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

July 3, 2024

David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 301 AND 297 RE: ADOPTION OF 10 CFR 50.69 "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2023-LLA-0085)

Dear David Rhoades:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 301 to Renewed Facility Operating License No. DPR-29 and Amendment No. 297 to Renewed Facility Operating License No. DPR-30 for Quad Cities Nuclear Power Station, Units 1 and 2. The amendments consist of changes to Renewed Facility Operating Licenses in response to your application dated June 8, 2023, as supplemented by letters dated March 19, 2024, and May 10, 2024.

The amendments adopt Title 10 of the *Code of Federal Regulations*, section 50.69, "Risk Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

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

Sincerely,

/RA/

Robert Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-254 and 50-265

Enclosures:
1. Amendment No. 301 to DPR-29
2. Amendment No. 297 to DPR-30
3. Safety Evaluation

cc: Listserv

CONSTELLATION ENERGY GENERATION, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 301
Renewed License No. DPR-29

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

   A. The application for amendment by Constellation Energy Generation, LLC (the licensee) dated June 8, 2023, as supplemented by letters dated March 19, and May 10, 2024, 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 the addition of paragraph 3.CC. to Renewed Facility Operating License No. DPR-29 which reads as follows:

CC. Constellation Energy Generation, LLC 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 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 Constellation's submittal letter dated June 8, 2023, and all its subsequent associated supplements as specified in License Amendment No. 301 dated July 3, 2024

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 the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2024.07.03 13:45:18 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: July 3, 2024

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

CONSTELLATION ENERGY GENERATION, LLC

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 297
Renewed License No. DPR-30

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

   A. The application for amendment by the Constellation Energy Generation, LLC (the licensee) dated June 8, 2023, as supplemented by letters dated March 19, and May 10, 2024, 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 the addition of paragraph 3.BB. to Renewed Facility Operating License No. DPR-29 which reads as follows:

   BB. Constellation Energy Generation, LLC 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 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 Constellation's submittal letter dated June 8, 2023, and all its subsequent associated supplements as specified in License Amendment No. 297 dated July 3, 2024.

   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 the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jeffrey A. Whited

Digitally signed by Jeffrey A. Whited
Date: 2024.07.03 13:45:37 -04'00'

Jeffrey A. Whited, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: July 3, 2024

ATTACHMENT TO LICENSE AMENDMENT NOS. 301 AND 297

RENEWED FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

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

Remove	Insert
License DPR-29 Page 10	License DPR-29 Page 10
License DPR-30 Page 10	License DPR-30 Page 10

(4) Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, Constellation Energy Generation, LLC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm2) is met for each insert; and,

(5) Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

CC. Constellation Energy Generation, LLC 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 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 Constellation's submittal letter dated June 8, 2023, and all its subsequent associated supplements as specified in License Amendment No. 301 dated July 3, 2024

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).

4. This renewed operating license is effective as of the date of issuance and shall expire at midnight on December 14, 2032.

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 28, 2004

(4) Ensure that if any inserts are identified as potentially failing the minimum certified Boron-10 areal density criterion, based on correlation of the coupon evaluation or insert service wear evaluation results to inserts, or other abnormal indications, Constellation Energy Generation, LLC will take affected inserts out of service until it can be positively demonstrated that the minimum certified Boron-10 areal density criterion (0.0116 g/cm2) is met for each insert; and,

(5) Submit a report to the NRC, within 90 days following completion of evaluations associated with Item 4 above, that describes the testing results, assessments performed, and interim and long-term corrective actions for abnormal indications.

BB. Constellation Energy Generation, LLC 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 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 Constellation's submittal letter dated June 8, 2023, and all its subsequent associated supplements as specified in License Amendment No. 297 dated July 3, 2024.

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).

4. This renewed operating license is effective as of the date of issuance and shall expire at midnight on December 14, 2032.

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 28, 2004

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 301 TO RENEWED FACILITY OPERATING LICENSE

NO. DPR-29 AND AMENDMENT NO. 297 TO RENEWED FACILITY OPERATING

LICENSE NO. DPR-30

CONSTELLATION ENERGY GENERATION, LLC

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC, the Commission) dated June 8, 2023, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23159A253), Constellation Energy Generation, LLC (CEG, the licensee) submitted a license amendment request (LAR). The LAR was supplemented by letters dated March 19 (ML24079A122), and May 10, 2024 (ML24131A079). The LAR, as supplemented, requested changes to the renewed facility operating licenses for the Quad Cities Nuclear Power Station, Units 1 and 2 (Quad Cities). The proposed changes would implement Title 10 of the *Code of Federal Regulations* (10 CFR), section 50.69, “Risk-informed categorization and treatment of structures, systems and components [SSCs] for nuclear power reactors” by adding the following license condition:

> Constellation Energy Generation, LLC 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 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE [individual plant examination for external events] screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS [American Society of Mechanical Engineers/American Nuclear Society] PRA Standard RA-Sa-2009 for other external hazards except seismic;

> and the alternative seismic approach as described in Constellation's submittal letter dated, June 8, 2023; and all its subsequent associated supplements 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 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].

The March 19, 2024, and May 10, 2024, supplements, 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 August 8, 2023 (88 FR 53537).

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.[2] 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.

In order for a licensee to adopt alternate treatments, section 50.69 of 10 CFR contains the requirements describing how to categorize SSCs using a risk-informed process; how to adjust treatment requirements consistent with the relative significance of the SSC; and how to manage 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.

