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

December 8, 2020

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 301 AND 301 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2019-LLA-0269)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 301 to Renewed Facility Operating License No. DPR-32 and Amendment No. 301 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station (Surry), Unit Nos. 1 and 2, respectively. The amendments modify the Surry licensing basis in response to your application dated December 6, 2019, as supplemented by letters dated June 29 and August 14, 2020. The amendments add a license condition to allow for the 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."

A copy of the related safety evaluation is also enclosed. The Commission's monthly *Federal Register* notice will include the notice of issuance.

Sincerely,

/RA/

Vaughn V. Thomas, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 301 to DPR-32
2. Amendment No. 301 to DPR-37
3. Safety Evaluation

cc: Listserv

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 301
Renewed License No. DPR-32

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated December 6, 2019, as supplemented by letters dated June 29 and August 14, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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;

   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.

2. Accordingly, the license is amended by changes to the Operating License as indicated in the attachment to this license amendment and new Paragraph 3.V of Renewed Facility Operating License No. DPR-32 will read as follows:

   V. License Condition

   The licensee 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) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301, dated December 8, 2020.

   Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2020.12.08 15:12:31 -05'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-32

Date of Issuance: December 8, 2020

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 301
Renewed License No. DPR-37

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated December 6, 2019, as supplemented by letters dated June 29 and August 14, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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;

   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.

2. Accordingly, the license is amended by changes to the Operating License as indicated in the attachment to this license amendment and new Paragraph 3.V of Renewed Facility Operating License No. DPR 37 will read as follows:

V. License Condition

The licensee 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) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301, dated December 8, 2020.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2020.12.08 15:13:06 -05'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. DPR-37

Date of Issuance: December 8, 2020

ATTACHMENT TO

AMENDMENT NO. 301 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 301 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

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

Renewed Facility Operating License No. DPR-32

REMOVE	INSERT
9	9
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Renewed Facility Operating License No. DPR-37

REMOVE	INSERT
9	9
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T. (Continued)

16. For the applicable UFSAR Chapter 14 events, Surry 1 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 1 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).

U. Deleted by Amendment No. 289

V. The licensee 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) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301 dated December 8, 2020.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

4. This renewed license is effective as of the date of issuance and shall expire at midnight on May 25, 2032.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Samuel J. Collins, Director

Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: March 20, 2003

T. (Continued)

16. For the applicable UFSAR Chapter 14 events, Surry 2 will re-analyze the transient consistent with VEPCO's NRC-approved reload design methodology in VEP-FRD-42, Rev. 2.1-A. If NRC review is deemed necessary pursuant to the requirements of 10 CFR 50.59, the accident analyses will be submitted to the NRC for review prior to operation at the uprate power level. These commitments apply to the following Surry 2 UFSAR Chapter 14 DNBR analyses that were analyzed at 2546 MWt consistent with the Statistical DNBR Evaluation Methodology in VEP-NE-2-A: • Section 14.2.7 - Excessive Heat Removal due to Feedwater System Malfunctions (Full Power Feedwater Temperature Reduction case only); • Section 14.2.8 - Excessive Load Increase Incident; • Section 14.2.9 - Loss of Reactor Coolant Flow; and • Section 14.2.10 - Loss of External Electrical Load	Prior to operating above 2546 MWt (98.4% RP).

U. Deleted by Amendment No. 289

V. The licensee 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) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, “Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization,” Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301 dated December 8, 2020.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach).

4. This renewed license is effective as of the date of issuance and shall expire at midnight on January 29, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

Samuel J. Collins, Director

Office of Nuclear Reactor Regulation

Attachment: Appendix A, Technical Specifications

Date of Issuance: March 20, 2003

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE ADOPTION OF 10 CFR 50.69

AMENDMENT NO. 301 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 301 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

DOMINION ENERGY VIRGINIA

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated December 6, 2019 (Reference [1]), as supplemented by letters dated June 29 (Reference [2]) and August 14, 2020 (Reference [3]), Virginia Electric and Power Company, Dominion Energy Virginia (Dominion, the licensee) submitted a license amendment request (LAR) for the Surry Power Station (Surry), Unit Nos. 1 and 2. The proposed amendments would modify the Surry licensing basis by adding a license condition to allow for the implementation of the provisions of 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."

The supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 11, 2020 (85 FR 7792).

The licensee proposed the following license condition to the Surry renewed facility operating licenses to allow the implementation of 10 CFR 50.69:

> The licensee 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) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the [10 CFR Part 50] Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988 [Reference [4]], "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess

seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports [Reference [5]]; and a screening of other external hazards [Reference [6]] updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009 ["Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference [7])]; as specified in License Amendment No. 301, dated December 8, 2020.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach.)

The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing SSCs according to their safety significance.[1]

To support its review, the NRC staff conducted an audit as described in the NRC letter dated March 5, 2020 (Reference [8]). Based on its review of the LAR and information provided in an online SharePoint site by the licensee, the NRC staff transmitted requests for additional information (RAIs) to the licensee by e-mail dated May 14, 2020 (Reference [9]), and no audit summary was needed. The licensee responded to the RAIs in supplemental letters dated June 29, 2020, and August 14, 2020.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design basis functions. For SSCs categorized as low safety significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety significance (HSS), requirements may not be changed.

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process; adjusts treatment requirements consistent with the relative significance of the SSC; and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety significance of SSCs and place the SSCs into one of four RISC categories.

[1] Regulatory Guide 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," May 2006 (Reference [10]), describes the SSC categorization process in its entirety as an overarching approach that includes multiple approaches and methods identified for a PRA hazard and non-PRA methods.

SSC categorization does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has HSS.

2.2 Regulatory Guidance

The NRC issued construction permits for Surry Units 1 and 2 before May 21, 1971; consequently, Surry Units 1 and 2 were not subject to the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants" (see SECY-92-223, "Resolution of Deviations Identified during the Systematic Evaluation Program," dated September 18, 1992 (Reference 13). Surry Units 1 and 2 meet the intent of the GDC published in 1967 (draft GDC). NRC staff reviewed the licensee's request against Updated Final Safety Analysis Report (UFSAR) Chapter 3, "Plant Design Criteria."

The NRC staff considered the following regulatory guidance during its review of the proposed changes:

- Regulatory Guide (RG) 1.201, Revision 1 (Reference [10]);
- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference [11]);
- RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference [12]);
- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Reference [13]); and
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Chapter 19, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference [14]).

Industry Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference [15]). This document is supported in RG 1.201, Revision 1, for trial use with clarifications and describes a method that the NRC considers acceptable for complying with 10 CFR 50.69 with respect to the categorization of SSCs that are considered in risk-informing special treatment requirements. This process determines the safety significance of SSCs and categorizes them into one of four RISC categories defined in 10 CFR 50.69.

