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

March 12, 2024

Mr. Bob Coffey
Executive Vice President, Nuclear Division
and Chief Nuclear Officer
Florida Power & Light Company
Mail Stop: EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: ST. LUCIE PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 254 and 209 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2022-LLA-0182)

Dear Mr. Coffey:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment Nos. 254 and 209 to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2, respectively. This is in response to your application dated December 2, 2022, as supplemented by letters dated September 11, 2023, September 26, 2023, and November 21, 2023.

The amendments add a new license condition to the renewed facility operating licenses to permit the implementation 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 provisions of 10 CFR 50.69 allow adjustment of the scope of the structures, systems, and components subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation).

A copy of the related safety evaluation is also enclosed. Notice of issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Natreon J. Jordan, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-335 and 50-389

Enclosures:
1. Amendment No. 254 to DPR-67
2. Amendment No. 209 to NPF-16
3. Safety Evaluation

cc: Listserv

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 254
Renewed License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company dated December 2, 2022, as supplemented by letters dated September 11, 2023, September 26, 2023, and November 21, 2023, 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 as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. DPR-67 is hereby amended to add paragraph 3.L.to read as follows:

   L. 50.69 License Condition

   FPL 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) models to evaluate risk associated with internal events, internal flooding, and internal fire; 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in FPL's original submittal letter dated December 2, 2022, and all its subsequent associated supplements; as specified in License Amendment No. 254 dated March 12, 2024.

   Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

   FPL shall complete the numbered items listed in Attachment 1, List of Categorization Prerequisites, of FPL letter dated September 26, 2023 (ML23269A150), prior to implementation. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DAVID WRONA

Digitally signed by DAVID WRONA
Date: 2024.03.12 11:25:00 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: March 12, 2024

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

FLORIDA POWER AND LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 209
Renewed License No. NPF-16

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

   A. The application for amendment by Florida Power & Light Company dated December 2, 2022, as supplemented by letters dated September 11, 2023, September 26, 2023, and November 21, 2023, 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 as indicated in the attachment to this license amendment, and Renewed Facility Operating License No. DPR-67 is hereby amended to add paragraph 3.Q. to read as follows:

   Q. 50.69 License Condition

   FPL 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) models to evaluate risk associated with internal events, internal flooding, and internal fire; 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in FPL's original submittal letter dated December 2, 2022, and all its subsequent associated supplements; as specified in License Amendment No. 209 dated March 12, 2024.

   Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

   FPL shall complete the numbered items listed in Attachment 1, List of Categorization Prerequisites, of FPL letter dated September 26, 2023 (ML23269A150), prior to implementation. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

DAVID WRONA

Digitally signed by DAVID WRONA
Date: 2024.03.12 11:25:17 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: March 12, 2024

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

ATTACHMENT TO LICENSE AMENDMENT NOS. 254 AND 209

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

DOCKET NOS. 50-335 AND 50-389

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

REMOVE	INSERT
License DPR-67	License DPR-67
Page 7	Page 7
---	Page 8
License NPF-16	License NPF-16
Page 8	Page 8
---	Page 9

b. For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced Surveillance interval begins upon completion of the first Surveillance performed after implementation of this amendment.

c. For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended Surveillance interval begins upon completion of the last Surveillance performed prior to implementation of this amendment.

d. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance subject to the modified acceptance criteria is due at the end of the first Surveillance interval that began on the date the Surveillance was last performed prior to the implementation of this amendment.

L. 50.69 License Condition

FPL 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) models to evaluate risk associated with internal events, internal flooding, and internal fire; 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in FPL’s original submittal letter dated December 2, 2022, and all its subsequent associated supplements; as specified in License Amendment No. 254 dated March 12, 2024.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

FPL shall complete the numbered items listed in Attachment 1, List of Categorization Prerequisites, of FPL letter dated September 26, 2023 (ML23269A150), prior to implementation. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

4. This renewed license is effective as of the date of issuance and shall expire at midnight on March 1, 2036.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan

Date of Issuance: October 2, 2003

2. Schedule for New and Revised Surveillance Requirements (SRs)

   The schedule for performing SRs that are new or revised in License Amendment 208 shall be as follows:

   a. For SRs that are new in this amendment, the first performance is due at the end of the first Surveillance interval, which begins on the date of implementation of this amendment.

   b. For SRs that existed prior to this amendment, whose intervals of performance are being reduced, the first reduced Surveillance interval begins upon completion of the first Surveillance performed after implementation of this amendment.

   c. For SRs that existed prior to this amendment, whose intervals of performance are being extended, the first extended Surveillance interval begins upon completion of the last Surveillance performed prior to implementation of this amendment.

   d. For SRs that existed prior to this amendment that have modified acceptance criteria, the first performance subject to the modified acceptance criteria is due at the end of the first Surveillance interval that began on the date the Surveillance was last performed prior to the implementation of this amendment.

Q. <u>50.69 License Condition</u>

FPL 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) models to evaluate risk associated with internal events, internal flooding, and internal fire; 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in FPL's original submittal letter dated December 2, 2022, and all its subsequent associated supplements; as specified in License Amendment No. 209 dated March 12, 2024.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

FPL shall complete the numbered items listed in Attachment 1, List of Categorization Prerequisites, of FPL letter dated September 26, 2023 (ML23269A150), prior to implementation. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

4. This renewed license is effective as of the date of issuance and shall expire at midnight on April 6, 2043.

FOR THE NUCLEAR REGULATORY COMMISSION

ORIGINAL SIGNED BY

J. E. Dyer, Director

Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A, Technical Specifications
2. Appendix B, Environmental Protection Plan
3. Appendix C, Antitrust Conditions
4. Appendix D, Antitrust Conditions

Date of Issuance: October 2, 2003

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 254 AND 209 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated December 2, 2022 (Reference 1), as supplemented by letters dated September 11, 2023 (Reference 2), September 26, 2023 (Reference 21), and November 21, 2023 (Reference 28), Florida Power and Light Company (FPL, the licensee) submitted a license amendment request (LAR) for St. Lucie Plant, Unit Nos. 1 and 2 (St. Lucie). Specifically, the licensee proposed to add a license condition to Renewed Facility Operating License Nos. DPR-67 and NPF-16 to allow the implementation 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," which would state:

> 50.69 License Condition
>
> FPL 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) models to evaluate risk associated with internal events, internal flooding, and internal fire; 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 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in FPL's original submittal letter dated December 2, 2022, and all its subsequent associated supplements; as specified in License Amendment No. [254/209] dated March 12, 2024.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

> FPL shall complete the numbered items listed in Attachment 1, List of Categorization Prerequisites, of FPL letter dated September 26, 2023 (ML23269A150), prior to implementation. All issues identified in the attachment will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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]

By correspondence dated August 14, 2023 (Reference 3) and October 31, 2023 (Reference 26), the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff requested additional information from the licensee. The licensee responded to the requests for additional information (RAIs) by supplemental letters dated September 11, 2023 (Reference 2), September 26, 2023 (Reference 21), and November 21, 2023 (Reference 28). 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 the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 16, 2023 (88 FR 31282).