The SSC categorization does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, 10 CFR 50.69 enables licensees to focus their resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as HSS, existing treatment requirements are maintained or potentially enhanced. Conversely, for SSCs categorized as LSS

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[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.

[2] The NRC staff notes that statements of consideration implementing 10 CFR 50.69 published in the Federal Register on November 22, 2004 (69 FR 68008, 68028-68029), Section III.4.10.2, "Section 50.36 Technical Specifications," stated that the Commission considered 10 CFR 50.36 as a candidate special treatment rule when developing 10 CFR 50.69. The Commission concluded it was not appropriate to revise 10 CFR 50.36 as part of the adoption of 10 CFR 50.69. Therefore, the requirements for TSs are unchanged by the adoption of 10 CFR 50.69.

that do not significantly contribute to plant safety on an individual basis, the regulation allows an alternative risk-informed approach to treatment that provides a reasonable level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve focus on equipment that has HSS.

2.2 Regulatory Guidance

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

- RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (ML061090627)

- RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (ML090410014),

- RG 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (ML17317A256),

- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking” (ML17062A466), and

- NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-water Reactor] Edition,” Chapter 19, Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance” (ML071700658)

2.2.1 NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline” (ML052910035), as endorsed by RG 1.201 for trial use with clarifications and describes a categorization 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 12 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 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). Also, 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. 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 change results in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

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

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3, and 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 Bases 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[3]
RISC–2: Nonsafety-related SSCs that perform safety significant functions
RISC–3: Safety-related SSCs that perform low safety significant functions
RISC–4: Nonsafety-related SSCs that perform low safety significant functions

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

Paragraph 50.69(b)(3) of 10 CFR states that the Commission will approve a licensee's implementation of this section by issuance of a license amendment if the Commission determines that the categorization process satisfies the requirements of 10 CFR 50.69(c). As stated in 10 CFR 50.69(b), after the NRC issues a license amendment to implement 10 CFR 50.69, licensee or applicant specified under 10 CFR 50.69(c)(1) 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 in RG 1.201, Revision 1, 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 (SE), used the framework provided in RG 1.174, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, including clarifications described in Section C.9 of RG 1.201.

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

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)
2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)

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[3] 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" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. DID Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. Integrated Decisionmaking Panel Review and Approval (Section 9 of NEI 00-04, Revision 0)
8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

Section 3.1.1 of the LAR stated that the risk-informed categorization process will be implemented in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. Sections 3.2.3 and 3.2.4 of the enclosure to the LAR, proposed the use of the Electric Power Research Institute (EPRI) Tier 1 alternate seismic approach as an alternative method to assess the applicable hazard contributions. The NRC notes that use of this alternative method is a deviation from the NEI 00-04 guidance as endorsed. A more detailed NRC staff review of the alternative method is provided in section 3.3 of this SE.

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

The regulatory requirements in 10 CFR 50.69 and 10 CFR part 50, appendix B, implemented in accordance with the guidance for monitoring outlined in NEI 00-04, Revision 0, and clarifications in RG 1.201, Revision 1, ensures that the SSC categorization process is sufficient to assure that the SSC functions continue to be met and that any performance deficiencies will be identified, and appropriate corrective actions taken. The licensee's SSC categorization program includes the appropriate steps/elements described in NEI 00-04, Revision 0, to assure 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 consistency with the NEI 00-04 guidance, as endorsed. Based on the above, the NRC staff concludes that the proposed program to implement risk-informed categorization and treatment of SSC meets the first key principle of RG 1.174 for risk-informed decision making.

3.2.2 Key Principle 2: Licensing Basis Change is Consistent With the DID Philosophy

In RG 1.174, Revision 3, the NRC identified the following considerations used for evaluating 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 paragraph 50.69(b)(1)(x) of 10 CFR are not to determine the proper RISC category for containment isolation valves or penetrations.

Section 3.1.1 of the enclosure to the LAR, clarified that if the DID assessment determines that the SSC must be categorized as HSS, the SSC will be categorized as HSS in accordance with NEI 00-04, Revision 0. The NRC staff finds that the licensee's process is consistent with DID philosophy and the NRC-endorsed guidance in NEI 00-04; therefore, Key Principle 2 of risk-informed decision-making in RG 1.174, Revision 3, is met and fulfills the 10 CFR 50.69(c)(1)(iii) criterion that requires DID 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 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, justification is provided to ensure sufficient safety margin will continue to exist.

The SSCs design-basis function as described in the plants' licensing basis, including the Updated Final Safety Analysis Report (UFSAR) and technical specifications (TSs) bases, do not change and, therefore, will continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. On this basis, the NRC staff concludes that the licensee's SSC categorization program ensures sufficient safety margins are maintained in accordance with the third key safety principle of RG 1.174, Revision 3, and therefore, 10 CFR 50.69(c)(1)(iv) is satisfied.

3.2.4 System Engineering Assessment (NEI 00-004, Revision 0, Section 4)

Section 4 of NEI 00-04, Revision 0, describes a process for system selection and boundary definition, identification of system functions, and mapping of components to functions. In section 2.2 of the enclosure to the LAR, "Reason for Proposed Change," the licensee identifies the selected systems by stating, in part, that "[t]he safety functions [in the categorization process] include the design basis functions, as well as functions credited for severe accidents (including external events)." Table 3-1 of the enclosure to the LAR comprises a flow chart in which the licensee will define system boundaries and define system functions. Section 3.1.1 of the enclosure to 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 enclosure to the LAR, the licensee also confirmed that "the mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components."

