



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

April 30, 2026

Mr. Jonathan Huecker
Site Vice President
Catawba Nuclear Station
Duke Energy Carolinas, LLC
4800 Concord Road
York, SC 29745

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 325 AND 321, REGARDING ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2025-LLA-0156)

Dear Mr. Huecker:

The U.S. Nuclear Regulatory Commission has issued the following enclosed Amendment No. 325 to Renewed Facility Operating License No. NPF-35 and Amendment No. 321 to Renewed Facility Operating License No. NPF-52 for the Catawba Nuclear Station (Catawba), Units 1 and 2, respectively. The amendments are in response to your application dated September 30, 2025, as supplemented by letter dated March 26, 2026.

The amendments allow implementation of the provisions of Title 10 of the *Code of Federal Regulations* Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors," and adds a license condition to Appendix B, "Additional Conditions," of the Renewed Facility Operating Licenses for Catawba, Units 1 and 2, respectively.

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

J. Huecker

- 2 -

If you have any questions, please contact me at Shawn.Williams@nrc.gov.

Sincerely,

/RA/

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

Docket Nos. 50-413 and 50-414

Enclosures:

1. Amendment No. 325 to NPF-35
2. Amendment No. 321 to NPF-52
3. Safety Evaluation

cc: Listserv



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

DUKE ENERGY CAROLINAS, LLC
NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION
DOCKET NO. 50-413
CATAWBA NUCLEAR STATION, UNIT 1
AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 325
Renewed License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. NPF-35 filed by the Duke Energy Carolinas, LLC (licensee), dated September 30, 2025, as supplemented by letter dated March 26, 2026, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to Paragraph 2.C.(7) of Renewed Facility Operating License No. NPF-35, and Appendix B, "Additional Conditions," is revised to add the following:

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 325, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

Appendix B, "Additional Conditions"

Duke Energy 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, including 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, Class 3, and non-class SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic, and the alternative seismic approach described in Duke Energy's submittal letter RA-18-0089 dated September 30, 2025; supplemented by letter dated March 26, 2026, as specified in License Amendment No. 325 dated April 30, 2026.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-35

Date of Issuance: April 30, 2026



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

DUKE ENERGY CAROLINAS, LLC
NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1
PIEDMONT MUNICIPAL POWER AGENCY
DOCKET NO. 50-414
CATAWBA NUCLEAR STATION, UNIT 2
AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 321
Renewed License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. NPF-52 filed by the Duke Energy Carolinas, LLC (licensee), dated September 30, 2025, as supplemented by letter dated March 26, 2026, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to Paragraph 2.C.(7) of Renewed Facility Operating License No. NPF-52, and Appendix B, "Additional Conditions," is revised to add the following:

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 321, are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

Appendix B, "Additional Conditions"

Duke Energy 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, including 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, Class 3, and non-class SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic, and the alternative seismic approach described in Duke Energy's submittal letter RA-18-0089 dated September 30, 2025; supplemented by letter dated March 26, 2026, as specified in License Amendment No. 321 dated April 30, 2026.

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael Markley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-52

Date of Issuance: April 30, 2026

ATTACHMENT

AMENDMENT NO. 325 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-35

AMENDMENT NO. 321 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-52

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

Renewed Facility Operating License Nos. NPF-35 and NPF-52

Replace the following pages of the Renewed Facility Operating Licenses and the Appendix B, "Additional Conditions," with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

License Pages

Remove

NPF-35, page 5
NPF-52, page 5

Insert

NPF-35, page 5
NPF-52, page 5

Additional Conditions

Remove

Insert

NPF-35, Appendix B, page 7
NPF-52, Appendix B, page 6

(6) Mitigation Strategies

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 325 are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

D. The facility requires exemptions from certain requirements of Appendix J to 10 CFR Part 50, as delineated below and pursuant to evaluations contained in the referenced SER and SSERs. These include, (a) partial exemption from the requirement of paragraph III.D.2(b)(i) of Appendix J, the testing of containment airlocks at times when the containment integrity is not required (Section 6.2.6 of the SER and SSE Rs #3 and #4), (b) exemption from the requirement of paragraph 111.A.(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #3), and (c) partial exemption from the requirements of paragraph 111.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER #3). These exemptions are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and are consistent with certain special circumstances as discussed in the reference SER and SSE Rs. These exemptions are, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

(6) Mitigation Strategies

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(b) Operations to mitigate fuel damage considering the following:

1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel mitigation measures

(c) Actions to minimize release to include consideration of:

1. Water spray scrubbing
2. Dose to onsite responders

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 321 are hereby incorporated into this renewed operating license. Duke Energy Carolinas, LLC shall operate the facility in accordance with the Additional Conditions.

- D. The facility requires exemptions from certain requirements of Appendix J to 10 CFR Part 50, as delineated below and pursuant to evaluations contained in the referenced SER and SSER. These include, (a) partial exemption from the requirement of paragraph III.D.2(b)(i) of Appendix J, the testing of containment airlocks at times when the containment integrity is no required (Section 6.2.6 of the SER and SSERs #5), (b) exemption from the requirement of paragraph 111.A.(d) of Appendix J, insofar as it requires the venting and draining of lines for type A tests (Section 6.2.6 of SSER #5), and (c) partial exemption from the requirements of paragraph 111.B of Appendix J, as it relates to bellows testing (Section 6.2.6 of the SER and SSER #5). These exemptions are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and are consistent with certain special circumstances, as discussed in the reference SER and SSER. These exemptions are, therefore, hereby granted pursuant to 10 CFR 50.12. With the granting of these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