Sections 2 through 10 of NEI 00-04 describe the following steps and elements of the SSC categorization process for meeting the requirements of 10 CFR 50.69:

- Sections 3.2 and 5.1 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(i).
- Sections 3, 4, 5, and 7 provide specific guidance corresponding to 10 CFR 50.69(c)(1)(ii).
- Section 6 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iii).
- Section 8 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(iv).
- Section 2 provides specific guidance corresponding to 10 CFR 50.69(c)(1)(v).
- Sections 9 and 10 provide specific guidance corresponding to 10 CFR 50.69(c)(2).

Additionally, Section 11 of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12 of NEI 00-04 provides guidance on the periodic review related to the requirements in 10 CFR 50.69(e). Maintaining change control and periodic review provides confidence that all aspects of the program reasonably reflect the current as-built, as-operated plant configuration and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

An acceptable approach for making risk-informed decisions about proposed licensing basis (LB) changes, including both permanent and temporary changes, is to show that the proposed LB changes meet the five key principles stated in Section C of RG 1.174, Revision 3 (Reference [12]). These key principles are:

Principle 1: The proposed LB change meets the current regulations, unless it is explicitly related to a requested exemption.

Principle 2: The proposed LB change is consistent with the defense-in-depth philosophy.

Principle 3: The proposed LB change maintains sufficient safety margins.

Principle 4: When the proposed LB change results in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

Principle 5: The impact of the proposed LB change should be monitored using performance measures strategies.

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3 and is pertinent to: (1) compliance with current regulations, (2) evaluation of defense-in-depth (DID), and (3) evaluation of safety margins.

3.2.1 Key Principle 1: Licensing Basis Change Meets the Current Regulations

Paragraph 50.69(c) of 10 CFR requires licensees to use an integrated decisionmaking process to categorize safety-related and nonsafety-related SSCs according to the safety significance of the functions they perform into one of the following four RISC categories, which are defined in 10 CFR 50.69(a), as follows:

RISC–1: Safety-related SSCs that perform safety significant functions[2]
RISC–2: Nonsafety-related SSCs that perform safety significant functions
RISC–3: Safety-related SSCs that perform low safety significant functions
RISC–4: Nonsafety-related SSCs that perform low safety significant functions

The SSCs are classified as having either HSS functions (i.e., RISC-1 and RISC-2 categories) or LSS functions (i.e., RISC-3 and RISC-4 categories). For HSS SSCs, 10 CFR 50.69 maintains current regulatory requirements for special treatment (i.e., it does not remove any requirements from these SSCs). For LSS SSCs, licensees can implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d). For RISC-3 SSCs, licensees can replace special treatment with an alternative treatment. For RISC-4 SSCs, 10 CFR 50.69 does not impose new treatment requirements.

Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the SSC categorization process satisfies the requirements of 10 CFR 50.69(c). As stated in 10 CFR 50.69(b), after the NRC approves an application for a license amendment, a licensee may voluntarily comply with 10 CFR 50.69, as an alternative to compliance with the following requirements for LSS SSCs:

(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) specified 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, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50
(x) specified requirements for containment leakage testing
(xi) specified requirements of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100

[2] NEI 00-04, Revision 0 (Reference [15]), uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

The NRC staff reviewed the licensee's SSC categorization process against the categorization process described in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1. The NRC staff's review, as documented in this safety evaluation (SE), used the framework provided in RG 1.174, Revision 3, and NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1.

Section 2 of NEI 00-04, Revision 0, in part, states that the SSC categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)

2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)

3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)

4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)

5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)

6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)

7. Integrated Decisionmaking Panel Review and Approval (Section 9 of NEI 00-04, Revision 0)

8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

In Section 3.1.1 of its LAR dated December 6, 2019, the licensee stated that it will implement the SSC categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1. In LAR Sections 3.1.1 and 3.2, the licensee described that the SSC categorization process uses PRA to assess risk from internal events (includes internal floods). For the other risk contributors, the licensee's process uses non-PRA methods for the risk characterization as follows:

- Internal fires are assessed using the fire safe shutdown equipment list (FSSEL) performed for the Surry fire protection program;

- Seismic hazard is assessed using an alternative approach supported by EPRI Report 3002012988 (Reference [4]);

- Other external hazards (e.g., high winds, external floods) are assessed using the Individual Plant Examination of External Events (IPEEE) screening analysis (Reference [6]);

- Shutdown events are assessed using the shutdown safety program described in Nuclear Management and Resources Council, Inc. (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference [16]); and

- Passive components are assessed using the ANO-2 passive categorization methodology (Reference [5]).

The NRC staff notes that use of the FSSEL for internal fires, the alternative approach for seismic hazard, and the ANO-2 passive categorization methodology for passive components are deviations from the NEI 00-04 guidance as endorsed by the NRC. NRC staff review of these alternative methods is provided in Section 3.3.1.2 of this SE.

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B, and the monitoring outlined in NEI 00-04, Revision 0, and clarifications in RG 1.201, Revision 1, ensures that the SSC categorization process is sufficient to assure that the SSC functions continue to be met and that any performance deficiencies will be identified and appropriate corrective actions taken. The NRC staff reviewed the licensee's SSC categorization program and concludes that it includes the appropriate steps and elements prescribed in NEI 00-04, Revision 0, to assure that the categorization of the SSCs are consistent with 10 CFR 50.69. The NRC staff performed a more detailed review of the specific steps and elements of the licensee's SSC categorization process, where necessary, to confirm consistency with the NEI 00-04 guidance, as endorsed. Based on the above, the NRC staff concludes that the proposed 10 CFR 50.69 program would meet regulatory requirements, and therefore, sufficiently address the first key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

3.2.2 Key Principle 2: Licensing Basis Change is Consistent with the Defense-In-Depth Philosophy

In RG 1.174, Revision 3, the NRC identified the following considerations for evaluating how the proposed LB change impacts DID:

- Preserve a reasonable balance among the layers of defense.
- Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
- Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
- Preserve adequate defense against potential common-cause failures.
- Maintain multiple fission product barriers.
- Preserve sufficient defense against human errors.
- Continue to meet the intent of the plant's design criteria.

RG 1.201, Revision 1, endorses the guidance in Section 6 of NEI 00-04, Revision 0, but notes that the containment isolation criteria in this section of the guidance are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Option A and Option B of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. The criteria provided in 10 CFR 50.69(b)(1)(x) are not to determine the proper RISC category for containment isolation valves or penetrations.