Based on its review, the NRC staff finds that the process described in the LAR is consistent with NEI 00-04, Revision 0. RG 1.201, Revision 1, states, in part, "this trial regulatory guide provides interim guidance for complying with the NRC's requirements in §50.69, by using the process described in Revision 0 of NEI 00-04." Because the process described in LAR meets the guidance in NEI 00-04, the NRC staff finds that the process meets the requirements set forth in 10 CFR 50.69(c)(1)(ii) and 10 CFR 50.69(c)(1)(iv).

3.3 Risk-Informed Assessment

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

The risk-informed considerations described in NEI 00-04, Revision 0, endorsed by RG 1.201, Revision 1 addresses the fourth and fifth key principles of the staff's standards for risk-informed decision making, pertaining to the assessment for change in risk and monitoring the impact of the LB change.

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

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

The NRC staff acknowledges that elements of the categorization process (e.g., system selection, system boundary definition, identification of system functions, and mapping of components to functions) are not always performed in chronological order and may be performed in parallel. This is further discussed in Section 3.2 of this SE. The licensee's risk categorization process uses probabilistic risk assessments (PRAs) to assess risks from the internal events (IEPRA) (including internal flooding), and fire (FRPA). For non-PRA methods that depart from the methodology prescribed in NEI 00-04, additional staff review is discussed in this section of the SE.

Section 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. Section 3.1.1 of the enclosure to the LAR states, in part, that "CEG will implement the risk categorization process in accordance with NEI 00-04, as endorsed by Regulatory Guide 1.201." Section 3.1.1 of the enclosure also describes an overall method for selecting systems and system boundaries consistent with NEI 00-04, Revision 0. Because NEI 00-04, Revision 0, was endorsed as an acceptable means to comply with the requirements of 10 CFR 50.69, the NRC staff finds the process described in the LAR, as supplemented, meets the requirements set forth in 10 CFR 50.69(c)(1)(v).

3.3.1.2 Component Safety Significance Assessment (NEI 00-04, 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 the NEI 00-04 guidance, 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., external floods)

- other hazards
- shutdown events
- passive categorization

In Sections 3.2.1 and 3.2.2 of the enclosure to the LAR, the licensee described that the Quad Cities categorization process uses PRA modeled hazards to assess risks for the internal events (includes internal flood) 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 report 3002017583 dated February 29, 2020 (ML21082A170), and qualitative insights about seismic risk at Quad Cities.

- Other External Hazards: Screening analysis performed for IPEEE in according with Generic Letter (GL) 88-20 (ML21082A170) and updated using criteria from Part 6 of the ASME/ANS-RA-Sa-2009, "Addendum A to RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009 (the PRA Standard), as endorsed by the NRC in Appendix A of RG 1.200.

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06 (ML14365A203).

- Passive Components: ANO-2 passive categorization methodology (ML090930246).

The approaches and methods proposed by the licensee to address internal events, 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 for passive components is consistent with the ANO-2 methodology for passive components approved for risk-informed safety classification and treatment for repair/replacement activities in class 2 and 3 moderate- and high-energy systems. The use of the ANO-2 methodology in the SSC categorization process is provided in section 3.3.1.4 of this SE. To address seismic hazard in the SSC categorization process, the licensee proposed to use an alternative method not endorsed by the NRC in NEI 00-04. A detailed NRC staff review of the licensee's proposed alternative seismic approach is provided in section 3.3.1.4 of this SE.

3.3.1.3 Scope of the PRA

The Quad Cities PRA comprises a full-power, internal events PRA (including internal flooding) and FPRA. Each one of these assessments evaluates risk metrics of core damage frequency (CDF) and large early release frequency (LERF).

The NRC staff finds that the information regarding the PRA review process provided in the LAR, as supplemented, is of sufficient detail to support the staff review of the technical acceptability of the IEPRA and FPRA and therefore, meets the requirements set forth in paragraph 50.69(b)(2)(iii) of 10 CFR.

Aspects considered by the staff to evaluate the scope of the PRA include: (1) peer-review history and results, (2) the independent assessment process, (3) credit for diverse and flexible coping strategies (FLEX) in the PRA, and (4) assessment of assumptions and approximations. The staff's review of these aspects of the PRA to assess for consistency with the applicable processes as endorsed by the NRC, where necessary, are provided below.

Evaluation of PRA Acceptability for Internal Events and Internal Fires

Internal Events PRA (Includes internal flooding)

In Enclosure 2, Section 3, of the LAR, the licensee explains that the internal events PRA model was subjected to a full-scope peer review in April 2017 against RG 1.200, Revision 2**.**

Subsequently, the licensee conducted independent assessments in February/March of 2021 to close the finding-level facts and observations (F&Os) using the Appendix X 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,” dated February 21, 2017 (ML17086A431). All finding-level F&Os were reviewed and closed using this NRC-accepted process. Hence, the LAR does not identify any open finding-level F&Os.

The NRC staff finds that the Quad Cities IEPRA (that includes internal flooding) was appropriately peer reviewed consistent with RG 1.200, Revision 2, and that all finding-level F&Os have been closed consistent with the Appendix X process guidance, as accepted, with conditions by the staff. Therefore, the staff concludes that the IEPRA (that includes internal flooding) is acceptable for use in the 10 CFR 50.69 Program.