<u>Amendment Number</u>	<u>Additional Condition</u>	<u>Implementation Date</u>
325	<p>Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including 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, Class 3, and non-class SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa- 2009 for other external hazards except seismic, and the alternative seismic approach described in Duke Energy’s submittal letter RA-18-0089 dated September 30, 2025; as supplemented by letter dated March 26, 2026, as specified in License Amendment No. 325 dated April 30, 2026.</p> <p>Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).</p>	Upon implementation of Amendment No. 325

Renewed License No. NPF-35
Amendment No. 325

<u>Amendment Number</u>	<u>Additional Condition</u>	<u>Implementation Date</u>
321	<p>Duke Energy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including 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, Class 3, and non-class SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa- 2009 for other external hazards except seismic, and the alternative seismic approach described in Duke Energy’s submittal letter RA-18-0089 dated September 30, 2025; as supplemented by letter dated March 26, 2026, as specified in License Amendment No. 321 dated April 30, 2026.</p> <p>Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).</p>	Upon implementation of Amendment No. 321

Renewed License No. NPF-52
Amendment No. 321



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 325 AND 321 TO

RENEWED FACILITY OPERATING LICENSE NOS. NPF-35 AND NPF-52

DUKE ENERGY CAROLINAS, LLC

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By application dated September 30, 2025 (Reference [1]), as supplemented in letter dated March 26, 2026 (Reference [2]), Duke Energy Carolinas, LLC (Duke Energy, the licensee) submitted a license amendment request (LAR) to the U.S. Nuclear Regulatory Commission (NRC, the Commission) for Catawba Nuclear Station, Units 1 and 2 (Catawba). The amendments would allow the licensee to implement 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 NRC staff participated in a regulatory audit (Reference [3]) to ascertain the information needed to support its review of the application and to develop requests for additional information (RAI), as needed. Following the regulatory audit, the licensee submitted a supplement letter dated March 26, 2026, that included additional information resulting from the audit. The NRC staff issued a regulatory audit report summary on April 3, 2026 (Reference [4]).

The licensee provided additional information in its supplement dated March 26, 2026. The additional information 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 January 20, 2026 (91 FR 2375). The NRC did not receive public comment on this finding.

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (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 significance¹. Special treatment refers to those requirements that provide increased assurance (beyond normal industry practices) that SSCs will perform their design-basis functions. For SSCs with a function that is categorized as low-safety-significant (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs with a function determined to be high-safety-significant (HSS), requirements may not be changed.

The regulation, 10 CFR 50.69, 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 process is employed to determine the safety significance of SSCs and assign each into one of four Risk-Informed Safety Class (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 requirements are maintained and may be 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 HSS equipment.

2.2 Regulatory Guidance

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

- Regulatory Guide (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 [5])
- RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Reference [6])
- RG 1.200, Revision 3, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Reference [7])
- RG 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance” (Reference [8])
- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking” (Reference [9])

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, 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.

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP) Chapter 19, Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance” (Reference [10])

NOTE: The licensee’s submittal cites RG 1.174, Revision 2, as a guidance document applicable to the proposed change. RG 1.174 has been updated to Revision 3. The update does not include any technical changes that impact the consistency with RG 1.201 and implementation of the SSC categorization program, therefore, the NRC staff finds the updated revision to RG 1.174 also applicable for use in the licensee’s adoption of 10 CFR 50.69 and will be the revision referred to in the remainder of the safety evaluation.

NOTE: The licensee’s submittal references RG 1.200, Revision 2, for the internal events, internal flooding, large early release frequency, and fire Probabilistic Risk Assessment (PRA) models. RG 1.200 has since been updated to Revision 3. The update does not include any technical changes that impact the consistency with the RG 1.200 and implementation of the SSC categorization program; therefore, the NRC staff finds the use of Revision 3 acceptable for demonstrating the acceptability of the CNS PRA models.

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline” (Reference [11]), (hereafter referred to as NEI 00-04 in this SE) as endorsed by RG 1.201, Revision 1, for trial use, with clarifications and 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 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).
- Section 11 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f).
- Section 12 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 changes, including both permanent and temporary changes, is to show that the proposed changes to the licensing basis meet the five key principles of risk-informed decision-making stated in Section C of RG 1.174, Revision 3:

- Principle 1: The proposed licensing basis change meets the current regulations unless it is explicitly related to a requested exemption (i.e., a specific exemption under 10 CFR 50.12).
- Principle 2: The proposed licensing basis change is consistent with the defense-in-depth philosophy.
- Principle 3: The proposed licensing basis change maintains sufficient safety margins.
- Principle 4: When proposed licensing basis 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 licensing basis change should be monitored using performance measurement strategies.

The engineering evaluations below address the five key principles of RG 1.174, Revision 3.

3.2 Overview of Categorization Process

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. They are placed into one of the following four RISC categories:

- RISC-1: Safety-related SSCs that perform safety significant functions²
- RISC-2: Non-safety-related SSCs that perform safety significant functions
- RISC-3: Safety-related SSCs that perform low safety significant functions
- RISC-4: Non-safety-related SSCs that perform low safety significant functions

The SSCs have functions that are HSS or LSS, and they are classified accordingly. For SSCs that perform HSS functions (i.e., RISC-1 and RISC-2 SSCs), 10 CFR 50.69 maintains current regulatory requirements for special treatment, that is, all existing special treatment requirements continue to apply. In addition, 10 CFR 50.69(d)(1) requires licensees to ensure that RISC-1 and RISC-2 SSCs perform their functions consistent with the categorization process assumptions by evaluating treatment being applied to these SSCs to ensure that it supports the key assumptions in the categorization process that relate to their assumed performance. For SSCs

² NEI 00-04, Revision 0, 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" as used in 10 CFR 50.69, which applies to RISC-1 and RISC-2 SSCs.

that perform LSS functions, licensees may implement alternative treatment requirements in accordance with 10 CFR 50.69(b)(1) and 10 CFR 50.69(d)(2). For RISC-3 SSCs, licensees may replace special treatment requirements with an alternative treatment approach that meets 10 CFR 50.69(d)(2). 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) the specified 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, and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The NRC staff's review, as documented in this safety evaluation, used the framework provided in RG 1.174, Revision 3, and RG 1.201, Revision 1.