In Section 3.1.1 of its letter dated December 6, 2019, the licensee clarified that it will require an SSC to be categorized as HSS based on the DID assessment performed in accordance with NEI 00-04, Revision 0. Based on the above, the NRC staff concludes that the proposed change is consistent with the DID philosophy described in key principle 2 of RG 1.174, Revision 3, and is, therefore, acceptable. The licensee's process is consistent with the NRC-endorsed guidance in NEI 00-04, Revision 0, and meets 10 CFR 50.69(c)(1)(iii) criterion that requires DID to be maintained.

3.2.3 Key Principle 3: Licensing Basis Change Maintains Sufficient Safety Margins

The regulations in 10 CFR 50.69(c)(1)(iv) requires the evaluations to provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in core damage frequency (CDF) and large early release frequency (LERF) resulting from changes in treatment are small. The engineering evaluation that will be conducted by the licensee under 10 CFR 50.69 for SSC categorization will assess the design function(s) and risk significance of the SSCs to assure that sufficient safety margins are maintained. With sufficient safety margins, (1) the codes and standards or their alternatives approved for use by the NRC are met and (2) safety analysis acceptance criteria in the licensing basis (e.g., FSAR, supporting analyses) are met or proposed revisions provide sufficient margin to account for uncertainty in the analysis and data. RG 1.174, Revision 3 provides guidelines for making that assessment including evaluations to ensure the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis; or, if such inputs are affected, justification is provided to ensure sufficient safety margin will continue to exist.

Consistent with the guidance provided in NEI 00-04 for review of safety margins, and in accordance with the implementation of the SSC categorization program, the only requirements that are relaxed for LSS SSCs (includes RISC-3) are those related to treatment. The SSCs' design-basis function as described in the plant's LB, including the updated final safety analysis report and technical specifications bases, do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant LB. In Section 3.1.1 of the LAR, the licensee states that it will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by RG 1.201. Based on the above, the NRC staff concludes that the licensee has established a program to ensure sufficient safety margins are maintained in accordance with NEI 00-04 described in the third key principle of RG 1.174, Revision 3, and would meet 10 CFR 50.69(c)(1)(iv).

3.3 <u>Risk-Informed Assessment</u>

3.3.1 Key Principle 4: Change in Risk is Consistent with the Safety Goals

The risk-informed considerations prescribed in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, address the fourth key principle of RG 1.174, Revision 3, pertaining to the assessment for change in risk and for monitoring the impact of the LB change.

A summary of how the licensee's SSC categorization process is consistent with the guidance and methodology prescribed in NEI 00-04, Revision 0, and RG 1.201, Revision 1, is provided in the sections below.

In Sections 3.1.1 and 3.2 of its letter dated December 6, 2019, the licensee described that the SSC categorization process uses a PRA to assess risk from internal events (includes internal floods). The SSC categorization process uses the following non-PRA methods to assess risks from the other risk contributors:

- Internal fires are assessed using the FSSEL performed for the Surry fire protection program;
- Seismic hazard is assessed using an alternative approach supported by EPRI Report 3002012988 (Reference [4]);
- Other external hazards (e.g., high winds, external floods) are assessed using the IPEEE screening analysis (Reference [6]);
- Shutdown events are assessed using the shutdown safety program described in NUMARC 91-06 (Reference [16]); and
- Passive components are assessed using the ANO-2 passive categorization methodology (Reference [5]).

The approaches and methods proposed by the licensee to address internal events, other external hazards, DID, and shutdown events are consistent with the approaches and methods in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1. To address internal fires in the SSC categorization process, the licensee proposed to use an alternative method not specified in NEI 00-04. A detailed NRC staff review of the licensee's proposed approach for use of the FSSEL is provided in Section 3.3.1.2 of this SE. To address seismic hazard in the SSC categorization process, the licensee proposed to use an alternative method not endorsed by the NRC in NEI 00-04. A detailed NRC staff review of the licensee's proposed alternative seismic approach is provided in Section 3.3.1.2 of this SE. The non-PRA method for the categorization of passive components is consistent with the plant-specific approval of the ANO-2 methodology for passive components (Reference [5]) for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and Class 3 moderate- and high-energy systems. The NRC staff's review for the use of the ANO-2 methodology in the SSC categorization process is provided in Section 3.3.1.2 of this SE.

3.3.1.1 Scope of the PRA

The Surry PRA is comprised of a full-power internal events PRA (IEPRA) that evaluates the core damage frequency and large early release frequency risk metrics. In Section 3.3 of its letter dated December 6, 2019, the licensee states that the IEPRA (includes internal floods) model has been assessed against RG 1.200, Revision 2. Furthermore, LAR Section 3.3 states that a finding closure review was conducted on the IEPRA model in March 2018 using the NRC-accepted process documented in the NEI letter to the NRC entitled, "Final Revision of Appendix X to NEI 05-04/07-12/12-[13], Close-Out of Facts and Observations (F&Os)," dated February 21, 2017 (Reference [17]), as accepted by NRC in its letter dated May 3, 2017 (Reference [18]). The NRC staff finds that the LAR, as supplemented, provides sufficient information to support the NRC staff's review of the IEPRA (includes internal floods) for technical acceptability, and therefore, meets the requirements set forth in 10 CFR 50.69(b)(2)(ii) and (iii).

The NRC staff evaluated the scope of the PRA including: (1) peer-review history and results (includes open F&Os); (2) the Appendix X, F&O closure process; (3) credit for mitigating strategies (FLEX) in the PRA; and (4) assessment of assumptions and approximations. On May 14, 2020 (Reference [9]), the NRC staff issued RAIs to further assess the acceptability of Surry IEPRA (includes internal floods) for consistency with RG 1.200, Revision 2; NEI 00-04, Revision 0; and RG 1.201, Revision 1. The NRC staff's review of those aspects of the PRA and supplemental responses to assess for consistency with the applicable processes as endorsed by the NRC, where necessary, are provided below.

Internal Events PRA (Includes Internal Floods) Peer-Review History

In Section 3.3 of its letter dated December 6, 2019, the licensee stated that the IEPRA model was subjected to several focused-scope peer reviews in May 2010, July 2012, October 2016, February 2018, and October 2018. These focused-scope peer reviews were performed against the applicable high-level requirements (HLRs) and supporting requirements (SRs) of ASME/ANS RA-Sa-2009 (Reference [7]), as endorsed by RG 1.200, Revision 2. The focused-scope peer reviews resulted in a number of F&Os. An F&O closure review of the finding-level F&Os was conducted in March 2018. The licensee closed all but six IEPRA finding-level F&Os. LAR Section 3.3 states, in part, for the IEPRA (includes internal floods), "[n]o PRA upgrades as defined by the ASME PRA Standard RA-Sa-2009 have occurred since the October 2018 focused-scope peer review." Therefore, no subsequent peer review was required.