Internal FPRA

In Section 4 of Enclosure 2 to the LAR, the licensee confirmed that the Quad Cities internal FPRA (IFPRA) model received a full-scope peer review in June 2013 using the ASME/ANS RA-Sa-2009 PRA Standard, and RG 1.200, Revision 2. After the peer review, focused-scope peer reviews were conducted in February and May 2021. Subsequent independent assessment for closure of F&Os using the Appendix X process, as accepted, with conditions by the NRC staff, was performed in March and May 2021, which resulted in closure of all but one finding-level F&O, F&O 9-1. In response to APLB RAI [request for additional information] 11 in the March 19, 2024, LAR supplement, the licensee provided the peer review description of the finding and recommended actions to close out the finding. Based on the information provided, the NRC staff concluded that the open F&O 9-1 does not impact this application since the remedy required updating documentation and required no PRA model updating.

In recent years the industry has submitted multiple FPRAs for NRC staff review. During this period both the industry and NRC staff determined that updated guidance, methods, and data could be needed for a licensee to develop a more realistic fire model. To ensure that FPRA models not previously evaluated by staff are in alignment with current guidance, methods, and data, the NRC staff provided a series of APLB RAI questions to ascertain the status of Quad Cities fire model. The licensee’s responses to APLB RAIs 01, 02, 03, 05, 06, 07, 08, 09, and 10, provided in the March 19, 2023, LAR supplement demonstrated that the Quad Cities fire model has incorporated all of the updated guidance, methods, and data that are applicable to Quad Cities.

The NRC staff finds that the Quad Cities FPRA was appropriately peer reviewed consistent with RG 1.200, Revision 2, and that all finding-level F&Os have either been closed consistent with the Appendix X process guidance or determined not to impact this application. Therefore, the staff concludes that the FPRA is acceptable for use in the 10 CFR 50.69 program.

Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3 of Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13 provides guidance to perform an Independent Assessment for the closure of F&O identified from a full-scope or focused-scope peer review.

In review of the LAR, the NRC staff concluded that all F&Os were appropriately assessed by the Independent Assessment team to assure that no new methods or upgrades were inadvertently incorporated into the IEPRA without a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA standard as endorsed by the NRC. Therefore, the NRC staff finds that the Quad Cities IEPRA (includes internal floods) and FPRA were appropriately peer reviewed consistent with RG 1.200, Revision 2 and meets the requirements set forth in 10 CFR 50.69(c)(1)(i).

Credit for FLEX Equipment

The NRC memorandum dated May 30, 2017, “Assessment of the Nuclear Energy Institute16-06, ‘Crediting Mitigating Strategies in Risk-Informed Decision Making,’ Guidance for Risk-Informed Changes to Plants Licensing Basis” (ML17031A269), 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.

In Attachment 6 of the LAR, the licensee notes a key source of assumption and uncertainty relates to FLEX and the PRA models FLEX component failures and human failure events associated with failure to align FLEX equipment. In addition, the LAR indicates FLEX component failures are estimated based on “like-components” by increasing that like-component’s failure rate by a factor of two and human error probabilities employ screening values.

In response to APLA RAI 07 in the March 19, 2024, LAR supplement the licensee provided details on how the Quad Cities PRA modeling of FLEX addresses the NRC staff uncertainty concerns regarding the modeling of FLEX. The licensee provided the results of a sensitivity study that removed all FLEX credit from the Quad Cities PRA models. The results of the study demonstrate that modeling of FLEX has a minimal impact on CDF and LERF metrics.

Based on the above, the NRC staff finds that the uncertainties associated with PRA modeling of FLEX equipment and operator actions have been adequately addressed consistent with NEI 00-04 guidance, as endorsed by RG 1.201, to support the Quad Cities 10 CFR 50.69 categorization process.

Assessment of Assumptions and Approximations

Identification of Key Assumptions and Sources of Uncertainty

The LAR, as supplemented, stated that NUREG-1855, Revision 1, was performed to identify, screen, and characterize those sources of model uncertainty and related assumptions in the base PRA that are relevant to this application. Substep E-1.4 of the guidance is a qualitative screening process that involves identifying and validating whether consensus[4] models have been used in the PRA to evaluate identified model uncertainties. The licensee confirmed that for the Quad Cities uncertainty analysis, some uncertainties and assumptions were screened based on the use of a consensus method. 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.

Treatment of the Key Assumptions and Sources of Uncertainty

The licensee considered PRA modeling uncertainties and their potential impact on the 10 CFR 50.69 program. In Attachment 6 of the LAR the licensee discussed the identification of key assumptions and sources of uncertainty along with providing the dispositions for impact on the risk-informed application of applicable sensitivities. The licensee evaluated the Quad Cities PRA model to identify the key assumptions and sources of uncertainty for this application consistent with the RG 1.200, Revision 2, definitions, using sensitivity and importance analyses to place bounds on uncertain processes, to identify alternate modeling strategies, and to provide information to users of the PRA.

In response to APLA RAI 01 in the March 19, 2024, LAR supplement, the licensee provided clarification on the sensitivity study conducted related to the data uncertainty of digital components, specifically the Quad Cities digital feedwater control system (DFWCS). The PRA model basic events used in the study represented all of the failure modes, reactor vessel overflow and loss of flow, of the DFWCS. The NRC staff determined that the sensitivity study performed by the licensee related to the DFWCS appropriately bounded the uncertainty and that the insights from the sensitivity study reasonably show that this source of uncertainty does not impact this application.