Section 2 of NEI 00-04 states that the categorization process includes the following eight primary steps:

1. Assembly of Plant-Specific Inputs
2. System Engineering Assessment
3. Component Safety Significance Assessment
4. Defense-in-Depth Assessment
5. Preliminary Engineering Categorization of Functions
6. Risk Sensitivity Study
7. Integrated Decision-Making Panel Review and Approval
8. SSC Categorization

3.3 Traditional Engineering Evaluation

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

3.3.1 Key Principle 1: Licensing Bases Change Meets the Current Regulations

In Section 3.1.1, "Overall Categorization Process," of the LAR, the licensee states that it will implement the risk categorization process in accordance with NEI 00-04 as endorsed by RG 1.201, Revision 1. The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in the endorsed guidance of NEI 00-04.

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B, as well as the monitoring outlined in NEI 00-04 will ensure that the SSC functions continue to be met, that any performance deficiencies will be identified, and that appropriate corrective actions will be taken. The NRC staff finds that the licensee's SSC categorization program includes the appropriate steps/elements prescribed in NEI 00-04 to assure that SSCs are appropriately categorized, consistent with 10 CFR 50.69. Therefore, the NRC staff concludes that the proposed 10 CFR 50.69 program meets the first key principle for risk-informed decision-making identified in RG 1.174, Revision 3.

3.3.2 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 how the licensing basis change is maintained for the defense-in-depth 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.

In Section 3.1.1 of the LAR enclosure, the licensee explains that, consistent with the guidance in NEI 00-04 and as shown in Table 3-1 of the LAR, any SSC categorized as HSS based on the defense-in-depth assessment is required to be categorized as HSS in the final categorization, and this cannot be changed by the integrated decision-making panel (IDP).

The NRC staff finds that the licensee's process for defense-in-depth is consistent with the NRC-endorsed guidance in NEI 00-04. Accordingly, the NRC staff concludes that the proposed change maintains the defense-in-depth philosophy. As a result, the staff finds that the proposed 10 CFR 50.69 program satisfies the second key principle for risk-informed decision-making as

described in Regulatory Guide 1.174, Revision 3, and meets the requirement of 10 CFR 50.69(c)(1)(iii) to preserve defense in depth.

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

The third key risk-informed principle in RG 1.174 states that the licensing basis change should maintain 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 SSCs to assure that sufficient safety margins are maintained. The guidelines used for making that assessment will include ensuring the categorization of the SSC does not adversely affect any assumptions or inputs to the safety analysis; or, if such inputs are affected, providing justification that sufficient safety margins will continue to exist. As stated by the licensee in Section 2.2 of the LAR enclosure states, in part, that, "The safety functions [in the categorization process] include the design basis functions, as well as functions credited for severe accidents (including external events)." Additionally, Section 3.1.1 of the LAR enclosure describes the collection of functional and risk significance information for different hazards and plant states, and documents that the SSC categorization process will include, among other items, system functions, their categorization bases, and mapping of components to supported functions.

The NRC staff notes that the design-basis functions of SSCs as described in the plants' licensing basis, including the Catawba Updated Final Safety Analysis Report (Reference [12]) and plant Technical Specification (TS) Bases, do not change and the safety margins described should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. Based on the above, the NRC staff concludes that safety margins are maintained by the proposed methodology, and the third key safety principle identified in RG 1.174, Revision 3, is satisfied. The NRC staff further finds that the process described in the LAR is consistent with NEI 00-04 as endorsed by the NRC in RG 1.201, Revision 1, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(c)(1)(iv).

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

The risk-informed considerations prescribed in NEI 00-04 address the fourth key principle of risk-informed decision-making identified in RG 1.174, Revision 3. A summary of how the licensee's SSC categorization process is consistent with the guidance and methodology in NEI 00-04 and RG 1.201, Revision 1, is provided in the sections below. The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel.

3.4.1 Probabilistic Risk Assessment

3.4.1.1 Scope of the PRA

The Catawba PRA is comprised of full-power, internal events PRA (IEPRA), Large Early Release Frequency (LERF) PRA, internal floods PRA (IFPRA), fire PRA (FPRA), and high winds PRA (HWPRA) which evaluates the core damage frequency (CDF) and LERF risk metrics. The licensee also states that the PRA models have been independently peer reviewed and there are no PRA upgrades that have not been peer reviewed.

In Section 3.3 of the LAR, "PRA Review Process Results (10 CFR 50.69(b)(2)(iii)),” the licensee provided a description of the peer reviews and the associated peer review Facts and Observations (F&O) findings closure reviews performed for the PRA models. F&O closure reviews were performed using the process documented in the NEI letter to the NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-13, 'Close-out of Facts and Observations (F&Os)'" (Appendix X) dated February 21, 2017 (Reference [13]), as accepted by the NRC in its letter dated May 3, 2017 (Reference [14]), and NEI 17-07 (Reference [15]), as endorsed by RG 1.200, Revision 3.

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 FLEX in the PRA, and (4) assessment of assumptions and approximations.

The information provided in the LAR is sufficient to support the NRC staff's review of the Catawba PRAs, and therefore, the NRC staff finds that it meets the requirements of 10 CFR 50.69(b)(2)(iii).

