-PRA methods used by the licensee in the 10 CFR 50.69 categorization process to assess the safety significance of active and passive components and finds these methods acceptable and consistent with RG 1.201 (Reference 7) and the NRC-endorsed guidance in NEI 00-04. The NRC staff approves the use of the following methods in the licensee's 10 CFR 50.69 categorization process:

- PRA to assess internal events risk
- Fire PRA to assess fire risk
- SMA to assess seismic risk
- Screening using IPEEE to assess risk from other external hazards (high winds, external floods)
- Shutdown Safety Plan to assess shutdown risk
- ANO-2 (Reference 19) passive categorization method to assess passive component failure risk

The licensee's implementation of 10 CFR 50.69 is accomplished with the addition of a license condition. The license condition identifies seven implementation items that shall be completed before additional SSCs are categorized. The license condition states, in part, that prior NRC approval is required for a change to the categorization process that is specified in the LAR and its supplements.



### 3.4 Assessment of Defense-in-Depth and Safety Margin

As specified in the LAR, the licensee's assessment of defense-in-depth and safety margin is consistent with the guidance provided in NEI 00-04, Section 6.0. This step in the process is performed after the active and passive risk assessment and before performing the risk sensitivity studies. For preliminary LSS components, this assessment confirms that the defense-in-depth is preserved, consistent with the principles of RG 1.174 (Reference 11). The criteria used in this assessment include criteria for core damage defense-in-depth based on preserving redundancy and diversity. Additional criteria for containment defense-in-depth are based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns. If any one of the defense-in-depth criteria is not met for a specific SSC, the SSC would be categorized as HSS. The NRC staff finds the licensee's process consistent with the NRC-endorsed NEI 00-04 guidance, including the frequency versus available functional train matrix in Figure 6-1 in the NEI document, and therefore acceptable.

NEI 00-04 Section 9.2.2, Review of Safety Related Low Safety-Significant Functions/SSCs, which provides guidance related to maintaining safety margins states:

*Because the only requirements that are relaxed for LSS SSCs are those related to treatment, existing safety margins for SSCs arising from the design technical and functional requirements would remain. It is also 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 50.69. As a result, individual SSCs continue to be capable of performing their design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results. Therefore, it can be concluded that the sufficient safety margins are preserved. Consequently, no specific assessment of safety margin is required by the IDP.*

RG 1.201 that endorses NEI 00-04 provides no clarifications to the safety margin discussion.

The maintenance of sufficient safety margins is mentioned in many places throughout the LAR and RAI responses. Specific guidance related to safety margins is, however, only found in Section 4.7, Safety Analysis Representative, in draft procedure NMP-ES-065-003, which states that it is the responsibility of the Safety Analysis Representative to "ensure[] that sufficient safety margins are maintained for RISC-3 components." This guidance is supplemented with the Note, "[b]ecause the only requirements that are relaxed for LSS SSCs are those related to treatment, existing safety margins for SSCs arising from the design technical and functional requirements would remain. Consequently, no specific assessment of safety margin is required." The NRC staff finds the guidance related to safety margin acceptable because the design and functional requirements will remain unchanged, the categorization process includes statements directing that consideration be given to ensuring safety margins are maintained, and the guidance is consistent with the NEI 00-04 guidance endorsed in RG 1.201.

### 3.5 Risk Sensitivity Study

Consistent with the NRC-endorsed NEI 00-04 guidance, an overall risk sensitivity study is performed for all the preliminary LSS components to confirm that the categorization process

meets the acceptance guidelines of RG 1.174. This study is performed after the preliminary categorization is finalized and before review by the IDP. Guidance on how to conduct this study is provided in Section 8.0 of NEI 00-04. This guidance is to increase the unreliability of all preliminary LSS SSCs by a factor of 3 to 5. Separate sensitivity studies are to be performed for each system categorized, as well as a cumulative sensitivity study for all the SSCs categorized through the 10 CFR 50.69 process.

In RAI No. 8, the NRC staff requested the unreliability factor selected and a summary of the risk sensitivity study. The licensee responded that it used a factor of 3 to increase the unavailability or unreliability. Consistent with the NEI 00-04 guidance, the licensee performed (and will perform) both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process reviewed in Section 3.8 of this safety evaluation assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee performs the risk sensitivity study consistent with the guidance in NEI 00-04 Section 8.0 and that the process is therefore acceptable.