In response to APLA RAI 02 in the March 19, 2024, LAR supplement, the licensee provided an updated sensitivity study regarding the uncertainty related to successful core cooling following containment failure where the original sensitivity demonstrated an impact ranging from 33 to 81 percent on CDF or LERF. The licensee stated that the original sensitivity was overly conservative in that the likelihood of unsuccessful core cooling was increased by a factor of ten. The updated sensitivity provided in response used a factor of two increase. The results of the study demonstrated that this source of uncertainty did not significantly impact the application. The NRC staff determined increasing the probability of failure by a factor of two was reasonable given in part the Quad Cities PRA model probability is a conservative value based on industry studies.

In response to APLB RAI 04 in the March 19, 2024, LAR supplement, the licensee provided the results of a sensitivity study that addressed the use of a 1×10^{-6} floor value for joint human error probabilities (JHEP) in the Quad Cities FPRA model used to categorize components when a value of 1×10^{-5} is recommended. The results of the study, which applied the 1×10^{-5} floor value, demonstrated that the JHEP floor value of 1×10^{-6} has a minimal impact. Based on the results of the sensitivity study the NRC staff determined that the use of a 1×10^{-6} JHEP floor value does not impact this application.

Based on the NRC staff's review of the licensee's dispositions provided in Attachment 6 to the LAR, as supplemented, the staff finds that the licensee performed an adequate assessment to

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[4] Per NUREG-1855, Revision 1, a consensus model is a model that has a publicly available published basis and has been peer reviewed and widely adopted by an appropriate stakeholder group

identify the potential sources of uncertainty, and that the identification of the key assumptions and sources of uncertainty was appropriate and consistent with the guidance in NUREG-1855, Revision 1 and associated EPRI TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments”, December 2008 and EPRI 3.2.4.1.6 PRA Scope and Acceptability Conclusions TR-1026511, “Practical Guidance of the Use on Probabilistic Risk Assessment in the Risk-informed Treatment of Uncertainty”, December 2012. Therefore, the NRC staff finds the licensee has satisfied the guidance in RG 1.177, Revision 2, and RG 1.174, Revision 3, and that the identification and treatment of assumptions and treatment of model uncertainties for risk evaluation to conduct categorizations appropriate for this application and is consistent with the NEI 00-04 guidance.

PRA Importance Measures and Integrated Importance Measures

The scope of modeled hazards for Quad Cities includes the IEPRA (includes internal floods) and FPRA. The NRC staff finds that the licensee’s use and treatment of importance measures is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. A more detailed staff review of the alternate methods for assessing the risk for seismic, shutdown, passive components, and other hazards is provided in section 3.3.1.2 of this SE.

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 and FPRA to support SSC categorization is endorsed by RG 1.201, Revision 1. Furthermore, pursuant to 10 CFR 50.69 (c)(1), the PRAs must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer-review process assessed against a standard that is endorsed by the NRC. 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 Standard using a peer-review process.

The licensee has subjected the IEPRA 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 NRC staff concludes that (1) the licensee's IEPRA and FPRA are acceptable to support the categorization of SSCs using the process endorsed by the NRC staff in RG 1.201, Revision 1, and (2) the key assumptions for the PRAs have been identified consistent with the guidance in RG 1.200, Revision 2, and NUREG-1855, as applicable, and addressed appropriately for this application.

The NRC staff finds the licensee provided the required information, and the IEPRA (includes internal floods) and FPRA, is acceptable and therefore meets the requirements set forth in 10 CFR 50.69(c)(1)(i).

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

The licensee’s categorization process uses the following non-PRA method(s), respectively:

- EPRI Technical Update 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," Tier 1.

- Screening analysis performed for the IPEEE for other external hazards (e.g., external flood); and updated using the external hazard screening significance process identified in the ASME/ANS PRA Standard.
- Safe Shutdown Risk Management program consistent with NUMARC 91-06.
- Passive Components: ANO-2 passive categorization.

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

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) requires a description of the measures taken to assure that the quality and level of detail of the systematic evaluation techniques in the risk-informed categorization process to be included in the application. The licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and described how the proposed alternative seismic approach would be used in the categorization process in Section 3.2.3 of the LAR. 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 Quad Cities 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 assuring the quality and level of detail for the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Therefore, based on the above, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(ii) for the proposed alternative seismic approach are met.

EPRI report 3002017583 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from seismic PRAs (SPRAs). The licensee states in Footnote 1, Item #2 of Section 3.2.3 of the LAR that the LAR incorporates a prior and similar response to APLC RAI 03 for the Clinton Power Station 10 CFR 50.69 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 provided sufficient plant-specific evaluation of the applicability of the case studies to Quad Cities. The information presented in the LAR provided a sufficient description of, and basis for acceptability of the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv) for the alternative seismic approach. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69(b)(2)(iv) are met for the proposed alternative seismic approach.

Evaluation of the EPRI 3002017583 Case Studies

In Section 3.2.3 of the LAR, the licensee stated that the plant-specific case studies from other licensees in the EPRI report 3002017583 are incorporated by reference to support its proposed alternative seismic approach. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in case studies for plants A, C, and D, in EPRI report 3002017583, and 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 (ML19330D909).

The NRC staff finds that the acceptability of 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 in the Calvert Cliffs amendment are applicable to this licensee's proposed plant-specific alternative seismic approach.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

In the LAR, the licensee states, in part, that the ground motion response spectrum (GMRS) peak acceleration for Quad Cities is below the safe shutdown earthquake (SSE) between 1.0 Hz (Hertz) and 10 Hz, which demonstrates that Quad Cities qualifies as a Tier 1 plant under the criteria in EPRI 3002017583.