The six IEPRA finding-level F&Os that remain open were included in Attachment 3 of LAR Enclosure 1. The licensee dispositioned each open F&O by either providing a description of how the F&O was resolved or providing an assessment of the impact of the F&O resolution on the SSC categorization results. The NRC staff evaluated each open F&O and the licensee's disposition to determine whether the F&O had any significant impact on the application. The NRC staff finds, with the exception of F&O QU-F2-01 in LAR Attachment 3, the open IEPRA F&Os were assessed and dispositioned properly to support the proposed application.

F&O QU-F2-01 indicated the licensee's PRA update process does not ensure the truncation screening values used to quantify the PRA converge towards a stable result. By letter dated June 29, 2020, in response to RAI 05 concerning F&O QU-F2-01, the licensee revised the Surry PRA update procedure to require performing a truncation level sensitivity study with each PRA update to confirm the truncation values used to quantify the PRA meet the requirements of the PRA standard. Based on the above, the NRC staff finds F&O QU-F2-01 is resolved for this application.

For SR SY-A15 of ASME/ANS RA-Sa-2009, SSCs can be screened from the PRA where the failure probabilities of these SSCs are much less than those of other SSCs in the same system train that results in the same effect on system operation. SR SY-A15 provides the quantitative criteria for conducting this screening analysis. However, regarding F&O SY-A11-01, some SSCs in the Surry IEPRA had been screened based on qualitative arguments (e.g., "failure frequency was negligible"), and therefore, may not have been appropriately screened under SR SY-A15. The NRC staff determined that these screened SSCs would receive the same categorization as their associated functions under Section 7 of NEI 00-04, and therefore, would be categorized appropriately. As required by 10 CFR 50.69(c)(1)(i) and in accordance with Section 3 of NEI 00-04 regarding the PRA being of sufficient quality and level of detail to support the categorization process and assessed against a standard endorsed by the NRC, the licensee will document the basis for excluding specific SSCs from the PRA based on SR SY-A15 and any SSCs not screened will be incorporated into the IEPRA (Reference [3]).

The NRC staff reviewed the LAR and finds that the IEPRA (includes internal floods) conforms to the applicable technical elements in ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2, and is acceptable to support the Surry 10 CFR 50.69 program. Based on the above, the NRC staff concludes the Surry 10 CFR 50.69 program uses an IEPRA that is of sufficient quality to meet the requirements set forth in 10 CFR 50.69(c)(1)(i) regarding PRA quality.

Credit for FLEX Equipment

The NRC memorandum dated May 30, 2017, entitled, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (Reference [19]), provides the NRC staff's assessment of challenges to incorporating FLEX (diverse and flexible coping strategy) equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200, Revision 2.

By letter dated December 6, 2019, the licensee stated in Attachment 6 of LAR Enclosure 1 that the Surry IEPRA does credit both FLEX equipment and FLEX strategies. LAR Attachment 6 describes the FLEX equipment and associated strategies modeled in the IEPRA, as well as the methodology used to assess operator actions associated with the FLEX equipment. The licensee indicated that the FLEX modeling was in accordance with ASME/ANS RA-SA-2009, as endorsed by RG 1.200, Revision 2. The licensee stated that a focused-scope peer review of the FLEX modeling was conducted in September 2016 and assessed to meet capability category II in ASME/ANS RA-SA-2009 with no findings.

It was not clear in LAR Attachment 6 how operator training/experience was reflected in the human error probabilities (HEPs) associated with deployment and installation of the portable FLEX equipment (e.g., deployment could utilize non-trained personnel that do not have the same training given to operators). By letter dated June 29, 2020, in response to RAI 03.c, the licensee clarified that security personnel are used to move portable FLEX equipment and are trained in accordance with a systematic approach to training to determine the appropriate level of training for the security personnel including the frequency for continued training. The qualified personnel continue to receive training to ensure their skill level is maintained. A focused-scope peer review was performed in September 2016 on the FLEX modeling with no findings.

Based on the above, the NRC staff finds that the modeling of FLEX equipment in the IEPRA has been sufficiently addressed to support the Surry 10 CFR 50.69 categorization process.

Identification and Treatment of Key Assumptions and Sources of Uncertainty

The NRC staff determined that several sources of uncertainty related to PRA modeling simplifications were screened based on not having a significant impact on overall plant risk. As described in its letter dated December 6, 2019, and in accordance with Section 9 of NEI 00-04, as endorsed by RG 1.201, Revision 1, the licensee's integrated decision-making panel (IDP) will use information and risk insights compiled in the initial categorization process, including awareness of the limitations and assumptions of the PRA, and combines that with other information from design bases, DID, and safety margins to finalize the categorization of functions/SSCs. As a result, the NRC staff finds the IDP will appropriately consider PRA assumptions and simplifications during the SSC categorization process. During its review, the NRC staff identified a potential key source of uncertainty related to crediting of non-environmental qualified (non-EQ) equipment in containment. Given the harsh environmental conditions within containment during certain accident sequences, the survival of non-EQ equipment would be questionable. By letter dated June 29, 2020, in response to RAI 04.a.ii, the licensee confirmed that no non-EQ equipment in containment was credited in the IEPRA. Based on the additional information, the NRC staff finds this issue is no longer a source of uncertainty.

The NRC staff identified a conservative modeling decision where load shedding the direct current bus during a station blackout event is always failed. The NRC staff notes that conservative modeling choices can impact an SSC risk categorization. By letter dated June 29, 2020, in response to RAI 04.a.iii, the licensee clarified that the PRA model was updated to include direct current bus load shedding for station blackout events. Based on the model update, the NRC staff finds that this issue is no longer a source of uncertainty.

The NRC staff requested additional information on the basis for the assumption that flooding isolation is successful in certain flooding scenarios with at least 2 hours available for operators to isolate the flood before equipment damage. By letter dated June 29, 2020, in response to RAI 04.a.iv, the licensee explained that operator actions to isolate these floods within 2 hours are highly reliable. The NRC staff determined that this approach is consistent with the internal flooding screening SR in ASME/ANS Ra-Sa-2009 and was peer reviewed, and therefore, the NRC staff finds this assumption is not a key source of uncertainty.