3.4.1.2 Internal Events, LERF Model, Internal Floods, and Internal Fires PRAs

Internal Events PRA

In Section 3.3 of the LAR, the licensee states that the internal events PRA (excluding LERF) model was subjected to a full-scope peer review in December 2015 using the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard RA-Sa-2009 (Reference [16]), as endorsed by RG 1.200, Revision 2. Subsequently, in 2017 and 2021, the licensee conducted Independent Assessments for closure of the finding-level F&Os against the requirements of Appendix X and NEI 17-07, respectively. In addition, focused-scope peer reviews were performed in 2017 and 2021 using the ASME/ANS PRA Standard RA-Sa-2009 and RG 1.200, Revision 2, to address model upgrades, and a follow-up F&O closure review was performed in 2022 against the requirements of NEI 17-07. The licensee concluded that all finding-level F&Os related to the IEPRA have been closed, all associated supporting requirements (SRs) meet Capability Category II, and there are no unreviewed PRA upgrades.

Large Early Release Frequency (LERF)

In Section 3.3 of the LAR, the licensee states that the LERF PRA model was subjected to a full-scope peer review in December 2012, using the ASME/ANS PRA Standard RA-Sa-2009, as endorsed by RG 1.200, Revision 2. Subsequently, in 2018 and 2022, the licensee conducted Independent Assessments for closure of the finding-level F&Os against the requirements of Appendix X and NEI 17-07, respectively. The licensee concluded that all finding-level F&Os related to the LERF PRA have been closed, all associated SRs meet Capability Category II, and there are no unreviewed PRA upgrades.

Internal Floods

In Section 3.3 of the LAR, the licensee states that the internal floods PRA model was subjected to a full-scope peer review in September 2012, using the ASME/ANS PRA Standard RA-Sa-2009, as endorsed by RG 1.200, Revision 2. Subsequently, in December 2018, the licensee conducted an Independent Assessment for closure of the finding-level F&Os against the requirements of Appendix X. The licensee concluded that all finding-level F&Os related to the

IFPRA have been closed, all associated SRs meet Capability Category II, and there are no unreviewed PRA upgrades.

Internal Fire

In Section 3.3 of the LAR, the licensee states that the FPRA model was subjected to full-scope peer review in 2010 using the ASME/ANS PRA Standard RA-Sa-2009, as endorsed by RG 1.200, Revision 2. Subsequently, the licensee conducted Independent Assessments for closure of the finding-level F&Os in August 2018, using Appendix X, and in October 2022, and December 2023, using NEI 17-07. Parallel focused-scope peer reviews were performed in August 2018 and October 2022 using the ASME/ANS PRA Standard RA-Sa-2009 and RG 1.200, Revision 2, to address model upgrades. The licensee concluded that all finding-level F&Os related to the FPRA have been closed, all associated SRs meet Capability Category II, and there are no unreviewed PRA upgrades.

Conclusion

In Section 3.3 of the LAR, for the IEPRA, IFPRA, LERF PRA, and FPRA, the licensee affirms there are no PRA upgrades that have not been peer reviewed. The NRC staff concluded that all peer reviews were conducted appropriately using the ASME/ANS PRA Standard RA-Sa-2009, as endorsed by RG 1.200, Revision 2. Finding-level F&Os were appropriately closed by Independent Assessments using Appendix X and NEI 17-07, as endorsed by the NRC, to ensure that no new methods or upgrades were incorporated inadvertently into these PRA models without a peer review.

The NRC staff performed a regulatory audit and reviewed the IEPRA, LERF PRA, IFPRA, and FPRA peer review documentation provided by the licensee. The NRC staff confirmed that the licensee applied the guidance for establishing PRA acceptability for these PRA models consistent with RG 1.200, Revision 3. Therefore, the NRC staff finds the Catawba IEPRA, LERF PRA, IFPRA, and FPRA models to be acceptable for use in the 10 CFR 50.69, SSC Categorization process.

3.4.1.3 High Winds PRA

In Attachment 4 of the LAR, the licensee stated that the high-winds PRA addresses risk from extreme winds and that the hazard screens from further consideration in its 10 CFR 50.69 program. In Section 4.4 of Enclosure 4 to the TSTF-505 LAR (Reference [17]) the licensee indicated that the Catawba high-wind events PRA model received a peer review in August 2013 using the PRA Standard ASME/ANS RA-Sb-2013. However, as documented in the LAR supplement dated March 26, 2026, there was a misstatement; the Catawba HWPRA model was actually peer reviewed in August 2013 against ASME/ANS RA-Sa-2009, using the guidance in NEI 12-13, Revision 0 (Reference [18]). Both ASME/ANS RA-Sa-2009 and NEI 12-13 are endorsed by Regulatory Guide 1.200, Revision 3, with Appendix X to NEI 12-13 accepted with conditions by the NRC in its letter dated May 3, 2017.

After the August 2013 peer review, an Independent Assessment for closure of F&Os was performed in October 2022, using the process documented in NEI 17-07, Revision 2, which resulted in closure of all finding-level F&Os. Therefore, the LAR does not identify any open finding-level F&Os and there are no unreviewed PRA upgrades. In addition, as stated in the LAR supplement dated March 26, 2026, there were no model assumptions or sources of uncertainty identified as key for high winds with respect to the 10 CFR 50.69 program.

Based on the above, the NRC staff finds that the Catawba high-winds PRA has been appropriately peer reviewed, that all the finding-level F&Os were closed using an NRC-approved approach, and that the high-winds PRA model maintenance is acceptable. Therefore, the NRC staff concludes that the Catawba high-winds PRA is acceptable for use in the Catawba 10 CFR 50.69 program.

3.4.1.4 Treatment of the Key Assumptions and Sources of Uncertainty

NUREG-1855, Revision 1, provides guidance on addressing PRA uncertainties to ensure that risk-informed decisions are made in the context of the specific application.

The licensee confirmed that sensitivity studies will be performed consistent with the guidance in NEI 00-04. In accordance with NEI 00-04, the results of these sensitivity studies are provided to the IDP for consideration in the final risk characterization, particularly for components initially classified as LSS that may be reclassified to HSS.