The NRC staff notes that the licensee's plant-specific evaluation is supported by its letter dated March 31, 2014 (ML14090A526). The NRC staff reviewed the LAR, as supplemented, and plant-specific evaluation and concludes that the proposed criteria in EPRI report 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 Section 3.2.3 of the LAR, the licensee compared the Quad Cities GMRS from the reevaluated seismic hazard developed and submitted by the licensee in response to Near-Term Task Force (NTTF) Recommendation 2.1 against the site's design basis SSE, as shown in Figure A4-1 of Attachment 4 of the LAR to demonstrate that the site meets the criteria for application of the proposed alternative seismic approach as a Tier 1 plant. The staff notes that the Quad Cities UFSAR discusses the two primary design and licensing basis spectra that represent the SSE approved during original licensing. For this LAR review, due to the non-standard shape of the SSE curve, the NRC staff reviewed in detail the licensee's 10 CFR 50.54(f) response in the March 31, 2014 letter and the NRC staff's assessment of the 10 CFR 50.54(f) response provided by letter dated February 10, 2016 (ML15309A493). In the letter dated March 31, 2014, the licensee states "The GMRS (Table 2.4-1) is compared to the 5% critical damping SSE Golden Gate Park (Table 3.1-1) and Housner (Table 3.1-2) spectra in the frequency range from 1 Hz to10 Hz for the screening of structures which were based on enveloped results of the Golden Gate Park and Housner earthquakes. The Housner spectrum envelops the GMRS at frequencies less than 3.77 Hz, which was the controlling frequency range for the Housner spectrum in the original design analysis. In the frequency range greater than 3.77 Hz for which the Golden Gate Park spectrum controlled the original design, the Golden Gate Park spectrum envelopes the GMRS. Therefore, the Quad Cities controlling SSE is greater than the GMRS in the 1 Hz to 10 Hz range for structures qualified using the enveloped results of the Golden Gate Park and Housner spectra. The NRC staff's review as stated in its February 10, 2016, letter confirmed the licensee's statements and the comparison of the GMRS from the reevaluated seismic hazard against the SSE. Based on its review, the NRC staff finds that the licensee's seismic hazard meets the criteria for 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. Section 2.2.2 of EPRI report which 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 Quad Cities TSTF-505 LAR (ML23159A249). The NRC staff noted that seismic core damage frequency (SCDF) contribution to total plant CDF for Quad Cities is low (i.e., approximately

9 percent for Unit 1 and 8 percent for Unit 2). The NRC staff reviewed seismic large early release frequency (SLERF) estimate in the Quad Cities TSTF-505 submittal and noted that SLERF for Quad Cities is about 37 percent for Unit 1 and 35 percent for Unit 2 of total plant LERF, respectively. The purpose of the seismic risk estimate in the Quad Cities TSTF-505 submittal is to provide a conservative estimate for use in calculating risk-informed completion times for technical specifications.

Further, as noted in Section 3.6.5 of EPRI report 3002017583, containment DID assessment addresses containment failures and containment bypass situations. Section 3.6.6 of EPRI report 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 seismic-induced failures led directly to core damage and large early release and that the 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 the other elements of the 10 CFR 50.69 categorization program, the approach will 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 SSC being categorized as HSS.

Evaluation of the Implementation of Conclusions from the Case Studies

The licensee stated that the proposed categorization approach for seismic hazards will include qualitative consideration of the mitigation capabilities of SSCs during seismically-induced events and seismic failure modes, based on insights obtained from prior seismic evaluations performed for Quad Cities. The licensee explained that the qualitative characterization of seismic risk performed for the independent decision-making panel will 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 Cliff's amendment is applicable to this licensee's proposed alternative seismic approach and that the plant-specific evaluation on the implementation of the alternative seismic approach is acceptable. There are no differences that exist between the Quad Cities proposed alternative approach and the approach used in the NRC staff approved Calvert Cliffs 10 CFR 50.69 safety evaluation. The NRC staff's review of the proposed alternative seismic approach, in conjunction with the requirements in 10 CFR 50.69 and the corresponding statement of consideration, finds that the proposed alternative seismic approach includes the evaluations required by 10 CFR 50.69(c)(1)(ii), as well as 10 CFR 50.69(c)(1)(iv) because:

1. The proposed alternative seismic approach includes qualitative consideration of seismic events at several steps of the categorization process, including documentation of the

information for presentation to the integrated decision-making panel (IDP) as part of the integrated, systematic process for categorization.

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

3. The insights presented to the IDP include potentially important seismically-induced failure modes, as well as mitigation capabilities of SSCs during seismically-induced design basis and severe accident events consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI 3002017583. The insights will use prior plant-specific seismic evaluations and, therefore, in conjunction with performance monitoring for the proposed alternative seismic approach, reasonably reflect the current plant configuration. Further, the recommendation for categorizing civil structures in the alternative seismic approach provides appropriate consideration of such failures from a seismic event.

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

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

Consideration of Changes to Seismic Hazard

The possibility exists for the seismic hazard at the site to increase such that the criteria for use of the proposed alternative seismic approach may no longer be appropriate. The licensee stated that the continued comparison of GMRS to SSE applies to the Quad Cities site. The licensee also stated that the seismic hazard at the plant is subject to periodic reconsideration as new information became 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 the same as that approved in the Calvert Cliffs amendment. Consequently, the NRC staff finds that the consideration of changes to the seismic hazard at Quad Cities 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 is clear and traceable, (2) the proposed alternative seismic approach includes periodic reconsideration of the seismic hazard as new information becomes available, (3) the proposed alternative seismic approach satisfies the requirements in 10 CFR 50.69 discussed above, and (4) the licensee has included a proposed license condition in the LAR to require NRC approval for a change to the specified seismic categorization approach.