### 3.6 Integrated Decision-Making Panel

As described in the LAR, the licensee's IDP process is consistent with the guidance in NEI 00-04. The IDP is responsible for ensuring that an IDP is employed which systematically considers in the categorization of SSCs 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., CDF, LERF, and importance measures); deterministic engineering insights (e.g., defense-in-depth, safety margins, and containment integrity); and other pertinent information (e.g., generic and plant-specific operational and performance experience, feedback, and corrective actions program).

The IDP is responsible for approving the final categorizations. As part of this approval, draft procedure NMP-ES-065-003 states that the IDP is responsible for: (1) evaluating PRA risk insights, passive risk insights, and qualitative risk insights to reach a consensus-based categorization for systems, functions and components; (2) reviewing results from performance monitoring and periodic reassessments to ensure that the basis for the categorization of SSCs remains valid and that any implemented alternative treatments have not significantly degraded the performance of the associated components; and (3) evaluating recommended changes to categorization results due to PRA model updates, changes to the plant, changes to operational practices, and other applicable changes.

As provided by the NEI 00-04 guidance, the process used by the IDP for the categorization of SSCs is described and documented in a plant procedure. The licensee's procedure NMP-ES-066-002, "Integrated Decision-Making Panel for Risk Informed SSC Categorization: Duties and Responsibilities," provided in Enclosure 2 of Reference 2 defines the duties and

responsibilities of the IDP, its composition, and the qualifications and training for the IDP members.

The IDP members collectively have expertise in the areas of PRA, operations, safety analysis, design engineering and system engineering. Procedure NMP-ES-066-002 identifies training requirements for members of the IDP, which include training on risk-informed defense-in-depth philosophy, PRA fundamentals, details on plant-specific PRA analyses (such as modeling scope and assumptions), interpretation of risk importance measures, role of sensitivity studies and changes in risk evaluations, and the categorization process. Refresher training is to be provided to IDP members every three years. The training and qualification of each IDP member is documented and maintained as a QA record for the life of the plant.

Procedure NMP-ES-066-002 also specifies the requirements for a quorum and the meeting frequency of 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 qualified), safety analysis, design engineering, systems engineering, and PRA. The IDP meets when a risk categorization is completed and ready for IDP review or when plant or PRA changes require re-evaluation of categorization results, or to review the results of the periodic review.

Procedure NMP-ES-066-002 requires that the minutes of the IDP meetings be recorded and maintained as QA records for the life of the plant. Consistent with the NEI guidance, the meeting minutes document the quorum of members present, meeting agenda and the results of the IDP activities, including the outcome of the categorization review, the basis for the determination, any differing opinions and any significant issues discussed leading to the decision.

As described in the NEI 00-04 guidance and in the licensee's LAR, components that are found to be 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 found to be HSS only from the fire PRA perspective may be categorized as LSS if the integrated assessment of component risk importance over all PRA models shows that the integrated risk importance measures meet the LSS criteria. (The integrated importance measures are discussed in Section 3.3.2.).

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decision-making. As outlined in NEI 00-04, the IDP may re-categorize components from HSS to LSS only if a credible failure of the component would not preclude the fulfillment of the HSS function and the component was not categorized as HSS based on the three criteria above (i.e., internal events PRA categorization, non-PRA evaluation, or passive categorization).

The NRC staff finds the licensee's IDP process, including its criteria allowing the IDP to change the categorization of components, consistent with the NEI 00-04 guidance and therefore acceptable.

### 3.7 Documentation

The documentation of the categorization process as listed in the licensee's draft procedure NMP-ES-065-003 includes: (1) procedures, instructions, or guidelines that describe the processes for the development, evaluation, and use of the SSC categorizations; (2) system functions, identified and categorized with the associated bases; (3) mapping of components to supported function(s); (4) PRA model results, including sensitivity studies; (5) hazards analyses, as applicable; (6) passive risk assessment results and bases; (7) categorization results for components, including all associated bases and the RISC classifications; (8) component critical attributes; (9) results of periodic reviews and SSC performance evaluations, and (10) IDP meeting minutes and qualification and training records of the IDP members. All the above are documented as Quality Assurance records and maintained for the life of the plant.