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.

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

[1] Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (Reference 4), 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.

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 Applicable Regulatory Guidance

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

- RG 1.201, Revision 1 (Reference 4)
- RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed Activities" (Reference 5)
- 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 6)
- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Reference 7)
- NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Reference 8)

2.3 Applicable NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 9), endorsed by the NRC in RG 1.201, Revision 1, subject to the regulatory positions and specific clarifications in RG 1.201, Revision 1, which describes a process that the NRC staff considers acceptable for complying with 10 CFR 50.69. This process determines the safety significance of SSCs and categorizes them into one of the four RISC categories defined in 10 CFR 50.69.

Sections 2 through 10 of NEI 00-04 describe the following steps/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 6). 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 (DID) philosophy.

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

Principle 4: When the proposed LB changes result in an increase in risk, the increases 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 measurement strategies.

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3 that are pertinent to: (1) compliance with current regulations, (2) evaluation of 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 decision-making process to categorize safety-related and non-safety-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: Non-safety-related SSCs that perform safety-significant functions

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[2] NEI 00-04, Revision 0 (Reference 9), 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.

RISC–3: Safety-related SSCs that perform LSS functions
RISC–4: Non-safety-related SSCs that perform LSS 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 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

The NRC staff reviewed the licensee's SSC categorization process against the categorization process described in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1 (References 9 and 4), and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The staff's review, as documented in this safety evaluation (SE), used the framework provided in RG 1.174, and NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

Section 2 of NEI 00-04, Revision 0, states, in part, that the categorization process includes the following 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 (IDP) 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 the LAR, the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. In Section 3.1.2 of the LAR, the licensee proposed the use of the ANO-2 Risk-Informed Repair/Replacement passive categorization method and in Section 3.2.3 of the LAR, the licensee proposed the Electric Power Research Institute (EPRI) 3002017583 Tier 1 alternate seismic approach, as alternative methods to assess the applicable hazard contribution(s). The NRC staff notes that use of these alternative methods is a deviation from the NEI 00-04 guidance. A more detailed staff review of the alternative methods is provided in Section 3.3.2 of this SE.

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

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, ensure that the SSC categorization process is sufficient to ensure that the SSC functions continue to be met and that any performance deficiencies will be identified, and appropriate corrective actions taken. The licensee's SSC categorization program includes the appropriate steps/elements prescribed in NEI 00-04, Revision 0, to ensure that SSCs specified are appropriately categorized consistent with 10 CFR 50.69. The NRC staff performed a more detailed review of specific steps/elements of the licensee's SSC categorization process where necessary to confirm its consistency with the NEI 00-04 guidance. In light of the above, the staff concludes that the proposed 10 CFR 50.69 program meets 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 used for evaluating philosophy for how the LB change is maintained for the DID philosophy:

- 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, 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 Options A and B of Appendix J 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 the LAR, 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. In light of the observations above, the NRC staff concludes that the proposed change is consistent with the DID philosophy and therefore satisfies the second key principle for risk-informed decision making prescribed in RG 1.174, Revision 3. The staff finds that the licensee's process is consistent with the NRC-endorsed guidance in NEI 00-04 and would meet the 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 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 SSC to ensure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring that 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 that sufficient safety margin will continue to exist.

The SSCs' design basis functions as described in the plants' LB, including the Updated Final Safety Analysis Report, do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant's LB. Therefore, the NRC staff concludes that the licensee has established a program to ensure that sufficient safety margins are maintained in accordance with the third key principle of RG 1.174, Revision 3, and, therefore, meets 10 CFR 50.69(c)(1)(iv).

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

In Section 2.2 of the LAR, the licensee stated that "[t]he safety functions [in the SSC categorization process] include the design basis functions, as well as functions credited for severe accidents (including external events)." Section *3*.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk significant information will be collected. In Section 3.1.1 of the LAR, the licensee confirmed that the SSC categorization process documentation will include, among other things, system functions, identified and categorized with the associated bases, and mapping of components to support function(s).

The NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and meets the requirements in 10 CFR 50.69(c)(1)(ii) and (iv).

3.3 Risk-Informed Evaluation

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 the NRC in RG 1.201, Revision 1, address the fourth and fifth key principles of the standards for risk-informed decision making, pertaining to the assessment for change in risk and monitoring the impact of the LB change.

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

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

The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel, such as the systematic process (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions) that is further discussed in Section 3.3.4 of this SE. The licensee's risk categorization process uses PRAs to assess risks from the internal events PRA (IEPRA) (including internal floods and the fire PRA (FPRA). For non-PRA methods that depart from the methodology prescribed in NEI 00-04, additional staff review is discussed in Section 3.3.2 of this SE.

Paragraph 50.69(c)(1)(v) of 10 CFR requires that SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. The NRC staff finds that the process described in the LAR for collecting and organizing information at the system level for defining boundaries, functions, and components is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and, therefore, meets the requirements set forth in 10 CFR 50.69(c)(1)(v).