Monitoring of Inputs to and Outcome of Proposed Alternative Seismic Approach

In Section 3.5 of the 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 NRC staff finds that:

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

2. The monitoring and configuration control program ensures that potential degradation of the seismic capacity would be detected and addressed before significantly impacting the plant risk profile.

Therefore, the NRC staff finds that the potential impact of the seismic hazard on the categorization is maintained acceptably low and the requirements in 10 CFR 50.69(c)(1)(iv) are met for the proposed alternative seismic approach.

External Floods

The Quad Cities flooding Integrated Assessment (IA) (ML18180A033), submitted to the NRC on June 29, 2018, demonstrated the need for Quad Cities to install fourteen barriers (metal stop logs, door panel installations, and hinged gates with inflatable seals) in the early response to local intense precipitation (LIP) events as they developed. Seven of the barriers are temporary and require manual installation. The remaining seven barriers are permanently installed exterior doors or plates that do not require manual action to perform their function. The NRC staff concluded in its staff assessment of the Quad Cities IA (ML19168196) that the UFSAR and LIP barrier measures appropriately addressed plant vulnerabilities to external flooding. For the combined effects floods (probable maximum flood (PMF), dam failure, and wind-generated waves), a probabilistic flood hazard assessment (PFHA) was performed in 2021. This evaluation characterized the frequency of floods exceeding plant grade and the LIP barriers. The PFHA showed that for the combined effects flood, the frequency of water exceeding the ground floor elevation (plant grade) of 595.0 feet is approximately $2x10^{-6}$/yr. To lower the risk of the combined effects flood hazard, the licensee determined that seven of the LIP barriers could be modified to protect the plant up to the height of 599.0 feet, which equates to a flood hazard exceedance frequency less than $1x10^{-6}$ per year.

The licensee estimated the human error probabilities (HEP) to install each of the LIP barriers to the 599.0 feet level. The licensee provided a revised HEP estimate in its March 19, 2024, supplement with more detailed analyses of the manual actions utilizing the NRC developed Integrated Human Event Analysis System for Event and Condition Assessment (IDHEAS-ECA) methodology. Using this methodology, the licensee estimated that the total HEP for installing all seven barriers is $1.1x10^{-2}$. However, the licensee used a conservative conditional core damage probability (CCDP) of $3x10^{-1}$, which is equal to the HEP of installing all the flood barriers calculated in the original LAR for additional conservatism. The analysis includes the assumption that failure to install the LIP gates and for them to not perform their designed function fails all flood mitigation and leads directly to core damage. The analysis also conservatively ignores all other mitigation capabilities, such as the licensee's approved FLEX strategies and temporary dams. The licensee combined the CCDP with the frequency of flood waters exceeding plant grade and then topping the LIP barriers to estimate the mean CDF. The NRC staff's review finds that: (1) the licensee's updated human reliability analysis is consistent with the NRC developed

IDHEAS-ECA methodology, (2) Quad Cities personnel have adequate time to enter the procedure, diagnose the scenario, and ensure proper installation of the LIP barriers, and (3) not crediting other mitigation capabilities discussed above and using the HEP value of $3x10^{-1}$ in the analysis represent a demonstrably conservative estimate of flooding risk at Quad Cities for this application.

In the March 19, 2024, response to NRC RAIs, the licensee confirmed that the 14 flood barriers in Table A4-1 to keep flood waters from entering QCNPS buildings and impacting safety related equipment are credited for screening per the NEI 00-04, Figure 5-6, guidance, and therefore, these SSCs will be considered HSS if categorized under 10 CFR 50.69. Based on its review, the NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of the flood barriers listed in Table A4-1 consistent with the guidance provided in NEI 00-04, as endorsed by the NRC and is, therefore, acceptable.

<u>Method for Assessing Other External Hazards</u>

This hazard category includes all non-seismic, and non-external flooding such as high winds, aircraft impact, and other external hazards. The licensee discussed its consideration of other external hazards and concluded that all external hazards, except for seismic, were screened from applicability to Quad Cities per a plant-specific evaluation in accordance with GL 88-20 and the criteria in ASME/ANS RA-Sa-2009 PRA Standard.

In the March 19, 2024, response to NRC RAIs, the licensee confirmed that, during categorization of SSCs, Quad Cities will follow the process consistent with the NEI 00-04 guidance in the Figure 5-6 flow chart to determine whether an SSC is credited in the screening of an external hazard (excluding seismic hazards) and should be categorized accordingly. The NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for all other external hazards consistent with the guidance provided in NEI 00-04, as endorsed by the NRC and is, therefore, acceptable.

Regarding the screening of nearby industrial facilities from this application, the licensee provided additional information in its March 19, 2024, Response to NRC RAIs. CF Industries, located 3.1 miles from the site, produces nitrogen fertilizers and agricultural chemicals. For this reason, Quad Cities has ammonia detectors installed on the intake of both trains of the Control Room HVAC system to protect the Control Room Operators if the ammonia concentration reaches a certain level. The response also states that screening criteria C1 (event damage potential is less than events for which plant is designed) applies based on the ammonia detectors' design function to protect the control room operators by isolating the control room heating ventilation and air conditioning system when the ammonia concentration reaches a specific level.