In Attachment 6 of the LAR, the licensee identified that there were no key assumptions or sources of uncertainty for this application, and, therefore, did not identify any additional sensitivity analyses required to address the Catawba PRA models.

The NRC staff recognizes that the licensee will perform routine PRA changes and updates to ensure the PRA continually reflects the as-built, as-operated plant, as well as conduct sensitivity analyses, as needed, to support the context of the analysis being performed. Paragraphs 50.69(e) and (f) stipulate the process for feedback and adjustment to ensure configuration control is maintained for these routine changes and updates to the PRA(s).

3.4.1.5 PRA Importance Measures and Integrated Importance Measures

Pursuant to 10 CFR 50.69(c)(1)(ii), the licensee's SSC categorization process is required to determine SSC functional importance using an integrated, systematic process that addresses both internal and external initiating events. Section 5 of NEI 00-04 provides guidance on the risk importance assessment process.

For Catawba, the scope of modeled hazards includes IEPRAs, LERF PRAs, IFPRAs, FPRAs, and HWPRAs. The NRC staff reviewed the LAR and found that the licensee's use and treatment of importance measures are consistent with the guidance in NEI 00-04, as endorsed by RG 1.201, Revision 1.

3.4.1.6 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 IEPRAs, LERF PRAs, IFPRAs, FPRAs, and HWPRAs 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 2009 PRA Standard using a peer-review process.

The licensee has subjected the IEPRAs, LERF PRAs, IFPRAs, FPRAs, and HWPRAs to the peer-review processes and submitted the results of those reviews. 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 NRC staff concludes that (1) the licensee's IEPRA, LERF PRA, IFPRA, FPRA, and HWPRA 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 3, and NUREG-1855, as applicable, and have been addressed appropriately for this application.

Based on the above, the NRC staff finds that the licensee provided the required information and that the IEPRA, LERF PRA, IFPRA, FPRA, and HWPRA are acceptable to support the categorization process, and therefore meets the requirements set forth in 10 CFR 50.69(c)(1)(i) and (ii).

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

The licensee's categorization process uses the following non-PRA methods, respectively:

- An alternative seismic approach for moderate seismic hazard/moderate seismic margin sites (Tier 2), as described in the Electric Power Research Institute (EPRI) report. (Note: Reference [19] defines a Tier 2 plant as "...Plants where the [ground-motion response spectra] GMRS to [safe shutdown earthquake] SSE comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited...")
- A screening analysis performed for the individual plant examination of external events (IPEEE) for external hazards (Reference [20]), updated to reflect part 6 of ASME/ANS PRA Standard RA-Sa-2009
- A safe shutdown risk management program consistent with NUMARC 91-06 (Reference [21])
- The Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization methodology for passive components (Reference [22])

The NRC staff's review of these methods is discussed below.

3.4.2.1 Alternative Seismic Approach

As part of its proposed process to categorize SSCs according to safety significance, the licensee proposed to use a non-PRA method to consider seismic hazards. The regulations in 10 CFR 50.69(b)(2)(ii) and 50.69(c)(1)(ii) permit the use of systematic evaluation techniques in the risk-informed categorization process. The licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and described how the proposed alternative seismic approach would be used in Section 3.2.3 of the enclosure to the LAR.

Specifically, the licensee based its plant-specific evaluation on case studies (also referred to as trial studies in the LAR) performed in EPRI 3002017583, stating that case studies A, C, and D are applicable to Catawba and are used in the alternative seismic approach. The licensee also described how the alternative seismic approach would be implemented in the categorization

process, and the measures in place to assure that the quality and level of detail of the approach are adequate for categorizing SSCs.

Based on the above, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(ii) and 50.69(c)(1)(ii) for the proposed alternative seismic approach are met.

Evaluation of the EPRI 3002017583 Case Studies

In Section 3.2.3 of the enclosure to the LAR, the licensee stated that the plant-specific case studies from other licensees in EPRI 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The licensee also indicated that it is following the same alternative seismic approach that was approved by the NRC staff for LaSalle County Station's 10 CFR 50.69 application, with noted exceptions, and referenced relevant LaSalle RAI responses and markups to EPRI 3002012988 as part of its technical basis.

The NRC staff independently reviewed and evaluated the technical acceptability of the PRAs used in the case studies for Plants A, C, and D in EPRI 3002017583, as well as the licensee's assertion of plant-specific applicability to the approach used in the amendment approved by the NRC for Calvert Cliffs on February 28, 2020 (Reference [23]). EPRI 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The licensee also states in Section 3.2.3 of the enclosure to its LAR that the Catawba LAR incorporates by reference a response to DRA/APLC RAI 03 for the Clinton Power Station 10 CFR 50.69 response to request for additional information (RAI) (Reference [24]) using the same alternative seismic approach.

The NRC staff's review confirmed that the case studies in EPRI 3002017583 used by the licensee to support its proposed alternative seismic approach provide sufficient plant-specific evaluation of the applicability of the case studies to Catawba. While the licensee's LAR references the LaSalle precedent and associated RAI responses as part of its technical justification, the NRC staff's findings for Catawba are based on an independent evaluation of the Catawba-specific submittal and supporting information, including site-specific hazard and plant characteristics, and are not dependent on prior NRC staff findings for LaSalle or any other plant.