In LAR Section 3.4.3 the licensee indicates that it follows the process in Section 11 of NEI 00-04 pertaining to program documentation. 10 CFR 50.69(f) and consequently the guidance in NEI 00-04 requires the licensee to update the Final Safety Analysis Report (FSAR) to reflect which systems have been categorized and the alternative treatment of these categorized systems. Initial implementation of 10 CFR 50.69 may also entail changes to the quality assurance plan to reflect alternative treatment for categorized systems.

This documentation as described by the licensee is in conformance with the requirements of 10 CFR 50.69(f) and therefore acceptable.

### 3.8 Periodic Review

As described in the LAR Section 3.2.1.1, the licensee has administrative controls in place to ensure that the PRA models used to support the categorization reflect the as-built, as-operated plant over time. The licensee's process includes regularly scheduled and interim (as needed) PRA model updates. The process includes provisions for monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience), for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

Plant changes determined to have significant impact on the PRA models are implemented in the PRA concurrently with the changes to the plant. Other plant changes are collected for implementation in the PRA models during a routine PRA update. Routine PRA updates are performed every two refueling cycles at a minimum.

If PRA model errors are discovered, they are reviewed to determine the quantitative impact on PRA results and prioritized for correcting the error based on the significance of the quantitative impact. If a PRA model change is required but cannot be immediately implemented for a significant plant change or model error, then:

- 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; or

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

In LAR Section 3.4.3, the licensee states that it follows the process in Section 12 of NEI 00-04 pertaining to periodic review. The licensee's periodic reviews of the SSC categorizations are conducted at least once every two refueling cycles to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized. The procedure also states that a more timely review should be undertaken in situations when changes to the plant risk profile are identified or if it is identified that a RISC-3 or RISC-4 SSC can prevent an HSS function from being satisfied. Per the licensee's draft procedure NMP-ES-065-003, periodic reviews have the following objectives: (1) evaluate changes to the plant, operational practices, and applicable plant and industry operational experience for impact on existing categorizations; (2) incorporate PRA model updates into the categorizations, including updated sensitivity studies results, as applicable; (3) incorporate new PRA modeling capabilities; (4) evaluate RISC-3 component performance since the last review to ensure that performance is acceptable and that no declining trends are noted; (5) evaluate RISC-2 component performance since the last review to ensure that no additional controls are needed to ensure that safety significant functions can still be supported.

Procedure NMP-ES-065-001 directs that absolute importance measures be estimated along with the relative importance measures when reevaluating previously categorized systems. This guidance is consistent with Section 12 of NEI 00-04. The procedure provides two examples of how the absolute and relative measures are used to determine if the previous safety significant determination should be changed. NEI 00-04 provides a table that includes all four different possibilities of changes between these measures. The two examples provided in the licensee's procedure are consistent with the NEI guidance. Based on these two examples being consistent with the NEI guidance the NRC staff finds that the proposed approach is consistent with the NRC-endorsed NEI guidance and therefore acceptable.

As indicated in LAR Section 3.4.2.1, as part of the 10 CFR 50.69 implementation, the licensee will define and implement a performance monitoring process to ensure that potential increases in failure probabilities of LSS components will be detected and addressed before reaching the increased probabilities assumed in the sensitivity study (an increase by a factor of three). Consistent with NEI 00-04, the monitoring will address all functional failures, not just maintenance preventable functional failures.

Consistent with NEI 00-04 guidance, failures of RISC-3 SSCs will be identified and tracked in the corrective action program. The licensee's LAR states that it will follow its corrective action process regarding potential conditions adverse to quality. In this process, the cause of the condition is determined and corrective action taken in a timely manner to preclude repetition. Also, consistent with NEI 00-04, failures of RISC-3 SSCs will be reviewed to determine the extent of condition. Failures will be assessed for groups of component types because low failure probabilities for individual components may make changes difficult to detect. The review of

component group failure data is also performed to detect the occurrence of potential inter-system common cause failures and to allow timely corrective action as necessary.

The NRC staff finds that the performance monitoring of categorized SSCs and PRA updates will capture and evaluate component failures to identify significant changes in the failure probabilities. The PRA update program and associated re-evaluation of component importance will appropriately consider the effects of changing failure probabilities and changing plant configuration on the component safety-significant categories. Therefore, the NRC staff finds the proposed process and program acceptable because they provide confidence that feedback and process adjustment requirements in 10 CFR 50.69(e) will be met.