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

This step in the licensee's categorization process assesses the safety significance of components using quantitative or qualitative risk information from a modeled PRA hazard, other hazards that can be screened, and non-PRA method(s). In NEI 00-04, component risk significance is assessed separately for the following hazard groups:

- internal events (including internal floods),
- internal fire events,
- seismic events,
- external hazards (e.g., high winds, external floods),
- other hazards,
- shutdown events, and
- passive categorization

In Sections 3.1.1 and 3.2 of the LAR, the licensee described that the St. Lucie categorization process uses PRA modeled hazards to assess risks for internal events (includes internal floods) and internal fires. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Seismic Hazard: Alternative seismic treatment using guidance from EPRI 3002017583 dated February 2020 (Reference 19) and qualitative insights about seismic risk at St. Lucie.
- Other External Hazards: Screening analysis performed for Individual Plant Examination of External Events (IPEEE) in accordance with NRC Generic Letter (GL) 88-20 (Reference 10) and updated using criteria from Part 6 of American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) 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" (the PRA Standard, Reference 15), as endorsed by the NRC in RG 1.200.
- Shutdown Events: Safe Shutdown Risk Management program consistent with Nuclear Management and Resources Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 11).
- Passive Components: ANO-2 passive categorization methodology (Reference 12).

The approaches and methods proposed by the licensee to address internal events, seismic, high winds, and other external events, DID, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0. The non-PRA method for the categorization of passive components is consistent with the ANO-2 methodology for passive components (Reference 12) that was approved by the NRC for risk-informed safety classification and treatment for repair/replacement activities in class 2 and class 3 moderate and high energy systems. The use of the ANO-2 methodology in the SSC categorization process is discussed in Section 3.3.1.2 of this SE. To address the 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 3.3.2 of this SE.

Scope of the PRA

The St. Lucie PRA is comprised of a full-power, Level 1, IEPRA and FPRA, which evaluates the core damage frequency (CDF) and large early release frequency (LERF) risk metrics. The licensee stated in Section 3.3 of the LAR that the IEPRA (includes internal floods) and the FPRA models were assessed against RG 1.200, Revision 2. Furthermore, LAR Section 3.3 stated that finding closure reviews were conducted on the identified PRA models in September 2017, August 2018, and April 2019, using the NRC-accepted process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-out of Facts and Observations (F&Os),'" dated February 21, 2017 (Reference 13).

The NRC staff finds that the LAR provided the information necessary to support the staff's review of the IEPRA (includes internal flooding) and the FPRA for technical acceptability, and therefore, meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

Aspects considered by the NRC staff to evaluate the scope of the PRA include: (1) peer-review history and results, (2) the Appendix X Independent Assessment process, (3) credit for Diverse and Flexible Coping Strategies (FLEX) in the PRA, and (4) assessment of PRA key assumptions and sources of uncertainty. By correspondence dated August 14, 2023 (Reference 3), the staff issued RAIs to further assess the acceptability of the St. Lucie IEPRA (including internal floods) and the FPRA for consistency with RG 1.200, Revision 2, and NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. The staff's review of these aspects of the PRA and the responses to the applicable RAIs are provided in the following subsections of this SE.

Internal Events PRA (includes internal floods) Peer-Review History

In Section 3.3 of the LAR, the licensee stated that the internal events PRA model was subjected to a full-scope peer review in July 2002 by the Combustion Engineering Owners Group (CEOG) prior to the issuance of RG 1.200. As a result, a self-assessment was conducted by FPL of the internal events PRA model in accordance with Appendix B of RG 1.200, Revision 2, to address the PRA technical adequacy requirements not considered in the 2002 peer review. The full-scope review was followed by focused-scope peer reviews in July 2009 (LERF), August 2009 (common cause failure methodology and data), April 2011 (data analysis, internal flooding, and human reliability analysis (HRA)), and October 2013 (loss of coolant accident sequences). Subsequently, in September 2017 and April 2019, FPL conducted an Independent Assessment for closure of the Finding-level F&Os and concluded that all the IEPRA (includes internal floods) F&Os have been closed. The licensee further stated that there are no PRA upgrades in the internal events PRA that have not had a subsequent focused-scope peer review.

Therefore, the NRC staff concludes that the St. Lucie IEPRA (including internal floods) was appropriately peer reviewed, consistent with RG 1.200, Revision 2, and that the F&Os have been adequately closed using an NRC approved process.

Internal Fire PRA Peer-Review History

The licensee's FPRA was subjected to a full-scope industry peer review in January 2010, consistent with RG 1.200, Revision 2. The Finding-level F&Os from this review were reviewed for closure by an Independent Assessment review team in September 2017, August 2018, and April 2019. The closure team closed all open F&Os with the exception of three F&Os that remained open, provided in Attachment 3 of the LAR. The NRC staff reviewed the disposition of the three remaining open F&Os and requested additional information as discussed below.

In response to RAI-03 and RAI-04 (APLA), the licensee stated that as a prerequisite to the implementation of the 10 CFR 50.69 categorization processes, two fire-related F&Os will be closed using an NRC approved F&O closure process. In Attachment 1 of the supplement to the LAR, dated September 26, 2023, the licensee defined the following prerequisite in Item No. 5:

> Since F&O findings CS-B1-01 and CS-A3-01 have not been formally closed using an NRC approved F&O closure process, F&O CS-B1-01 along with CS-A3-01 will be closed using an NRC approved F&O Closure process as a prerequisite to implementing the 10 CFR 50.69 categorization processes.

The third open Finding-level F&O (i.e., SF-A1-03) found that there was no assessment or review of the potential impact of a seismic event on the fire PRA suppression, procedure review, and brigade training. The NRC staff notes that this finding was written against a Supporting Requirement (SR) under the Seismic-Fire interaction (SF) technical element, which calls for qualitative (as opposed to quantitative) assessment of potential seismic/fire interactions issues. The staff also notes that it is a seismic PRA issue because it concerns a seismic event and subsequent fire. Accordingly, the issue is best addressed by the seismic PRA; therefore, this F&O has no impact on the fire PRA used for the 10 CFR 50.69 categorization.