The NRC staff observed that a sensitivity study was performed for the loss of offsite power (LOOP) recovery curves for human reliability analysis and determined the impact was not significant with regards to plant risk. The NRC staff notes that minimal changes in risk can impact an SSC risk categorization. Based on the licensee’s June 29, 2020, response to RAI 04.a.v, the NRC staff finds that the LOOP recovery curves used in the IEPRA are based on reasonable data sources. In addition, the NRC staff finds that the principal SSCs used to mitigate an LOOP event (e.g., emergency diesel generators, auxiliary feedwater turbine driven pump) have a risk achievement worth (RAW) and Fussell-Vesely (F-V) in the base IEPRA greater than the risk-significant threshold criteria (i.e., these SSCs are HSS); therefore, the categorization of these SSCs will not be affected by uncertainties in the LOOP recovery curves. The NRC staff also notes that the HEP sensitivity analyses required by Table 5-2 of NEI 00-04 would address uncertainties associated with the LOOP recovery curves. Based on the discussion above, the NRC staff finds this issue to be resolved.

In LAR Attachment 6, the licensee provided a list of key assumptions and sources of modeling uncertainties associated with the IEPRA (includes internal floods) and dispositioned each. The licensee concluded that the following additional sensitivity analysis beyond those required under Section 5 of NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, is necessary to address Surry PRA model-specific assumptions and sources of uncertainty:

- Perform a sensitivity analysis for the independent FLEX failure probabilities using the 5th and 95th percentile values (Reference [3]).

The NRC staff recognizes that the licensee will perform routine PRA changes and updates to assure the PRA continually reflects the as-built, as-operated plant, in addition to changes made to the PRA to support the context of the analysis being performed (e.g., sensitivity analyses).

To address 10 CFR 50.69(e) and (f), which stipulate the process for feedback and adjustment to assure configuration control is maintained for routine changes and updates to the PRA, Section 3.5 of the LAR states that after a PRA model update an SSC categorization review will be performed to assess its impact.

The NRC staff finds that the assessment performed by the licensee to identify and address the key assumptions and sources of modeling uncertainty for the IEPRA (includes internal floods) is consistent with the guidance provided in NUREG-1855, Revision 1, and Section 5 of NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, and, therefore, meets the requirements set forth in 10 CFR 50.69(b)(2)(ii) and (c)(1)(ii).

PRA Acceptability Conclusions

The licensee has subjected the IEPRA to the peer-review process and closed all but six F&Os associated with the peer review in accordance with an NRC-accepted process. The NRC staff reviewed the peer-review history, F&O closure review history, dispositions of the remaining open F&Os, and the identification and disposition of key assumptions and sources of uncertainty. The NRC staff concludes: (1) the licensee's IEPRA (includes internal floods) is sufficient to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1; and (2) the key assumptions and sources of model uncertainty for the IEPRA have been identified consistent with the guidance in RG 1.200, Revision 2, and NUREG-1855, Revision 1, and addressed appropriately for this application.

The NRC staff concludes that the IEPRA (includes internal floods) is acceptable, and therefore, meets the requirements set forth in 10 CFR 50.69(c)(1)(i) and (ii).

3.3.1.2 Evaluation of the Use of Non-PRA Methods in SSC Categorization

Method for Assessing Internal Fires

In the absence of a Fire Induced Vulnerability Evaluation (FIVE) or fire PRA specified in NEI 00-04 as approaches to address risk from fire, the licensee describes in Section 3.2.2 of its letter dated December 6, 2019, an alternate approach for addressing risk due to internal fires. The alternate approach considers use of the Surry FSSEL for the evaluation of an SSC's safety significance as it pertains to internal fires. The licensee proposes that all SSCs on the FSSEL will be categorized as HSS and the IDP cannot recategorize to LSS, consistent with the guidance in NEI 00-04 for non-PRA methods. The licensee's FSSEL is the result of Surry safe shutdown analysis methodology used to identify, select, and analyze systems, components, and cables needed to demonstrate compliance with 10 CFR Part 50, Appendix R.

Figure 3-2 of the LAR illustrates the process that will be used to assess the fire risk during the 10 CFR 50.69 categorization process. The process assesses: (1) whether the SSC being categorized is on the FSSEL, and (2) if the SSC is not on the FSSEL, whether the SSC is relied upon to maintain safe shutdown for fire. A positive confirmation for either question results in the SSC being categorized as HSS due to fire risk considerations. The licensee further confirms in LAR Section 3.2.2 that the fire protection system SSCs, including fire detection equipment, suppression equipment, and fire barriers (e.g., fire dampers) that may not be included on the FSSEL will be categorized as HSS. The licensee's approach also considered existing regulatory exemptions related to the fire safe shutdown program and fire-induced multiple spurious operations (MSOs) to identify any additional equipment relied upon to establish and maintain safe shutdown.

The NRC staff finds that the use of the FSSEL in the SSC categorization process is acceptable because the licensee uses the deterministic criteria from 10 CFR Part 50, Appendix R, to identify the functions necessary to achieve and maintain safe shutdown under postulated fire conditions and assigns HSS to all those SSCs that support the Appendix R functions. The NRC staff finds that the licensee's approach for addressing risk due to internal fires is acceptable, because the approach categorizes SSCs conservatively as HSS in the FSSEL, the fire protection SSCs, and any SSCs relied upon to establish and maintain safe shutdown under postulated fire conditions. The NRC staff notes that the licensee will evaluate plant changes to determine their impact on the FSSEL and the SSC categorization results. The NRC staff finds the licensee's approach to address risk for internal fires, when integrated with the other elements in NEI 00-04 as endorsed by the NRC, is acceptable for use in the 10 CFR 50.69 SSC categorization program.

Alternative Seismic Approach

As part of its proposed process to categorize SSCs according to safety significance, the licensee proposed to use a non-PRA method to consider seismic hazards. The regulations in 10 CFR 50.69(b)(2)(ii) and 50.69(c)(1)(ii) permit the use of systematic evaluation techniques in the risk-informed categorization process. The licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and described how the proposed alternative seismic approach would be used in the categorization process in Section 3.2.3 of the enclosure to its letter dated December 6, 2019, and its supplement dated June 29, 2020. In part, the licensee based its plant-specific evaluation on the case studies performed in EPRI Report 3002012988 (Reference [4]) and stated that the case studies are applicable to Surry and are used in the alternative seismic approach. In its supplement dated June 29, 2020 the licensee cited a precedent for its proposed alternative seismic approach (Reference [2]). The licensee identified differences between Surry and the NRC staff's approval of the precedent as documented in the Calvert Cliffs SE (Reference [20]). The information presented in the LAR and supplements, as well as in EPRI Report 3002012988, provided sufficient detail for the NRC staff to evaluate the licensee's proposed alternative seismic approach; how the proposed alternative seismic approach would be used in the categorization process; and the measures for assuring that the quality and level of detail of the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Based on the above, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(ii) for the proposed alternative seismic approach are met.

