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

May 27, 2021

Mr. David P. Rhoades
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 249 AND 235 RELATED TO APPLICATION TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2020-LLA-0017)

Dear Mr. Rhoades:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 249 and 235 to Renewed Facility Operating License Nos. NPF-11 and NPF-18, for the LaSalle County Station, Unit Nos. 1 and 2, respectively. The amendment consists of changes to the Renewed Facility Operating Licenses in response to your application dated January 31, 2020 (Agencywide Documents Access Management System (ADAMS) Accession No. ML20031E699), as supplemented by letters dated October 1, October 16, October 29, 2020, and January 22, 2021 (ADAMS Accession Nos. ML20275A292, ML20290A791, ML20303A307, and ML21022A130, respectively).

The amendment modified the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors."

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

Sincerely,

/RA/

Bhalchandra K. Vaidya, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-373 and 50-374

Enclosure:
1. Amendment No. 249 to NPF-11
2. Amendment No. 235 to NPF-18
3. Safety Evaluation

cc: Listserv

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 249
Renewed License No. NPF-11

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated January 31, 2020, as supplemented by letters dated October 1, October 16, and October 29, 2020, and January 22, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes as indicated in the attachment to this license amendment, and new Paragraph 2.(C)(47) of Renewed Facility Operating License No. NPF-11 will read as follows:

(47) <u>Adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants</u>"

Exelon Generation Company, LLC (EGC) 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 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 the EGC submittal letter dated January 31, 2020, and all its subsequent associated supplements, as specified in License Amendment No. 249 dated May 27, 2021.

EGC will complete the implementation items listed in Table APLA-01.2 in Attachment 1 of EGC letter to NRC dated October 29, 2020, prior to implementation of 10 CFR 50.69 program. All issues identified will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA Standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.05.27 15:41:42 -04'00'

Nancy L. Salgado, 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: May 27, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 249

LASALLE COUNTY STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-373

Renewed Facility Operating License No. NPR-11

Replace the following pages of the Renewed Facility Operating License No. NPR-11 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating area of change

REMOVE	INSERT
10	10
11	11
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Enclosure 2

Am. 249
05/27/21

(47) Adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants"

Exelon Generation Company, LLC (EGC) 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 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 the EGC submittal letter dated January 31, 2020, and all its subsequent associated supplements, as specified in License Amendment No. 249 dated May 27, 2021.

EGC will complete the implementation items listed in Table APLA-01.2 in Attachment 1 of EGC letter to NRC dated October 29, 2020, prior to implementation of 10 CFR 50.69 program. All issues identified will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA Standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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).

Am. 102 03/16/95

D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J and 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement No. 1, No. 2, No. 3 to the Safety Evaluation Report.

(b) DELETED

(c) DELETED

(d) DELETED

Am. 226 11/16/17

(e) DELETED

Am. 112 04/05/96

(f) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (f) is described in the safety evaluation accompanying Amendment No. 112 to this License.

These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted there from), and the provisions of the Act.

E. This renewed license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum dated November 1978, and the Final Supplemental Environmental Impact Statement dated August 2016, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

Am. 178 06/14/06

F. Deleted

G. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This renewed license is effective as of the date of issuance and shall expire April 17, 2042.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

WILLIAM M. DEAN, DIRECTOR
OFFICE OF NUCLEAR REACTOR REGULATION

Am. 194
08/28/09

Attachments:
1. DELETED
2. Appendix A – Technical Specifications (NUREG-0861)
3. Appendix B – Environmental Protection Plan

Date of Issuance: October 19, 2016

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 235
Renewed License No. NPF-18

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

   A. The application for amendment by Exelon Generation Company, LLC (the licensee) dated January 31, 2020, as supplemented by letters dated October 1, October 16, and October 29, 2020, and January 22, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes as indicated in the attachment to this license amendment, and new Paragraph 2.(C)(36) of Renewed Facility Operating License No. NPF-18 will read as follows:

    (36) Adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants"

    Exelon Generation Company, LLC (EGC) 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 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 the EGC submittal letter dated January 31, 2020, and all its subsequent associated supplements, as specified in License Amendment No. 235 dated May 27, 2021.