The NRC staff finds that the technical acceptability of PRAs used for the Plant A, C, and D case studies in EPRI 3002017583, the mapping approach used in those case studies, and the conclusions regarding the determination of unique high-safety-significant (HSS) SSCs from the case studies are applicable to this licensee's proposed plant-specific alternative seismic approach. Therefore, the NRC staff concludes that the PRAs for Plants A, C, and D were technically acceptable and applicable for use in support of the licensee's proposed alternative seismic approach. The NRC staff determined that the mapping of SSCs between the seismic PRA, the full-power internal events PRA, and, as applicable, the fire PRA for the Plant A, C, and D case studies were appropriate. The NRC staff finds that the licensee's plant-specific evaluation is sufficient to determine unique HSS SSCs from seismic PRAs equivalent to the same process utilized in the Plant A, C, and D case studies from EPRI 3002017583. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed alternative seismic approach.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In the LAR, the licensee states, in part, that the GMRS peak acceleration for Catawba is below the SSE up to approximately 6 Hz and then exceeds the SSE. The licensee further states that

the NRC screened out Catawba from performing a seismic PRA to respond to Recommendation 2.1 of the Near-Term Task Force 50.54(f) letter. This demonstrates that Catawba is considered a Tier 2 plant under the criteria in EPRI 3002017583.

The NRC staff notes that the licensee's plant-specific evaluation is supported by its 10 CFR 50.54(f) response dated March 20, 2014 (Reference [25]). The NRC staff reviewed the licensee's submittal, supplement, and plant-specific evaluation and concludes that the proposed criteria in EPRI 3002017583 to determine the applicability and use of the proposed seismic Tier 2 approach are acceptable.

Evaluation of Applicability of Criteria for 10 CFR 50.69

In response to the 10 CFR 50.54(f) letter associated with post-Fukushima Near-Term Task Force (NTTF) Recommendation 2.1 (Reference [26]), the licensee submitted its seismic hazard screening report in a letter dated March 20, 2014. The NRC staff evaluation of the licensee's submittal (Reference [27]) included confirmatory analysis of the seismic hazard and concluded that the licensee's seismic hazard screening report was responsive to the 10 CFR 50.54(f) request and that the GMRS determined by the licensee adequately characterizes the reevaluated hazard for the Catawba site.

In its letter dated September 30, 2025, the licensee provided its basis for Catawba being a Tier 2 plant and states, in part, that,

As defined in Reference 17 [of the LAR] [19], [Catawba Nuclear Station] CNS meets the Tier 2 criteria for a "Moderate Seismic Hazard / Moderate Seismic Margin" site. The Tier 2 criteria are as follows:

"Tier 2: Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited."

Note: Reference 17 [of the LAR] applies to the Tier 2 sites in its entirety except for Sections 2.2 (Tier 1 sites) and 2.4 (Tier 3 sites).

For comparison, Tier 1 plants are defined as having a GMRS peak acceleration at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz. Tier 3 plants are defined where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 10 CFR 50.54(f) letter (Reference 20) [of the LAR] (Reference [26]).

The NRC did not require CNS to perform an SPRA as stated in its revised seismic screening and prioritization letter dated December 22, 2016 (Reference 37 [of the LAR] (Reference [27])).

As such, it is appropriate that CNS is considered a Tier 2 plant. The basis for CNS being classified as Tier 2 will be documented and presented to the CNS IDP for each system that is categorized.

In Section 3.2.3 of the enclosure to the LAR, "Seismic Hazards," the licensee stated that it compared the Catawba GMRS (derived from the seismic hazard) to the SSE (seismic

design basis capability) to demonstrate that the plant meets the criteria for application of the proposed alternative seismic approach.

Since the same hazard is used for comparison against the criteria for use of the proposed alternative seismic approach, the NRC staff's previous assessment on the reevaluated hazard is applicable to this review. The NRC staff finds that the plant's GMRS is above the Tier 1 criteria. Further, the NRC staff's review confirmed that the licensee did not perform a seismic PRA as part of the NRC's post-Fukushima actions.

In summary, the NRC staff finds that the licensee's basis for applying the proposed alternative seismic approach to its site is acceptable because the licensee meets the Tier 2 criteria for use of the proposed alternative seismic approach based on its seismic hazard screening report.

Evaluation of the Implementation of Conclusions from the Case Studies

The categorization conclusions from the case studies in EPRI 3002017583 indicate that seismic-specific failure modes resulted in HSS categorization uniquely from seismic PRAs. Therefore, seismic-specific failure modes, such as correlated failures, interaction failures, relay chatter, and passive component structural failure modes can influence the categorization process. The licensee discussed the implementation of its alternative seismic approach in its letter dated September 30, 2025, Section 3.2.3, "Seismic Hazards." The NRC staff reviewed this information to evaluate whether the categorization related to conclusions from the EPRI report was appropriately included and implemented.

The proposed alternative seismic approach includes a combination of qualitative and quantitative considerations of the mitigation capabilities as well as seismic failure modes of SSCs in the categorization process. These considerations are based on plant-specific walkdowns for the SSCs undergoing categorization, quantification of the impact of seismic failure of SSCs subject to correlated or interaction failures, and insights obtained from prior seismic evaluations performed for Catawba. Based on the above, the NRC staff finds that the licensee has sufficiently implemented the conclusions from the EPRI case studies in its alternative seismic approach.

Consideration of Changes to Seismic Hazard

The NRC staff recognizes that the seismic hazard at any site could potentially increase such that the categorization process may be impacted from a seismic risk perspective, either solely due to the seismic risk or via the integrated importance measure determination. In Section 3.2.3 of the enclosure to the LAR, the licensee stated that if the Catawba seismic hazard changed at some future time, and if its feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69, it will seek prior NRC approval for use of such an approach. The NRC staff notes that seeking prior NRC approval for the use of a process different from the proposed alternative seismic approach is consistent with the new license condition proposed by the licensee, as stated by the licensee in its letter dated September 30, 2025, Section 3.2.3. The licensee further stated that, after receiving NRC approval, it will follow its categorization review and adjustment process to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e). The NRC staff finds this approach acceptable, as it ensures continued regulatory oversight and compliance in the event of changes to the seismic hazard at the site.