EPRI Report 3002012988 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The NRC staff's review confirmed that the case studies in EPRI Report 3002012988 used by the licensee to support its proposed alternative seismic approach, as well as the information in its supplements, provided sufficient plant-specific evaluation of the applicability and differences for Surry as compared to the precedent (Reference [20]). The information presented in the LAR and the supplements provided a sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv) for the alternative seismic approach. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed alternative seismic approach.

Evaluation of Technical Acceptability of the PRAs Used for Case Studies Supporting the Proposed Alternative Seismic Approach

In its supplement dated June 29, 2020, the licensee responded to the NRC staff's requests for information concerning the precedent including the case studies, mapping approach, and conclusions on the determination of unique HSS SSCs from the case studies which were used by the licensee to support its proposed alternative seismic approach. The licensee stated that the case study Plants A, C, and D, pertaining to the technical acceptability of the PRAs used, as well as the technical adequacy of certain technical details of the conduct of the case studies are applicable to Surry. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in the case studies for Plants A, C, and D in EPRI Report 3002012988 and the licensee's assertion of plant-specific applicability to the precedent. The NRC staff also evaluated the peer review process and resolution of peer-review findings, and key assumptions and sources of uncertainty for Plants A, C, and D.

Based on the above, the NRC staff finds that the technical acceptability of PRAs used for the Plant A, C, and D case studies in EPRI Report 3002012988, the mapping approach used in those case studies, and the conclusions on the determination of unique HSS SSCs from the case studies in the precedent (Reference [20]) are applicable to this licensee's proposed plant-specific alternative seismic approach. Therefore, the NRC staff concludes that the Plant A, C, and D PRAs were technically acceptable and applicable for use in the corresponding case studies supporting the licensee's proposed alternative seismic approach; the mapping of SSCs between the SPRA, the full-power IEPRA, and, as applicable, the fire PRA (FPRA) for the Plant A, C, and D case studies. The licensee's plant-specific evaluation is technically justifiable to support conclusions on the determination of unique HSS SSCs from SPRAs in Plant A, C, and D case studies in the EPRI Report 3002012988; and applicable to Surry and the licensee's proposed alternative seismic approach..

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In its letter dated December 6, 2019, the licensee states, in part, that:

> The Ground Motion Response Spectrum (GMRS) peak acceleration for Surry is at or below approximately 0.2g or where the GMRS is below or approximately equal to the Safe Shutdown Earthquake (SSE) between 1.0 Hz and 10 Hz.

In its letter dated June 29, 2020, the licensee stated that the GMRS-to-SSE comparison demonstrates that Surry qualifies as a Tier 1 plant under the criteria in the EPRI report and this comparison confirms the expected seismic risk at Surry would be very low. The NRC staff notes that the licensee's plant-specific evaluation is supported by its NRC 10 CFR 50.54(f) response dated March 31, 2014 (Reference [21]). The NRC staff reviewed the licensee's submittal and supplements and plant-specific evaluation and concludes that the licensee's proposed criteria of GMRS peak acceleration at or below approximately 0.2 g or the GMRS below or approximately equal to the SSE between 1.0 and 10 Hz, to determine the applicability and use of the proposed alternative seismic approach, is acceptable.

Evaluation of Applicability of Criteria for this Application

In Section 3.2.3 of the enclosure to its December 6, 2019 letter, the licensee compared the GMRS from the reevaluated seismic hazard for Surry developed and submitted by the licensee in response to Near-Term Task Force (NTTF) Recommendation 2.1 against the site's design-basis SSE in its March 31, 2014, response to the NRC 10 CFR 50.54(f) letter associated with NTTF Recommendation 2.1 (Reference [21]), to demonstrate that the site meets the criteria for application of the proposed alternative seismic approach. In Section 3.2.3 of the enclosure to its letter dated December 6, 2019, the licensee stated that the NRC staff concluded that the methodology used by the licensee in determining the GMRS was acceptable and that the GMRS determined by the licensee adequately characterized the reevaluated hazard for the Surry site. The NRC staff's review confirmed the licensee's statements and the comparison of the GMRS from the reevaluated seismic hazard against the SSE. Based on its review, the NRC staff finds that the licensee's seismic hazard meets the criteria for its proposed alternative seismic approach.

In Section 3.2.3 of the enclosure to its letter dated December 6, 2019, the licensee stated that the contribution of seismic risk to the integral assessment will be low such that seismic risk is unlikely to influence an HSS decision. In its supplement to the LAR dated June 29, 2020, the licensee emphasized the low site seismic risk and seismic capacity margins beyond the GMRS seismic level. Additionally, the licensee provided an estimate of the seismic core damage frequency and described recent plant changes that are expected to lower the estimated seismic core damage frequency.

The NRC staff verified the licensee's estimate by convolving the median seismic capacity with composite uncertainty provided in the LAR supplement dated June 29, 2020, and the reevaluated seismic hazard for the mean peak ground acceleration (Reference [21]). The licensee does not have a FPRA, and, as noted in Section 3.3.1.2, "Method for Assessing Internal Fires," of this SE, this licensee is using its FSSEL to categorize SSCs; and all SSCs on the FSSEL will be considered HSS and the IDP cannot recategorize them to LSS. Therefore, based on its evaluation and review of the insights from the EPRI Report 3002012988 that support the licensee's proposed alternative seismic approach and the HSS categorization for all SSCs on the licensee's FSSEL as part of this licensee's proposed 10 CFR 50.69 program, the NRC staff concludes that the seismic risk contribution for the licensee would not solely result in an SSC being categorized as HSS.

In summary, the NRC staff's review finds that the licensee's basis for applying the proposed alternative seismic approach is acceptable because: (1) the reevaluated hazard meets the criteria for use of the proposed alternative seismic approach, and (2) the seismic risk contribution is not expected to solely result in an SSC being categorized as HSS.

Evaluation of the Implementation of Conclusions from the Case Studies

The categorization conclusions from the EPRI Report 3002012988 case studies, performed for GMRS to SSE ratios significantly higher than Surry, indicated that seismic-specific failure modes resulted in HSS categorization uniquely from SPRAs. Therefore, such seismic-specific failure modes, such as correlated failures, relay chatter, and passive component structural failure mode, can influence the categorization process. The NRC staff reviewed the proposed alternative seismic approach to evaluate whether the categorization-related conclusions from EPRI Report 3002012988 were appropriately included and implemented.