    EGC will complete the implementation items listed in Table APLA-01.2 in Attachment 1 of EGC letter to NRC dated October 29, 2020, prior to implementation of 10 CFR 50.69 program. All issues identified will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA Standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

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

FOR THE NUCLEAR REGULATORY COMMISSION

Nancy L. Salgado

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Date: 2021.05.27 15:43:04 -04'00'

Nancy L. Salgado, 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: May 27, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 235

LASALLE COUNTY STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-374

Renewed Facility Operating License No. NPR-18

Replace the following pages of the Renewed Facility Operating License No. NPR-18 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating area of change

REMOVE	INSERT
10	10
11	11
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3. The licensee shall notify the NRC in writing within 30 days after having accomplished item (b)1 above and include the status of those activities that have been or remain to be completed in item (b)2 above.

Am. 235
05/27/21

(36) Adoption of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants"

Exelon Generation Company, LLC (EGC) 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 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 the EGC submittal letter dated January 31, 2020, and all its subsequent associated supplements, as specified in License Amendment No. 235 dated May 27, 2021.

EGC will complete the implementation items listed in Table APLA-01.2 in Attachment 1 of EGC letter to NRC dated October 29, 2020, prior to implementation of 10 CFR 50.69 program. All issues identified will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA Standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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).

Am. 87 03/16/95

D. The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include:

(a) Exemptions from certain requirements of Appendices G, H and J to 10 CFR Part 50, and to 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement Numbers 1, 2, 3, and 5 to the Safety Evaluation Report.

Am. 181 08/28/09

(b) DELETED

Am. 212 11/16/17

(c) DELETED

Am. 181 08/28/09

(d) DELETED

Am. 97 04/05/96

(e) An exemption was granted to remove the Main Steam Isolation Valves (MSIVs) from the acceptance criteria for the combined local leak rate test (Type B and C), as defined in the regulations of 10 CFR Part 50, Appendix J, Option B, Paragraph III.B. Exemption (e) is described in the safety evaluation accompanying Amendment No. 97 to this License.

These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

E. Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum dated November 1978, and the Final Supplemental Environmental Impact Statement dated September 2016, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

Am. 164 06/14/06 F. Deleted

Am. 164 06/14/06 G. Deleted

H. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

I. This renewed license is effective as of the date of issuance and shall expire at Midnight on December 16, 2043.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

WILLIAM M. DEAN, DIRECTOR
OFFICE OF NUCLEAR REACTOR REGULATION

Attachments:
1. DELETED
2. DELETED
3. Appendix A – Technical Specifications (NUREG-1013)
4. Appendix B – Environmental Protection Plan

Date of Issuance: October 19, 2016

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 249 AND 235 TO RENEWED

FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By application dated January 31, 2020 (Reference 1), as supplemented by letters dated October 1, 2020 (Reference 2), October 16, 2020 (Reference 3),October 29, 2020 (Reference 4), and January 22, 2021 (Reference 5), Exelon Generation Company (EGC or licensee) submitted a license amendment request (LAR) for changes to the Renewed Facility Operating Licenses (RFOLs) for LaSalle County Station, Unit Nos. 1 and 2 (LaSalle).

Specifically, the proposed amendments would modify the licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to the special treatment controls identified in 10 CFR 50.69(b) (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows improved focus on equipment that has higher safety significance resulting in improved plant safety.

The supplemental letters dated October 1, October 16, and October 29, 2020, and January 22, 2021, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 7, 2020 (85 FR 19511).