In RAI-05 (APLA) (Reference 3), the NRC staff noted that based on auditing the applicable peer and closure review documents, there appeared to be three additional FPRA F&Os that remained open after the 2018 F&O closure process. Accordingly, the staff requested (1) clarification of F&Os closed after the August 2018 F&O closure review and (2) disposition for the 10 CFR 50.69 application of any F&O not assessed as closed. In response to RAI-05 (APLA), the licensee explained that as a prerequisite to implementing the 10 CFR 50.69 process: (1) a review for any open F&Os will be performed; (2) an F&O closure process will be performed on any open Finding-level F&Os; and (3) all open Finding-level F&Os will be closed using an NRC approved process. In Attachment 1 of the supplement to the LAR dated September 26, 2023, the licensee defined the following prerequisite in Item No. 6:

> To clarify the status of open F&Os and to implement a closure review team assessment of F&Os not formally closed, a review of open F&Os and closure of these F&Os will be implemented. FPL will generate an F&O disposition matrix with reference to the closure review that closes each F&O and references the NRC approved F&O Closure process used to close the F&O. Closure of all F&Os will be performed as a prerequisite to implementing the 10 CFR 50.69 categorization processes.

In RAI-06 (APLA), the NRC staff observed that the 2018 F&O closure review determined that four FPRA F&Os (i.e., F-5 (ES-C1-01), F-6 (ES-CW-01), F-8(FQ-C1-01), and F-24 (HRA-C1-01)) required a focused scope peer review because the associated changes in the FPRA constituted a PRA Upgrade. The staff also observed that the 2019 Independent Assessment Team (IAT) identified these four F&Os as “PRA Maintenance.” Given this possible inconsistency, the staff requested: (1) a description of the F&Os and associated model changes along with clarification of the status of the closure the four cited F&Os from the 2018 review; (2) an explanation of the differences between the 2018 and 2019 disposition of these F&Os; and (3) a proposal of a mechanism to perform a focused scope peer review on any of the F&Os determined to be PRA Upgrades and subsequent closure of any open F&Os using an NRC approved process. In response to RAI-06 (APLA), the licensee provided a description of cited F&Os, model change resolutions by FPL, and assessments by the 2018 and 2019 IATs. The licensee explained that during the 2019 focused scope review performed by the IAT, three of the cited F&Os were determined to be PRA Maintenance. For the fourth unevaluated F&O (F-8 (FQ-C1-01)), the licensee stated that, as a prerequisite, a focused scope peer review will be performed prior to implementing the 10 CFR 50.69 categorization processes, and that any further F&Os that were generated would be closed using an NRC approved process. In Attachment 1 of the supplement to the LAR, dated September 26, 2023, the licensee defined the following prerequisite in Item No. 7:

> FQ-C1-01 will have a Focused Scope Peer Review of the Fire PRA followed by any necessary F&O closure items by an Independent Assessment Team per NEI 05-04/07-12/12-06 Appendix X: “Close-Out of Facts and Observations (F&Os),” as a prerequisite to implementing the 10 CFR 50.69 categorization processes.

In Attachment 2 of the supplement to the LAR dated September 26, 2023, the licensee proposed a license condition that included completion of the seven numbered items (Implementation Items) in Attachment 1 prior to the implementation of the 10 CFR 50.69 categorization process, which includes prerequisite Nos. 5, 6, and 7 cited above.

The NRC staff reviewed the FPRA peer review results and the licensee’s resolution of the RAIs regarding the results and concludes that the St. Lucie FPRA was appropriately peer-reviewed, consistent with RG 1.200, Revision 2. The staff also finds that the F&Os have been closed or will be closed using an NRC approved F&O closure process as a prerequisite to the implementation of the 10 CFR 50.69 categorization processes to assess the impact on the risk-informed application.

Credit for FLEX Equipment

The NRC memorandum dated May 6, 2022, “Updated Assessment of Industry Guidance for Crediting Mitigating Strategies in Probabilistic Risk Assessments” (Reference 14), provides the NRC staff’s assessment of challenges to incorporating FLEX 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.

The licensee stated that the St. Lucie PRA does not credit FLEX equipment or FLEX strategies. However, in RAI-01 (APLA), the NRC staff observed that St. Lucie credits other portable equipment in the PRA models used in the LAR. Therefore, the staff requested: (1) clarification of what portable equipment is credited in the PRA models; (2) discussion of the method for assessing the failure probabilities of that equipment including the basis for the parameter values; and (3) discussion of how the method used to assess operator actions compares to the NRC approved methodology and justification that the modeling is not a PRA Upgrade as defined by RG 1.200. In response to RAI-1 (APLA), the licensee explained that the St. Lucie PRA models credit a non-FLEX diesel-driven portable fire pump. The licensee explained that as a prerequisite to the implementation of the 10 CFR 50.69 categorization processes FPL would (1) update the current assessment of failure probabilities for the diesel-driven pump using NRC accepted data sources and (2) update the modeling of operator actions to use the pump based on NRC accepted HRA methodologies. In addition, the licensee would perform a focused scope peer review on the updated modeling and perform an F&O closure review to close any resulting Findings using an NRC approved process. In Attachment 1 of the supplement to the LAR dated September 26, 2023, the licensee defined the following two prerequisites in Item Nos. 2 and 3, respectively:

> As a prerequisite to implementing the 10 CFR 50.69 categorization processes, FPL will update failure probabilities for the portable pump modeled in its PRA and apply NRC accepted data sources.
>
> As a prerequisite to implementing the 10 CFR 50.69 categorization processes, FPL will update the operator action for the portable pump by utilizing NRC accepted HRA methodologies and perform a FSPR [focused scope peer review] of this update followed by any necessary F&O closure items by an Independent Assessment Team IAT per NEI 05-04/07-12/12-06 Appendix X: “Close-Out of Facts and Observations (F&Os).”

In Attachment 2 of the LAR supplement dated September 26, 2023, the licensee proposed a license condition that included completion of the seven numbered items (Implementation Items) in Attachment 1 prior to the implementation of the 10 CFR 50.69 categorization process, which includes prerequisite Nos. 2 and 3 cited above.

In addition, Attachment 1 of the supplement includes another implementation item (i.e., Item No. 1) that is also encompassed by the proposed licensee condition and states:

> Any FLEX equipment credited in the future will apply NRC accepted data sources and HRA methodologies as a prerequisite to implementing the 10 CFR 50.69 categorization processes.

Therefore, the NRC staff finds that the IEPRA (includes internal floods) and the FPRA do not credit FLEX equipment but do credit other portable equipment for the SSC categorization process. The staff finds that, through the prerequisite items (Implementation Items) 1, 2, and 3, the licensee ensures the updating of the PRA models using NRC approved equipment data and operator error modeling consistent with NRC approved methods. Moreover, any new methods or upgrades will be peer reviewed and the Findings will be closed using NRC approved processes for independent peer review and F&O closure consistent with the ASME/ANS RA-Sa-2009 PRA standard, as endorsed by the NRC in RG 1.200.