3.4.2.2 Methods for Assessing Other External Hazards

This category includes all non-seismic external hazards other than fire and internal flooding. In Section 3.2.4 of the LAR, the licensee stated, in part, that all other external hazards (i.e., not seismic, fire, or internal flooding) were screened from applicability to Catawba through a plant-specific evaluation in accordance with GL 88-20 and were updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009.

In attachment 4 of the LAR, the licensee provided the results of its plant-specific evaluation that assessed the IPEEE results using the endorsed criteria in the ASME/ANS PRA Standard RA-Sa-2009 and current plant hazard information. The NRC staff notes that this plant-specific evaluation and its results were not peer reviewed against part 6 of the ASME/ANS PRA Standard Ra-SA-2009, as endorsed in RG 1.200, Revision 2.

In attachment 4 of the LAR, the licensee also provided the results for the external flooding hazard and stated that the LIP (local intense precipitation) was reevaluated and only found to exceed site grade at one location requiring a permanently installed barrier at door AX656B (Unit 2 Electrical Penetration Room door – Elevation 594 feet). The FE (focused evaluation) concluded that the station has adequate flood protection from the LIP hazard, which was supported by the NRC staff assessment of the FE (Reference [28]). The NRC staff finds this acceptable since this permanently installed flood protection was approved previously by the NRC staff assessment of the FE.

The external flooding hazard was screened using the screening criteria “C1 – design base” and “C3 - event can’t occur close enough to the plant to affect it.” Only the LIP hazard was found to exceed site grade in one location, but the hazard is mitigated by a permanently installed barrier. The NRC staff’s review finds that the licensee has appropriately considered the risk from external flooding and that the external flooding hazard has an insignificant contribution to risk.

In summary, the use of the Catawba IPEEE results described by the licensee in the LAR, information provided in the LAR supplement, and the licensee’s assessment of the other external hazards (i.e., high winds, tornadoes, and external flooding) is consistent with Section 5 of NEI 00-04 as endorsed in RG 1.201, Revision 1. Based on the above, the NRC staff concludes that the licensee’s treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

3.4.2.3 Shutdown Risk

Consistent with the guidance in NEI 00-04, the licensee proposed using the shutdown safety assessment based on NUMARC 91-06. NUMARC 91-06 provides considerations for maintaining defense-in-depth for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment-primary/secondary. NUMARC 91-06 also specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. This approach is accomplished by designating a primary (running) and an alternative system/train to accomplish each key safety function.

The use of NUMARC 91-06 described by the licensee in the submittal is consistent with the guidance in NEI 00-04 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 NRC 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).

3.4.2.4 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, passive components are defined as those components having only a pressure retaining function. This process also addresses the passive function of active components, such as the pressure or liquid retention of the body of a motor-operated valve.

In Section 3.1.2 of the LAR, the licensee proposed using a plant-specific categorization method, for passive components, that is not specifically cited in NEI 00-04 or RG 1.201, Revision 1, for passive component categorization, but which was authorized as a plant-specific alternative by the NRC for ANO-2. The ANO-2 precedent 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, Section XI, Division 1" (Reference [29]). The ANO-2 plant-specific authorization 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. As a result, the categorization will not be affected by changes in frequency arising from changes to the treatment. Based on its review of the Catawba plant-specific request and applicability, the NRC staff finds that the use of the ANO-2 repair/replacement precedent is acceptable for passive component categorization of Class 2 and Class 3 SSCs.

In Section 3.1.2, of the enclosure to its LAR, the licensee stated, in part, "the passive categorization process can apply the same risk-informed process accepted by the NRC in Reference 5 [of the LAR pertaining to Approval of Request for Alternative ANO2-R&R-004] for the passive categorization of Class 2, 3, and non-class components." Consistent with ANO2-R&R-004, all ASME Code Class 1 pressure retaining SSCs, as well as their supports, within the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP. That is, the ANO-2 repair/replacement methodology does not allow a Class 1 pressure retaining SSC to be recategorized from HSS to LSS. Based on the above, the NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.4.3 Assembly of Plant-Specific Inputs

The licensee's risk categorization process uses PRAs to assess risks from internal events (including internal floods and LERF) and internal fire hazards. The licensee evaluated high winds and screened them from consideration in the 10 CFR 50.69 categorization process. Non-PRA methods are used to assess shutdown safety, passive component risk, and the significance of other external hazards. The process for collecting and organizing information at the system level, including defining system boundaries, functions, and components, is described in the LAR, as supplemented, and is consistent with NEI 00-04. The NRC staff finds that, since the process is consistent with NEI 00-04 as clarified and endorsed in RG 1.201, Revision 1, it meets the requirements set forth in 10 CFR 50.69(c)(1)(v).

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

Section 3.1.1 of the LAR states that an unreliability factor of three will be used for the sensitivity studies described in Section 8, "Risk Sensitivity Study," of NEI 00-04. Section 3.2.7 of the LAR confirms that a cumulative sensitivity study will 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 the application of a factor of three for the sensitivities is consistent with the guidance in NEI 00-04 as endorsed by RG 1.201, Revision 1.

In Section 3.1.1 of the LAR, for the "Overall Categorization Process," the licensee stated that according to RG 1.201, "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)]." The NRC staff notes that this sensitivity study, together with the periodic review process discussed in Section 3.5.2 of this safety evaluation, ensures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in Section 8 of NEI 00-04, 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).

3.4.5 Integrated Decision-Making

Appendix B of SRP Chapter 19, Section 19.2 provides guidance to the NRC staff on expectations for the licensee's integrated decision-making process. The appendix states in part, "[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 NRC staff reviewed these two steps to ensure the processes is well-defined, systematic, repeatable, and scrutable and is described below.