For snow loading on critical structures, the licensee provided, in its March 19, 2024, response to NRC RAIs, details of the amount of snow which would be needed to challenge Quad Cities critical structures compared to the largest snowfall totals ever recorded on a single day and over a two-day period to show a large margin available. The licensee also cited their two severe weather procedures to show that snow would not be allowed to accumulate on roofs of critical buildings. In addition, the licensee's pre-storm guidelines as well as their snow removal plan include all necessary equipment. The licensee cites Criteria C1 as well as Criteria C5 (event develops slowly, allowing adequate time to eliminate or mitigate a threat). Based on its review, the NRC staff concludes that the industrial facilities and snowfall hazards were appropriately screened for this application.

In summary, the NRC staff finds that the use of the Quad Cities IPEEE results described by the licensee in the LAR, the March 19, 2024, response to NRC RAIs, and the licensee's assessment of the other external hazards (e.g., extreme wind and tornado missiles, external flooding) is consistent with Section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. 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 the shutdown safety assessment based on NUMARC 91-06. NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment-primary/secondary. 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 described by the licensee in the submittal is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in 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 to assess shutdown safety 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. 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 enclosure to 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, for passive component categorization, but was approved by the NRC ANO-2. 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, Section XI, Division 1,” July 2002. 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 at Quad Cities.

In section 3.1.2 of the enclosure to the LAR, the licensee stated, "[t]he passive categorization process is intended to apply the same risk-informed process accepted in the AN0 2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS for passive categorization which will result in HSS for its risk-informed safety classification that 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. Therefore, the NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process for Class 1, Class 2, and Class 3, pressure retaining SSCs.

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

Section 3.1.1 of the enclosure to 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, Revision 0. Section 3.2.7 of the LAR further 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, Revision 0, as endorsed by RG 1.201, Revision 1.

In section 3.1.1 of the enclosure to the LAR, for the "Overall Categorization Process," the licensee 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 §50.69(c)(1)(iv)." In Section 3.4 of the LAR, the licensee states, "Sensitivity studies described in NEI 00-04, Section 8, will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF." This sensitivity study together with the periodic review process discussed in section 3.3.2 of this SE, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in Section 8 of NEI 00-04, Revision 0, and therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.3.1.6 Integrated Decision Making

Appendix B of SRP Chapter 19, Section 19.2, provides guidance and the staff 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 staff review of the two steps to ensure the process is 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 will make 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 enclosure to the LAR, the licensee stated, in part, "... 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 in RG 1.201, Revision 1, and therefore, meets the requirements of 10 CFR 50.69.

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

In Section 3.1.1 of the enclosure to the LAR, the licensee states 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. Based on this information, the NRC staff finds that the IDP will comprise the required expertise consistent with Section 9.1 of NEI 00-04, Revision 0.

The guidance in NEI 00-04, Revision 0, as endorsed 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 discusses that at least three members of the IDP will have a minimum of 5 years of experience at the plant, and there will be at least one member of the IDP who has a minimum of 3 years of experience in modeling and updating of the plant-specific PRA. The licensee further states 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, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and 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 the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0, as endorsed 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 the licensee ensures, 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 Quad Cities categorization process, is consistent with the endorsed guidance in NEI 00-04, Revision 0, and therefore, fulfills 10 CFR 50.69(c)(1)(iv).

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

3.3.2 Key Principle 5: Monitor the Impact of the Proposed 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 overtime are continually incorporated.

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

Sections 11 and 12 of NEI 00-04, Revision 0, includes 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 Quad Cities as-built, as-operated plant. A more detailed NRC staff review is provided as follows:

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

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In Section 3.2.6 of the enclosure to 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 enclosure to 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 for facilities licensed under 10 CFR part 50, appendix B, Criterion VI, for Document Control, procedures are considered formal plant documents that require 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 NRC staff finds that the elements provided in Section 3.1.1 of the enclosure to the LAR for the Quad Cities 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, are sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control and records.

Periodic Review (NEI 00-04, Section 12)

Section 50.69(e), “Feedback and Process Adjustment,” 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 of the enclosure to the LAR, the licensee described the process for maintaining and updating the Quad Cities PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, the licensee confirmed that the Quad Cities risk management process ensures that the applicable PRA mode(s) 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.

Section 12.1 of NEI 00-04, Revision 0, states, in part, “Scheduled 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 enclosure to the LAR, the licensee states, in part, “Scheduled periodic reviews at least once every other refueling outage will evaluate new insights resulting from available risk information.” Therefore, the NRC staff finds the risk management process described by the licensee in the LAR is consistent with section 12 of NEI 00-04, Revision 0, guidance as endorsed by the NRC, and therefore, satisfies the requirements of 10 CFR 50.69(e)(1). Furthermore, based on the above, the staff has determined that the proposed change satisfies the fifth key principle for risk-informed decision making described in RG 1.174.

3.4 Changes to the Operating Licenses

The NRC staff’s finding on the acceptability of the PRA evaluation in the licensee’s proposed 10 CFR 50.69 process is conditioned upon the License Condition provided in section 1.0 of this SE.

The NRC staff finds that the proposed license condition is acceptable because it adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations on, the Illinois State official was notified of the proposed issuance of the amendment on June 6, 2024. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility’s components located within the restricted area as defined in 10 CFR, part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (88 FR 53537; dated August 8, 2023). 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.

6.0 CONCLUSION

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

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Date of Issuance: July 3, 2024

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 301 AND 297 RE: ADOPTION OF 10 CFR 50.69 "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2023-LLA-0085) DATED JULY 3, 2024

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