Assessment of PRA Key Assumptions and Sources of Uncertainty

NUREG-1855, Revision 1 (Reference 7) provides guidance on how to treat PRA uncertainties in risk-informed decision-making. The licensee confirmed in response to RAI-02 (APLA) that sensitivity studies will be performed consistent with the NEI 00-04 guidance, and that their results will be provided to the integrated decision-making panel (IDP) for consideration in the final risk characterization for components initially classified as LSS that may be reclassified as HSS.

In RAI-02 (APLA), the NRC staff requested: (1) a description of the uncertainty evaluation process performed to determine sources of uncertainty that could impact 10 CFR 50.69 categorization and (2) the results of any sensitivity studies performed for key sources of uncertainty, along with justification that SSC risk informed categorization is not adversely impacted. In the LAR, as supplemented by the response to RAI-02 (APLA), the licensee explained the process used for the evaluation of key assumptions and sources of uncertainty. The licensee stated that it used guidance from NUREG-1855, Revision 1, and EPRI Technical Report 1016737, “Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments” (Reference 22), for identifying, characterizing, and qualitatively screening model uncertainties. The LAR stated that as part of this evaluation, non-conservative treatment or methods that are not commonly accepted were reviewed. The licensee further stated that the list of uncertainties compiled in the licensee’s uncertainty notebooks was reviewed to identify which of the assumptions or sources of uncertainty could significantly impact the risk calculations that support the SSC categorization process. The licensee explained that those assumptions or sources of uncertainty that were determined to have the potential to significantly impact categorization of SSCs were considered “key” and were presented in Attachment 6 of the LAR.

The NRC staff reviewed the key sources of uncertainties provided in the LAR and noted that two uncertainties were dispositioned by the licensee as conservative, because they would have the effect of categorizing more components into HSS than LSS. The third key source of uncertainty was related to consideration of human-induced errors in the support system initiating event frequency. In its response to RAI-02 (APLA), the licensee stated that human-induced errors are excluded from the support system initiating event models, which potentially impacts 10 CFR 50.69 categorization. Accordingly, the licensee stated that, as a perquisite to implementing the 10 CFR 50.69 categorization process, the initiating event models will be updated to remove the source of uncertainty (the human error modeling will be incorporated). In Attachment 1 of the supplement to the LAR dated September 26, 2023, the licensee defined the following prerequisite in Item No. 4:

> The key uncertainty involving Human-Induced errors in support system initiating event models has the potential to affect categorization of SSCs and the support system initiating event fault tree structure will be revised to include any applicable human failure events contributing to the support system initiating event and thereby remove this key source of uncertainty.

In Attachment 2 of the LAR supplement dated September 26, 2023, the licensee proposed a license condition that included completion of the seven numbered items (Implementation Items) in Attachment 1 prior to the implementation of the 10 CFR 50.69 categorization process, which includes prerequisite No. 4 cited above.

Given the above, the NRC staff finds that the assessment performed to identify the key assumptions/sources of uncertainty is consistent with the guidance provided in NUREG-1855, Revision 1.

In addition, the NRC staff recognizes that the licensee will perform routine PRA changes and updates to ensure that 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 (i.e., sensitivities). Paragraphs 50.69(e) and (f) of 10 CFR stipulate that the process for feedback and adjustment to ensure that configuration control is maintained for these routine changes and updates to the PRAs.

PRA Acceptability Conclusions

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 the IEPRA (includes internal floods) and the FPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. The PRAs must be acceptable to support the categorization process and must be subjected to a peer-review process assessed against a standard that is endorsed by the NRC. Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 PRA standard using a peer-review process.

The licensee has subjected the IEPRA (includes internal floods) and FPRA to the peer-review processes and submitted the results of the peer review. The NRC staff reviewed the peer-review history (which included the results and findings), the licensee's resolution of peer-review findings, and the identification and disposition of key assumptions and sources of uncertainty. The staff concludes that (1) the licensee's IEPRA (includes internal floods) and FPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC in RG 1.201, Revision 1, and (2) the key assumptions for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2 and NUREG-1855, as applicable, and are addressed appropriately for this application.

Based on the above, the NRC staff concludes that the licensee meets the requirements set forth in 10 CFR 50.69(c)(1)(i) and (ii).

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

Alternative Seismic Approach

As part of its proposed integrated decision-making 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(c)(1)(ii) and 50.69(b)(2)(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 how the proposed alternative seismic approach would be used in the categorization process in Section 3.2.3 of the LAR and the supplements dated September 11, 2023 and November 21, 2023. In part, the licensee based its plant-specific evaluation on the case studies performed in EPRI 3002017583 and stated that the case studies are applicable to St. Lucie and are used in the alternative seismic approach. The licensee described how its proposed alternative seismic approach would be used in the categorization process and the measures for ensuring that the quality and level of detail for its proposed alternative seismic approach are adequate for the categorization of SSCs. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(b)(2)(ii) for the proposed alternative seismic approach are met.

EPRI 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs. In Section 3.2.3 of the LAR, the licensee stated that the St. Lucie alternative seismic approach, supported by EPRI 3002017583, is similar to that in the R. E. Ginna Nuclear Power Plant adoption of 10 CFR 50.69 (Reference 23) which was approved by the NRC staff. The staff's review confirmed that the case studies in EPRI 3002017583 used by the licensee, as well as the information in its supplement (Reference 28), provide sufficient plant-specific basis for applicability of its proposed alternative seismic approach to St. Lucie. Accordingly, EPRI 3002017583 and its cited case studies previously approved by the staff, and information presented in the LAR and its supplements, provide a sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv). Therefore, the staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed alternative seismic approach.