2.0 REGULATORY EVALUATION

2.1 Applicable Regulations

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

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process; adjusts treatment requirements consistent with the relative significance of the SSC; and manages the SSC categorization 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 Risk-Informed Safety Class (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 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 their functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve their focus on equipment that has been categorized using the requirements in 10 CFR 50.69 as HSS.

2.2 Regulatory Guides

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

- Regulatory Guide (RG) 1.201, Revision 1, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance,” May 2006 (Reference 6).

- RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” May 2009 (Reference 7).

- RG 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” January 2018 (Reference 8).

- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” March 2017 (Reference 9).

- NUREG-0800, “Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Chapter 19, Section 19.2, “Review of Risk

Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (Reference 10).

The licensee's submittal cites Revision 2 of RG 1.200; the RG has been updated since the time of the submittal. The updates do not include any technical discrepancies that would deviate or impact the staff's review for consistency with the endorsement of RG 1.201, therefore the NRC staff finds RG 1.200, Revision 2 and RG 1.200, Revision 3, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," December 2020 (Reference 11), acceptable for the implementation of the SSC categorization program.

NRC-Endorsed Guidance

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 12), as endorsed by RG 1.201 for trial use with clarifications, which describes a process that the NRC staff considers acceptable for complying with 10 CFR 50.69. This process determines the safety significance of SSCs and categorizes them into one of four RISC categories defined in 10 CFR 50.69.

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

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

Additionally, Section 11 of NEI 00-04 provides guidance on program documentation and change control related to the requirements of 10 CFR 50.69(f). Section 12 of NEI 00-04 provides guidance on the periodic review related to the requirements in 10 CFR 50.69(e). Maintaining change control and periodic review provides a reasonable level of confidence that all aspects of the program 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 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 defense-in-depth, and (3) evaluation of safety margins.

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

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

The SSCs 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.

Section 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).

[1] NEI 00-04, Revision 0, uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC staff 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.

As stated in 10 CFR 50.69(b), after the NRC approves an application for a license amendment, a licensee may voluntarily comply with 10 CFR 50.69, as an alternative to compliance with the following requirements for LSS SSCs:

(i) 10 CFR Part 21
(ii) a portion of 10 CFR 50.46a(b)
(iii) 10 CFR 50.49
(iv) 10 CFR 50.55(e)
(v) certain requirements of 10 CFR 50.55a
(vi) 10 CFR 50.65, except for paragraph (a)(4)
(vii) 10 CFR 50.72
(viii) 10 CFR 50.73
(ix) Appendix B, "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 review, as documented in this safety evaluation (SE), used the framework provided in RG 1.174, Revision 3, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1.

Section 2 of NEI 00-04, Revision 0, in part, states 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)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. Integrated Decisionmaking Panel Review and Approval (Section 9 of NEI 00-04, Revision 0)
8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

In Section 3.1.1 of the LAR, the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. In Section 3.2.3 of the LAR, the licensee proposed the use of an alternative seismic approach to assess the contribution of seismic risk in the licensee's categorization process. In Section 3.1.2 of the LAR, the licensee proposed the use of an alternative approach for passive categorization. The NRC staff notes that use of these alternative methods are a deviation from the NEI 00-04 guidance as endorsed by the NRC in RG 1.201, Revision 1. The NRC staff's evaluation of the licensee's proposed alternative methods is provided in Sections 3.3.1.2 and 3.3.1.2.3 of this SE.

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

The regulatory requirements in 10 CFR 50.69 and 10 CFR Part 50, Appendix B, and the monitoring outlined in NEI 00-04, Revision 0, and clarifications in RG 1.201, Revision 1, ensure that the SSC categorization process is sufficient to 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 prescribed in NEI 00-04, Revision 0 to assure that SSCs specified are appropriately categorized consistent with 10 CFR 50.69. In light of the above, the NRC staff concludes that the proposed 10 CFR 50.69 program meets the first key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3.

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

In RG 1.174, Revision 3, the NRC identified the following considerations used for evaluating the impact of a LB change on defense-in-depth:

- 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 (“Defense-In-Depth Assessment”) 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.