Section 3.2.3 of the enclosure to its letter dated December 6, 2019, the licensee discussed the proposed alternative seismic approach. The licensee stated that the proposed categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for Surry. Additional information on the proposed alternative seismic approach is discussed by the licensee in Section 3.1.1 of the enclosure to the LAR.

In its supplement to the LAR dated June 29, 2020, the licensee stated that its plant-specific evaluation considered differences in the proposed alternative seismic approach between Surry and the precedent previously reviewed and approved by the NRC staff (Reference [20]). The NRC staff's review of the licensee's proposed alternative seismic approach determined that the precedent is applicable to this licensee's proposed alternative seismic approach and the plant-specific evaluation on the implementation of the alternative seismic approach is acceptable. The NRC staff's review of the licensee's proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statement of consideration, finds that the proposed alternative seismic approach provides reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(ii) as well as 10 CFR 50.69(c)(1)(iv) because:

1. The proposed alternative seismic approach includes qualitative consideration of seismic events at several steps of the categorization process, including documentation of the information for presentation to the IDP as part of the integrated, systematic process for categorization.

2. The proposed alternative seismic approach presents system-specific seismic insights to the IDP for consideration as part of the IDP review process as each system is categorized, thereby providing the IDP a means to consider potential impacts of seismic events in the categorization process.

3. The insights presented to the IDP include potentially important seismically-induced failure modes, as well as mitigation capabilities of SSCs during seismically-induced design basis and severe accident events consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI Report 3002012988. The insights will use prior plant-specific seismic evaluations, and therefore, in conjunction with performance monitoring for the proposed alternative seismic approach, reasonably reflect the current plant configuration. Further, the recommendation for categorizing civil structures in the alternative seismic approach provides appropriate consideration of such failures from a seismic event.

4. The proposed alternative seismic approach presents the IDP with the basis for the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

5. The proposed alternative seismic approach includes qualitative consideration and insights related to the impact of a seismic event on SSCs for each SSC that is categorized and does not limit the scope to SSCs from the case studies supporting this application.

Consideration of Changes to Seismic Hazard

An important input to the NRC staff's evaluation of the proposed alternative seismic approach is the current knowledge of the seismic hazard at the plant. The possibility exists for the seismic hazard at the site to increase such that the criteria for use of the proposed alternative seismic approach are challenged. In such a situation, the categorization process may be impacted from a seismic risk perspective either solely due to the seismic risk or by the integrated importance measure determination.

In Section 3.2.3 of the enclosure to its letter dated December 6, 2019, the licensee stated that "U.S. nuclear power plants that utilize the 50.69 Seismic Alternative (EPRI 3002012988) will continue to compare GMRS to SSE." Since the alternative seismic approach explicitly cites and is based on EPRI Report 3002012988, the continued comparison of GMRS to SSE applies to Surry. The licensee also stated that the seismic hazard at the plant is subject to periodic reconsideration as new information becomes available through industry evaluations.

The NRC staff's review finds that consideration of changes to seismic hazard in the licensee's plant-specific proposed alternative seismic approach is appropriately considered the precedent (Reference [20]). Therefore, the NRC staff's evaluation of the consideration of changes to the seismic hazard against the requirements in 10 CFR 50.69(e)(1), 10 CFR 50.69(e)(3), and 10 CFR 50.69(d)(2)(ii) as well as the resulting conclusion on consideration of changes to the seismic hazard in the precedent is applicable to this licensee's proposed alternative seismic approach. Consequently, the NRC staff finds that the consideration of changes to the seismic hazard at Surry that exceed the criteria for use of the proposed alternative seismic approach is acceptable because: (1) the criteria for use of the proposed alternative seismic approach is clear and traceable, (2) the proposed alternative seismic approach includes periodic reconsideration of the seismic hazard as new information becomes available, (3) the proposed alternative seismic approach satisfies the requirements in 10 CFR 50.69 discussed above, and (4) the licensee has included a proposed license condition in the LAR to require NRC approval for a change to the specified seismic categorization approach.

Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

In Section 3.5 of the enclosure to its letter dated December 6, 2019, the licensee described its feedback and adjustment process. By letter dated June 29, 2020, in response to RAI 09, the licensee provided a description of its periodic review process to review the impact of plant changes and a list of items to be included in the design effects and consideration review. Further, the licensee cited a precedent for its proposed alternative seismic approach (Reference [20]).

The NRC staff's review found that consideration of the feedback and adjustment process in the licensee's proposed alternative seismic approach is acceptable. The NRC staff finds that (1) the licensee's programs provide reasonable assurance that the existing seismic capacity of LSS components would not be significantly impacted, and (2) the monitoring and configuration control program ensures that potential degradation of the seismic capacity would be detected and addressed before significantly impacting the plant risk profile. Therefore, the NRC staff finds that reasonable confidence exists that the potential impact of the seismic hazard on the categorization is maintained acceptably low and the requirements in 10 CFR 50.69(c)(1)(iv) are met for the proposed alternative seismic approach.

Method for Assessing Other Non-Seismic External Hazards

The licensee discussed its consideration of other non-seismic external hazards in Section 3.2.4 of the enclosure to its letter dated December 6, 2019. The licensee stated that as part of the categorization assessment of other external hazard risk, the licensee will perform an evaluation to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario and consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.

Other non-seismic external hazards include high winds, external flood hazards, and other hazards listed in Appendix 6-A of ASME/ANS RA-Sa-2009. The licensee evaluated all non-seismic external hazards for the 10 CFR 50.69 application using a plant-specific evaluation in accordance with Generic Letter 88-20 (Reference [6]) and the criteria in ASME/ANS RA-Sa-2009. The NRC staff reviewed the licensee's evaluation, which was provided in Attachments 4 and 5 of the enclosure to the LAR.

LAR Attachment 4, "External Hazards Screening," states that Dominion Energy is in the process of reevaluating the external flooding hazard and the tornado missile hazard, and that any identified discrepancies will be tracked in the corrective action program. In RAI 07, the licensee was requested to provide a summary of the reevaluation of external flooding and tornado missile hazard, and to discuss the licensee's approach.

For the external flooding hazard, in its supplement dated June 29, 2020, in response to RAI 07, the licensee discussed the results of its flood focused hazard evaluation and integrated assessment (Reference [23]). Regarding the local intense precipitation, the evaluation concluded that the event frequency was less than 1E-06/year and, therefore, could be screened. For storm surges a bounding analysis was performed that determined core damage frequency to be less than 1E-06 per year. The licensee also noted that any SSC credited in the screening of this hazard will be categorized in accordance with Figure 5-6 of NEI 00-04. Based on the review of the information provided by the licensee in Attachment 4 of the enclosure to the LAR and its supplement dated June 29.2020, the NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of any SSCs for the external flooding hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

For the extreme wind or tornado hazard, in its supplement to the LAR dated August 14, 2020, the licensee provided details of its screening approach. The licensee cited its Updated Final Safety Analysis Report (UFSAR; Reference [25]), which provides the design basis for tornado hazard with a maximum wind speed of 360 miles per hour. The NRC staff's review of the design basis and NUREG/CR-4461, "Tornado Climatology of the Contiguous United States" (Table 6-1, Reference [26]), finds that the occurrence frequency of the design basis wind speeds for the site is less than 1E-7 per year.