Evaluation of the EPRI 3002017583 Case Studies

In its supplement to the LAR dated November 21, 2023, the licensee stated that the plant-specific case studies from other licensees included in EPRI 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The licensee also stated that there are no differences between the proposed St. Lucie alternative seismic approach and the approach used for the Calvert Cliffs Nuclear Power Plant 10 CFR 50.69 LAR, which was reviewed and approved by the NRC staff (Reference 24). The staff finds that the acceptability of the PRAs used in the Plants A, C, and D case studies in EPRI 3002017583, the mapping approach used in those case studies, and the conclusions on the determination of unique HSS SSCs from the case studies, which were reviewed and approved by the staff and used as a technical basis for the Calvert Cliffs Nuclear Power Plant 10 CFR 50.69 LAR, are applicable to the proposed alternative seismic approach for St. Lucie.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In the LAR, the licensee stated, in part, that the ground motion response spectrum (GMRS) peak spectral acceleration for St. Lucie is below the safe shutdown earthquake (SSE) between 1 Hertz (Hz) and 10 Hz, which demonstrates that St. Lucie qualifies as a Tier 1 plant under the criteria in EPRI 3002017583. In its supplemental letter dated September 11, 2023, the licensee presented the SSE and GMRS curves showing that the GMRS is below or equal to the SSE between 1 and 10 Hz. The NRC staff notes that the licensee's plant-specific evaluation is supported by its response to the NRC's 10 CFR 50.54(f) request dated January 7, 2016 (Reference 18). The staff reviewed the licensee's submittal, as supplemented, and plant-specific evaluation and concludes that use of the proposed criteria in EPRI 3002017583 to determine the applicability and use of the proposed seismic Tier 1 approach is acceptable.

Evaluation of Applicability of Criteria for 10 CFR 50.69

In the LAR supplement dated September 11, 2023, the licensee compared the St. Lucie GMRS from the reevaluated seismic hazard, developed and submitted by the licensee in response to Near-Term Task Force (NTTF) Recommendation 2.1, with the site's design basis SSE to demonstrate that the site meets the criteria for application of the proposed alternative seismic approach. The NRC staff's review confirmed the licensee's statements and its comparison of the GMRS with the SSE. Based on its review, the staff finds that the licensee's seismic hazard meets the criteria for the proposed alternative seismic approach.

In Section 3.2.3 of the LAR, the licensee stated that the small percentage contribution of seismic to total plant risk makes it unlikely that an integral importance assessment for a component, as defined in NEI 00-04, would result in an overall HSS determination. The NRC staff notes that Section 2.2.2 of the EPRI report identifies the expectation that low contribution of seismic risk to the total plant risk reduces the likelihood of a unique seismic condition that would cause an SSC to be designated HSS.

The NRC staff's evaluation of seismic risk to total plant risk was based on information in the St. Lucie TSTF-505 LAR (Reference 25). The NRC staff noted that seismic CDF contribution to the total plant CDF is low (i.e., about 5% for Unit 1 and about 4% for Unit 2). The staff reviewed the seismic LERF estimate in the St. Lucie TSTF-505 LAR and noted that seismic LERF is about 4% of the total LERF for both units. Therefore, overall seismic risk is relatively low compared to total plant risk due to its low seismic CDF and seismic LERF.

Further, as noted in Section 3.6.5 of EPRI 3002017583, containment DID assessment addresses containment failure and containment bypass situations. Section 3.6.6 of EPRI 3002017583, used for the licensee's proposed alternative seismic approach, recommends that if the licensee chooses to categorize civil structures housing HSS SSCs, the structures are considered as HSS. Therefore, based on its evaluation and review, the NRC staff concludes that the proposed alternative seismic approach, in conjunction with the other elements of the 10 CFR 50.69 categorization program, will appropriately determine the safety significance of any SSCs whose (1) seismic-induced failures lead directly to core damage and large early release and (2) seismic risk contribution would not solely result in any additional SSC being categorized as HSS.

The NRC staff finds that the licensee's basis for applying the proposed alternative seismic approach to its site is acceptable because (1) the reevaluated hazard meets the criteria for use of the proposed alternative seismic approach, (2) in conjunction with other elements of the 10 CFR 50.69 categorization program, the approach would appropriately determine the safety significance of any SSCs whose seismic-induced failures would lead directly to core damage and large early release, and (3) the seismic risk contribution would not solely result in any additional SSCs being categorized as HSS.

Evaluation of the Implementation of Conclusions from the Case Studies

In the LAR, as supplemented, the licensee stated that the proposed categorization approach for seismic hazards would 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 St. Lucie. The licensee explained that the qualitative characterization of seismic risk performed for the independent decision-making panel would include information from the various post-Fukushima seismic reviews, including results of seismic walkdowns, seismic mitigation strategy assessment, and seismic high-frequency evaluations. The objective of the alternative seismic approach is to identify plant-specific seismic insights derived from the components in the system being categorized.

The NRC staff's review of the licensee's proposed alternative seismic approach determined that the approach used in the Calvert Cliffs Nuclear Power Plant LAR is applicable to this licensee's proposed alternative seismic approach because the staff confirmed that no differences existed between the two methods. Therefore, the staff determined that the plant-specific evaluation of the implementation of the alternative seismic approach is acceptable. The staff's review of the proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statements of consideration, finds that the proposed alternative seismic approach includes the evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) for the following reasons:

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 seismic PRAs in EPRI 3002017583. The insights would 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 using the proposed alternative seismic approach, including the low seismic hazard for the plant and the criteria for the 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

The possibility exists for the seismic hazard at the site to increase such that the criteria for the use of the proposed alternative seismic approach are challenged. The licensee stated that the continued comparison of GMRS to SSE applies to the St. Lucie site. 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 finds that the consideration of changes to the seismic hazard in the licensee's plant-specific proposed alternative seismic approach is substantively the same as that approved in the Calvert Cliffs Nuclear Power Plant LAR. Consequently, the staff finds that the consideration of changes to the seismic hazard at St. Lucie that exceed the criteria for use of the proposed alternative seismic approach is acceptable for the proposed approach because (1) the criteria for use of the proposed alternative seismic approach are 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 as 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 LAR, the licensee stated that its configuration control process ensures that changes to the plant, including a physical change and changes to documents, are evaluated to ensure that the qualitative determinations for the seismic hazard continue to remain in compliance with the requirements of 10 CFR 50.69.

Based on its review, the NRC staff found that consideration of the feedback and adjustment process in the licensee's proposed alternative seismic approach is acceptable. The staff finds that:

1. The licensee's programs provide reasonable assurance that the existing seismic capacity of LSS components would not be significantly impacted.