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

In Section 3.1.1 of the LAR enclosure, the licensee acknowledged the NRC staff's clarification of NEI 00-04 that "...if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS." The licensee also states, "Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to LSS."

Based on the above, the NRC staff finds that the description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, and is, therefore acceptable.

3.4.5.2 IDP Review and Approval (NEI 00-04, Sections 9 and 10)

In Section 3.1.1 of the LAR enclosure, the licensee states:

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. 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 the modeling and updating of the plant-specific PRA.

The IDP will be trained in the specific technical aspects and requirements related to the categorization process. 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, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.

The guidance in NEI 00-04 as endorsed by the NRC in RG 1.201, 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). Based on the above, the NRC staff finds that the licensee's proposed IDP would have the expertise to 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 as endorsed by the NRC in RG 1.201.

3.4.6 Conclusion for Key Principle 4

The NRC staff reviewed the acceptability of the licensee's internal events PRA, internal floods and LERF, fire PRA, and HWPRA. The NRC staff also reviewed the use of PRA importance measures and integrated importance measures, the use of non-PRA methods, risk sensitivity studies, and integrated decision-making. Based on these reviews and the findings described in Sections 3.4.1 through 3.4.5 above, the NRC staff finds that the proposed change satisfies the fourth key principle for risk-informed decision-making identified in RG 1.174, Revision 3, and ensures that any potential increases in risk are small, as required by 10 CFR 50.69(c)(1)(iv).

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

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

3.5.1 Program Documentation and Change Control

Paragraph 50.69(f) of 10 CFR requires program documentation, change control, and records. In Section 3.2.6, "PRA Maintenance and Updates," of the LAR enclosure, the licensee stated that it will implement a process that addresses the requirements in Section 11 of NEI 00-04

pertaining to program documentation and change control records. In Section 3.1.1 of the LAR enclosure, the licensee states that the RISC categorization process documentation will include the following 10 elements:

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

The NRC staff also recognizes that for facilities licensed under 10 CFR Part 50, Appendix B Criterion VI, "Document Control," procedures are considered formal plant documents requiring that "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality."

The elements provided in Section 3.1.1 of the LAR enclosure, in addition to the list of items provided in Attachment 1 of the LAR enclosure, will ensure that the Catawba 10 CFR 50.69 categorization process is documented in formal licensee procedures. This is consistent with Section 11 of NEI 00-04, as endorsed by the NRC in RG 1.201, Revision 1. Based on the above, the NRC staff finds that the procedures will be sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control, and records.

3.5.2 Periodic Review (NEI 00-04, Section 12)

Paragraph 50.69(e) of 10 CFR requires that periodic updates to the licensee's PRA and SSC categorization must be performed. 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 3.2.6, "PRA Maintenance and Updates," of the LAR enclosure, the licensee described the process for maintaining and updating the Catawba PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, the licensee confirmed that the Catawba risk management process ensures that the 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). The process also includes provisions for assessing the risk impact of unincorporated changes and for controlling the model and associated computer files.

Section 12.1 of NEI 00-04 states, in part, "[s]cheduled periodic reviews (e.g. once per two fuel cycles in a unit) should evaluate new insight resulting from available risk information." In Section 3.5, of the LAR enclosure, the licensee states:

Scheduled periodic reviews are completed at least once every two refueling cycles in accordance with Duke Energy procedures and will evaluate new insights resulting from available risk information (i.e., PRA model or other

analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated.

The NRC staff finds that the risk management process described by the licensee in the LAR is consistent with the guidance in Section 12 of NEI 00-04. Based on the above, the NRC staff finds that the proposed change satisfies the fifth key principle for risk-informed decision-making identified in RG 1.174, Revision 3.

4.0 CHANGES TO THE OPERATING LICENSE

The licensee proposed the following amendment to the Renewed Facility Operating Licenses for Catawba, Units 1 and 2. The proposed license condition states:

Duke Energy 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, including 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, Class 3, and non-class SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic, and the alternative seismic approach described in Duke Energy's submittal letter RA-18-0089 dated September 30, 2025; as specified in License Amendment No. [XXX] dated [DATE].

Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).

The NRC staff finds that the proposed license condition is acceptable because it adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with the applicable NRC regulations and NRC-endorsed guidance.

The NRC staff notes that the guidance for implementing 10 CFR 50.69 provided by the Commission in the *Federal Register* notice dated November 22, 2004 (69 FR 68008, 68028-68029), Section III.4.10.2, "Section 50.36 Technical Specifications," stated that the 10 CFR 50.69 rule does not include 10 CFR 50.36 in the list of special treatment requirements that may be replaced by the alternative 10 CFR 50.69 requirements for RISC-3 and RISC-4 SSCs when implementing a 10 CFR 50.69 license amendment. As a result, the NRC staff does not consider the technical specifications (including Improved Technical Specifications (ITS) and the associated Technical Requirements Manual) to be part of the 10 CFR 50.69 rule. Therefore, the licensee needs to address proposed changes to its technical specifications separately.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendment on March 23, 2026. On March 23, 2026, the State official confirmed that the State of South Carolina had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This action relates to authorizations under 10 CFR Part 50 with respect to installation or use of a facility component. The NRC staff has determined that any ground disturbance is limited to previously disturbed areas. Additionally, the NRC staff has determined that the action involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, no significant increase in individual or cumulative public or occupational radiation exposure, and no significant increase in the potential for or consequences from radiological accidents. Finally, the NRC staff has determined that a categorical exclusion applies and that special circumstances under 10 CFR 51.22, "Categorical exclusions," are not present that would preclude reliance on the categorical exclusion. Accordingly, this action meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(d)(8). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment need be prepared in connection with the action.

7.0 CONCLUSION

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

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SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 325 AND 321, REGARDING ADOPTION OF 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2025-LLA-0156) DATED APRIL 30, 2026

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