The NRC staff's review notes that, since tornado bounds the extreme wind hazard, the 1E-6 per year tornado wind speed is less than the design basis tornado maximum wind speed of 360 miles per hour. In addition, the primary concern for high straight winds is LOOP caused by the winds and the IEPRA already includes LOOP events due to severe weather, including high, straight winds. The licensee stated that Figure 5-6 in NEI 00-04 will be followed for the extreme winds or tornado hazard as part of the categorization of SSCs. The NRC staff's review finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for the extreme winds or tornado hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

The licensee identified design basis calculations for areas of the fuel building, auxiliary building, and turbine building that did not agree with the tornado wind velocity listed in the UFSAR. The occurrence frequency of 1E-6 per year tornado is at wind speed is less than the tornado wind velocity used in the design basis calculations. Therefore, the NRC staff finds that the licensee's reanalysis of these structures is not expected to change the staff's conclusion on the consideration of the extreme wind and tornado hazard in the licensee's categorization process.

In its supplement to the LAR dated August 14, 2020, the licensee provided an explanation for consideration of tornado missile hazards in its proposed categorization approach. Based on its review, the NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of any SSCs for the tornado missile hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

In summary, the NRC staff's review finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for non-seismic external hazards and other hazards in Attachment 4 of the enclosure to its letter dated December 6, 2019 consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

Method for Assessing Passive Components

In Section 3.1.2 of its letter dated December 6, 2019, the licensee proposed using a categorization method for passive components not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, but that was a plant-specific relief request approved by the NRC for ANO-2 (Reference [5]). The ANO-2 precedent is a risk-informed safety classification and treatment program for repair/replacement activities for passive Class 2 and Class 3 pressure retaining items and their associated supports (exclusive of Class CC and Class MC items), using a modification of ASME Code Case N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1" (Reference [30]). The ANO-2 plant-specific approval relied 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 the event, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measure the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization is not generally affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff found for ANO that the use of the repair/replacement methodology was acceptable as an alternative for passive component categorization of Class 2 and Class 3 SSCs.

In Section 3.1.2 of the LAR, the licensee stated, "The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in ANO2-R&R-004 for the passive categorization of Class 2, 3, and non-class components … All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP." Because all Class 1 SSCs and supports will be considered HSS, and only Class 2 and Class 3 SSCs will be categorized for Surry using a similar modification of ASME Code Case 660 for the passive categorization precedent. The Surry plant-specific approach is consistent with previous NRC staff approval, the NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the SSC categorization process and meets the requirements set forth in 10 CFR 50.69(b)(2)(ii) and (c)(1)(ii).

3.3.1.3 Key Principle 4 Conclusions

Based on the above, the NRC staff review for IEPRA (includes internal floods) acceptability and evaluation of the use of non-PRA methods, concludes that the proposed change satisfies the fourth key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

3.3.2 Key Principle 5: Monitor the Impact of the Proposed Change

The guidance under NEI 00-04, Revision 0, includes programmatic configuration control and a periodic review to ensure all aspects of the 10 CFR 50.69 program, including use of PRA models, continue to reflect the as-built, as-operated plant and that updates to the PRA are continually incorporated.

Sections 11 and 12 of NEI 00-04, Revision 0, include discussions on periodic review, program documentation, and change control. Maintaining change control and periodic review will also maintain confidence that all aspects of the 10 CFR 50.69 program, including risk categorization of SSCs, continually reflect the Surry as-built, as-operated plant.

The NRC staff finds the risk management process described by the licensee in the LAR is consistent with Section 12 of NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, and consistent with the requirements in 10 CFR 50.69(e). Based on the above, the NRC staff has determined that the proposed change satisfies the fifth key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

4.0 TECHNICAL CONCLUSION

In the December 6, 2019, LAR, as supplemented by letter dated June 29, 2020, the licensee proposed to add a license condition to the Surry renewed facility operating licenses to allow the implementation of 10 CFR 50.69. Based on the NRC staff's review of the licensee's LAR and response to RAIs, the NRC staff identified specific actions, as described below that are identified as being necessary to support the NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69, the guidance in RG 1.201, Revision 1, and NEI 00-04, Revision 0. Note: Additional actions (e.g., final procedures) need not, and have not been submitted or reviewed by the NRC staff for issuance of the SE but will be completed before implementation of the program as specified in the 10 CFR 50.69 rule.

The NRC staff's finding on the acceptability of the SSC categorization process and PRA evaluation in the licensee's proposed 10 CFR 50.69 program is conditioned upon the license condition provided below. For the clarifications to the NEI 00-04, Revision 0 guidance and other changes that were described by the licensee, the NRC staff finds them to be routine and systematically addressed through the configuration management and control and periodic update processes as described in Sections 3.3.1.1 and 3.3.2 of this SE.

The licensee proposed to add Paragraph 3.V to Renewed Facility Operating License Nos. DPR-32 and DPR-37 for Surry, Unit Nos. 1 and 2, respectively. The proposed license condition states:

> V. The licensee 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) using: Probabilistic Risk Assessment (PRA) model to evaluate risk associated with internal events, including internal flooding; the Appendix R program to evaluate fire risk; a modified version of the Electric Power Research Institute (EPRI) 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1 approach to assess seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas

> Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; as specified in License Amendment No. 301, dated December 8, 2020.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from an Appendix R program fire risk evaluation to a fire probabilistic risk assessment approach.)

The NRC staff finds that the proposed license condition is acceptable, because: (1) it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC; and (2) the evaluation in SE Section 3.3.1.2, finds the non-PRA methods for assessing risk for internal fires, seismic, and passive components, which are deviations from NEI 00-04, to be acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC notified an official from the Virginia Division of Radiological Health of the proposed issuance of the amendment. On September 24, 2020, the Virginia State official confirmed that the Commonwealth of Virginia had no comments.

6.0 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 previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on February 11, 2020 (85 FR 7792), and the agency has received no public comments on this finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Under 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 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.

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Date issued: December 8, 2020

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 301 AND 301 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2019-LLA-0269) DATED DECEMBER 8, 2020

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