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 the potential impact of the seismic hazard on the categorization is maintained acceptably low, and the requirements of 10 CFR 50.69(c)(1)(iv) are met by the proposed alternative seismic approach.

Method for Assessing Other External Hazards

This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation, nearby facility accidents, and other hazards. In the safety evaluation report for the St. Lucie IPEEE for Units 1 and 2, the NRC staff stated, in part, "[t]he high winds, floods, transportation and other external events (HFO) areas were eliminated based on either compliance with 1975 NRC Standard Review Plan (SRP) criteria or on the basis of a bounding probabilistic assessment resulting in a CDF estimate less than 1E-6 per reactor year, i.e., below the NUREG-1407 screening criterion [(Reference 17)]" (Reference 27).

In Section 3.2.4 of the LAR, the licensee stated, in part, that all other external hazards, except for seismic, were screened from applicability to St. Lucie per a plant-specific evaluation in accordance with GL 88-20 and updated to use the criteria in the ASME/ANS RA-Sa-2009 PRA Standard (Reference 15). The licensee confirmed that for St. Lucie the external hazards (except for seismic) would be subjected to the process described by the flow chart in NEI 00-04, Figure 5-6, which provides guidance to be used to determine SSC safety significance for these external hazards. The NRC staff finds that FPL will assess the risk from all other external hazards consistent with Figure 5-6 of NEI 00-04, as endorsed by the NRC in RG 1.201, Revision 1.

Regarding the external flooding, the licensee provided additional information in its supplemental letter dated September 11, 2023, in relation to the screening of the hazard from this application. The NRC staff observed that Attachment 4 of the LAR screens the external flooding hazard using criterion C1, "Event damage potential is less than events for which plant is designed." The staff assessment of the St. Lucie Flood Evaluation (Reference 20) concluded that effective flood protection depends on procedural action for Unit 2 to install portable stoplogs. In its supplemental letter dated September 11, 2023, the licensee cites (1) the portion of the St. Lucie Unit 2 Updated Final Safety Analysis Report (USFAR) that specifically describes stoplog installation, (2) St. Lucie Unit 2 Technical Specification (TS) 3/4.7.6, “Flood Protection,” Limiting Condition for Operation 3.7.6.1 that refers to the installation of stop logs for flood protection, and (3) the TS Bases for St. Lucie Unit 2 TS 3/4.7.6, which states “[t]he installation of the stoplogs ensures adequate protection from wave run-up effects where no permanent adjacent structures exist and provides protection to safety-related equipment.” Thus, stoplogs are part of the St. Lucie Unit 2 current licensing basis.

In summary, the use of the updated St. Lucie IPEEE results described by the licensee in the LAR and the licensee’s assessment of other external hazards (i.e., high winds, tornadoes, and external flood) in the LAR, as supplemented, are consistent with Section 5 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. Therefore, the NRC staff concludes that the licensee’s treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0, the licensee proposed using a shutdown safety assessment based on NUMARC 91-06 (Reference 11). 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. NUMARC 91-06 also 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.

The use of NUMARC 91-06, as described by the licensee in its submittal, is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the staff finds that the licensee’s use of NUMARC 91-06 is acceptable, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

Component Safety Significance Assessment for Passive Components

Passive components are not modeled in the PRA; therefore, a different assessment method is necessary to assess the safety significance of these components. For the purposes of 10 CFR 50.69 categorization, 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 Section 3.1.2 of the LAR, 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 approved by the NRC for ANO-2 (Reference 12). The ANO-2 methodology 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, “Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities” (Reference 16). 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 consequences, which measures the safety significance of the pipe (given that it ruptures), is conservative compared to including the rupture frequency in the categorization. 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 ANO-2 repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In Section 3.1.2 of the LAR, the licensee stated, "[t]he passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO 2-R&R-004 for the passive categorization of Class 2, 3, and non-class components." Consistent with ANO-2-R&R-004, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP.

Based on the above, the NRC staff finds that the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.3.3 Risk Sensitivity Study (NEI 00-04, Section 8)

Section 3.1.1 of the LAR states that an unreliability factor of three would be used for the sensitivity studies described in Section 8, "Risk Sensitivity Study," of NEI 00-04, Revision 0. Section 3.2.8 of the LAR further confirms that a cumulative sensitivity study would be performed where 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 three. The NRC staff finds that the application of a factor of three for the sensitivity studies is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

In Section 3.1.1 of the LAR for the "Overall Categorization Process," FPL specifically noted that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv)." These sensitivity studies, together with the periodic review process discussed in Section 3.6 of this SE, ensure 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 studies. The NRC staff finds that the licensee will perform the risk sensitivity studies consistent with the guidance in Section 8 of NEI 00-04, Revision 0, and, therefore, will ensure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.3.4 Integrated Decision Making

Appendix B of SRP Section 19.2 provides guidance and the NRC staff expectations for the licensee's integrated decision-making process. The appendix states, in part, that "[r]isk-informed applications are expected to require a process to integrate traditional engineering and probabilistic considerations to form the basis for acceptance." NEI 00-04 guidance identifies two steps in the categorization process, (1) Preliminary Engineering Categorization of Function and (2) IDP Review and Approval, that are responsible for the integrated assessment of the traditional engineering analyses and the risk results from the PRA and non-PRA assessments that are performed to make a determination and approval of the safety significance of the SSC for categorization. The staff's review of the two steps to ensures that the processes are well-defined, systematic, repeatable, and scrutable are provided as follows.

Preliminary Engineering Categorization of Function (NEI 00-04, Section 7)

All the information collected and evaluated in the licensee's engineering evaluations is provided to the IDP as described in Section 7 of NEI 00-04, Revision 0. The IDP makes the final decision about the safety significance of SSCs based on guidelines in NEI 00-04, Revision 0, the information they receive, and their expertise.

In Section 3.1.1 of the LAR, the licensee stated, in part, 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 [DID] assessment (Section 6), the associated system function(s) would be identified as HSS." The licensee also stated that, "[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS."

The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and is, therefore, acceptable.

Integrated Decisionmaking Panel Review and Approval (NEI 00-04, Sections 9 and 10)

In Section 3.1.1 of the LAR, the licensee stated that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. Therefore, the IDP will comprise the required expertise.