#### 4.0 CHANGES TO THE OPERATING LICENSE

The licensee proposed amendment to Appendix D, Additional Conditions, Facility Operating License No. NPF-68 and No. NPF-81 was provided in letter dated July 22, 2014 (Reference 6) and stated the following:

*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 August 31, 2012.*

*The licensee shall implement the items listed in enclosure 1, Implementation Items of SNC letter NL-14-0960, dated July 22, 2014, prior to categorizing systems under the process.*

*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).*

In the LAR and in the licensee's responses to the NRC staff's requests for additional information there were certain specific actions that the NRC staff identified as being necessary to support the conclusion that the proposed program met the requirements in 10 CFR 50.69 and the guidance in RG 1.201 and NEI 00-04. The licensee did not complete some of the actions while other actions were completed but updated documents were not provided (e.g., final procedures) that incorporated the actions. The NRC staff finding on the acceptability of the proposed process is depended on the completion of seven of these changes. These seven changes are identified as "Implementation Items" in Enclosure 1 in SNC letter NL-14-0960, dated July 22, 2014 (Reference 6). Other changes that were described by the licensee are less important and are similar to occasional future changes to the PRA and PRA methods that may occur over time and therefore can be addressed and resolved using the licensee's periodic review process.

The NRC staff finds that this proposed license condition, and its referenced Implementation Items in Enclosure 1 of SNC letter dated July 22, 2014 (Reference 6), is acceptable because it adequately implements 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors" using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed as acceptable to the NRC. For each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that each

remaining item will incorporate into the program upon its completion. The NRC staff, through an onsite audit or during future inspections, may choose to examine the closure of the implementation items, with the expectation that any variations discovered during this review, or concerns with regard to adequate completion of the implementation item, would be tracked and dispositioned appropriately under the licensee's corrective action program and could be subject to appropriate NRC enforcement action as they are part of the proposed license conditions.

## 5.0 SAFETY EVALUATION

The NRC staff has reviewed the SNC's 10 CFR 50.69 categorization process. The licensee's process, as supplemented by the license condition, is consistent with the NRC-endorsed NEI 00-04 guidance and thus satisfies the requirements of 10 CFR 50.69(c). Based on its review summarized in this safety evaluation, the NRC staff finds the proposed categorization process to be acceptable to categorize the safety significance of plant SSCs. Specifically the NRC staff has found that the licensee's categorization process: (1) considers results and insights from plant-specific internal events and fire PRAs that are of sufficient quality an level of detail to support the categorization process and have been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC; (2) determines SSC functional importance using an integrated, systematic process that reasonably reflects the current plant configuration and operating practices, and applicable plant and industry operational experience; (3) maintains DID; (4) includes evaluations that provide 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; and (5) is performed for entire systems and structures, not for selected components within a system or structure. Furthermore, the NRC staff has found that the sensitivity studies and the periodic reviews of the SSCs categorization are adequate for ensuring that the potential cumulative risk increase from implementing 10 CFR 50.69 is maintained acceptably small and that the SSCs performance monitoring program allows for the timely identification of changes to SSC failure characteristics and associated failure probabilities that may have an impact on the categorization.

Several deviations from the methods endorsed by RG 1.201 and several PRA technical adequacy issues were identified and dispositioned during the NRC staff review, as discussed in this safety evaluation. However based on the licensee's description of how the deviations have been, or will be, brought into compliance and how the PRA adequacy issues will be addressed, combined with the above NRC staff findings, the NRC staff concludes that the licensee's application of the alternative treatments authorized by 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2) for its SSCs that are categorized as LSS would pose no undue risk to public health and safety.

## 6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments. The State official had no comments.

## 7.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no

significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (79 FR 52067, September 2, 2014). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

## 8.0 CONCLUSION

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

Principal Contributor: S. Dinsmore

Date of issuance: December 17, 2014

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C. R. Pierce

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A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

**/RA/**

Robert Martin, Senior Project Manager  
Plant Licensing Branch II-1  
Division of Operating Reactor Licensing  
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:

1. Amendment No. 173 to NPF-68
2. Amendment No. 155 to NPF-81
3. Safety Evaluation

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