The guidance in NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process as required by 10 CFR 50.69(c)(1)(ii). In Section 3.1.1 of the LAR, the licensee discussed that at least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in modeling and updating of the plant-specific PRA. The licensee further stated that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. This training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant-specific PRA including the modeling, scope, and assumptions; the interpretation of risk importance measures; the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and the requirements to maintain this philosophy. The NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2) and that the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1.

As discussed in NEI 00-04, Revision 0, the only LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and 10 CFR 50.69(d)(2)(i) requires that the licensee ensure, with reasonable confidence, that RISC-3 SSCs remain capable of performing their safety-related functions under design basis conditions. Therefore, the NRC staff finds that the IDP for the St. Lucie categorization process is consistent with the NRC-endorsed guidance in NEI 00-04, Revision 0, and, therefore, fulfills 10 CFR 50.69(c)(1)(iv).

In light of the above NRC staff review for: (1) IEPRA and FPRA acceptability, (2) PRA importance measures and integrated importance measure, (3) evaluation of the use of non-PRA methods, (4) risk sensitivity studies, and (5) integrated decision making, the staff has determined that the proposed change satisfies the fourth key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

3.4 Key Principle 5: Monitor the Impact of the Proposed Licensing Basis Change

NEI 00-04, Revision 0, provides guidance that includes programmatic configuration control and a periodic review to ensure that the all aspects of the 10 CFR 50.69 program (i.e., includes traditional engineering analyses) and PRA models used to perform the risk assessment continue to reflect the as-built, as-operated plant and that plant modifications and updates to the PRA over time are continually incorporated.

Programmatic Configuration Control (NEI 00-04, Sections 11 and 12)

Sections 11 and 12 of NEI 00-04, Revision 0, include discussion on periodic review and 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 and risk categorization for SSCs continually reflect the St. Lucie as-built, as-operated plant. A more detailed NRC staff review is provided as follows.

Periodic Review (NEI 00-04, Section 12)

Section 50.69(e), “Feedback and process adjustment,” of 10 CFR requires periodic updates to the licensee’s PRA and SSC categorization. Changes over time to the PRA and to the SSC reliabilities are inevitable and such changes are recognized by the 10 CFR 50.69(e) requirement for periodic updates.

In Section 3.2.7 of the LAR, the licensee described the process for maintaining and updating the St. Lucie PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, Revision 0, the licensee confirmed that the St. Lucie risk management process ensures that the applicable PRA models used in this application continue to reflect the as-built and as-operated plant. The licensee’s 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), assessing the risk impact of unincorporated changes, and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

Routine PRA updates are performed every two refueling cycles at a minimum. The NRC staff finds that 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 the NRC in RG 1.201, Revision 1. Furthermore, in light of the above, the staff concludes that the proposed change satisfies the fifth key principle for risk-informed decision making prescribed in RG 1.174, Revision 3.

Program Documentation and Change Control (NEI 00-04, Section 11)

Section 50.69(f), "Program documentation, change control and records," of 10 CFR requires, in part, program documentation, change control, and records. In Section 3.2.7 of the LAR, the licensee stated that it will implement a process that addresses the requirements in Section 11 of NEI 00-04, Revision 0, pertaining to program documentation and change control records. Section 3.1.1 of the LAR states that the RISC categorization process documentation will include the following ten elements:

- Program procedures used in the categorization
- System functions, identified and categorized with the associated bases
- Mapping of components to support function(s)
- PRA model results, including sensitivity studies
- Hazards analyses, as applicable
- Passive categorization results and bases
- Categorization results including all associated bases and RISC classifications
- Component critical attributes for HSS SSCs
- Results of periodic reviews and SSC performance evaluations
- IDP meeting minutes and qualification/training records for the IDP members

The NRC staff also recognizes that under 10 CFR Part 50, Appendix B, Criterion VI, "Document Control," procedures are considered formal plant documents that require measures to be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. The staff finds that the elements provided in Section 3.1.1 of the LAR, in addition to the list of implementation items provided in Attachment 1 of Enclosure 1 to the supplemental letter dated September 26, 2023, for the St. Lucie 10 CFR 50.69 categorization process will be documented in formal licensee procedures consistent with Section 11 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1, and, therefore, sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control, and records.

4.0 PROPOSED REVISION TO THE LICENSES

In the LAR, the licensee proposed the addition of a license condition to Renewed Facility Operating License Nos. DPR-67 and NPF-16 for St. Lucie, Units 1 and 2, respectively. The NRC staff finds that the proposed license condition is acceptable because it adequately implements the program that the staff found above to be consistent with the NRC's regulations.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Florida official on February 27, 2024, of the proposed issuance of the amendments. The State official 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, "Standards for Protection Against Radiation." The NRC staff has determined that the amendments involve no significant change in the types or significant increase in the amounts 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, which was published in the *Federal Register* on May 16, 2023 (88 FR 31282), that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. 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.

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.

8.0 REFERENCES

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Date: March 12, 2024

SUBJECT: ST. LUCIE PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 254 and 209 TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2022-LLA-0182) DATED MARCH 12, 2024

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ADAMS Accession No. ML24017A209 ***by e-mail** ****by memorandum**

OFFICE	DORL/LPL2-2/PM*	DORL/LPL2-2/LA*	DRA/APLA/BC**	DSS/APLC/BC**	DEX/EEEB/BC*
NAME	NJordan	RButler	BPascarelli	SVasavada	WMorton
DATE	1/17/2024	1/24/2024	11/28/2023	12/01/2023	1/9/2024
OFFICE	DEX/EMIB/BC*	DEX/EICB/BC*	DNRL/SNSB/BC*	DNRL/NPHP/BC*	DNRL/NVIB/BC*
NAME	SBailey	FSanko	PSahd	MMitchell	ABuford
DATE	1/30/2024	1/10/2024	12/23/2023	1/2/2024	1/18/2024
OFFICE	DSS/SCPB/BC*	OGC (NLO)*	DORL/LPL2-2/BC*	DORL/LPL2-2/PM*	
NAME	RScully	JWatchutka	DWrona	NJordan	
DATE	1/3/2024	2/29/2024	3/7/2024	3/12/2024	

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