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

3.2.3 Key Principle 3: LB Change Maintains Sufficient Safety Margins

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

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

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

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

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

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

In Sections 3.1.1, 3.2.1, and 3.2.2, of the LAR, the licensee explained that the LaSalle categorization process uses PRA-modeled hazards to assess risks for the internal events (includes internal flood) and internal fires. The licensee's categorization process uses an alternative seismic approach for consideration of seismic risk in the categorization of SSCs. For the other risk contributors, the licensee's process uses non-PRA methods to characterize the risk provided in Section 3.3.1.2 of this SE.

The approaches and methods proposed by the licensee to address internal events, internal fires, external events (except for seismic), other hazards, defense in depth, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0. The non-PRA method for the categorization of passive components is consistent with the Arkansas Nuclear One, Unit 2 (ANO-2) methodology for passive components approved on April 22, 2009 (Reference 13), for risk-informed safety classification and treatment for repair/replacement activities in class 2 and 3 moderate- and high-energy systems. The NRC staff's evaluation of the proposed use of the ANO-2 methodology in the SSC categorization process is provided in Section 3.3.1.2.3 of this SE. To address the seismic hazard, the licensee proposed to use an alternative method not specified in NEI 00-04 guidance as endorsed by the NRC for ANO-2. A detailed staff review of the licensee's proposed approach for the use of the alternate seismic method is provided in Section 3.3.1.2.1 of this SE.

3.3.1.1 Scope of the PRA

The LaSalle PRA is comprised of a full power, internal events PRA (IEPRA) and fire PRA (FPRA) that evaluates the CDF and LERF risk metrics. The licensee stated in Section 3.3 of the LAR that the IEPRA (includes internal floods) and FPRA models had been assessed against RG 1.200, Revision 2, consistent with NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007 (Reference 14).

The NRC staff finds the information provided in the LAR, as supplemented, is sufficient to support its review of the IEPRA (includes internal floods) and FPRA for technical acceptability and, therefore, the LAR meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

The NRC staff evaluated the scope of the PRA including: (1) peer-review history and results, (2) the Appendix X, Independent Assessment process (Reference 15), (3) credit for FLEX in the PRA, and (4) assessment of assumptions and approximations. In a letter to the licensee dated September 29, 2020 (Reference 16), the NRC staff issued requests for additional information (RAIs) to further assess the acceptability of the LaSalle IEPRA and FPRA for consistency with RG 1.200, Revision 2, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. The NRC staff's review of these aspects of the IEPRA and FPRA and the licensee's supplemental responses to assess for consistency with the applicable guidance, as endorsed by the NRC, are provided below.

Internal Event PRA (Includes internal floods) Peer Review History

In Section 3.3 of the LAR, the licensee confirmed that the LaSalle IEPRA model (includes flooding) received a peer review in April 2008 using NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," November 2008 (Reference 17), the American Society of Mechanical Engineers (ASME) PRA Standard ASME RA-Sc-2007, "Addenda to ASME RA-S-2002: Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 18), and RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007 (Reference 19). The licensee confirmed that the facts and observations (F&Os) identified from the 2008 IEPRA (includes internal floods) peer review were addressed as a part of the PRA maintenance process during the periodic PRA updates. Furthermore, consistent with RIS 2007-06, a gap analysis was performed as a part of the 2014 PRA update to assess the IEPRA (includes internal floods) using ASME/ANS (American Nuclear Society) PRA Standard RA-Sa-2009 (Reference 20). Two subsequent independent assessments for closure of the F&Os using Appendix X to NEI 05-04, 07-12, and 12-13, as accepted, with conditions by the NRC staff were performed in 2017 and 2019 for the IEPRA (includes internal floods). The NRC staff performed a review of the remaining open F&Os and self-assessment open items, along with the dispositions provided by the licensee in Attachment 3 of the LAR, and concluded that the dispositions were sufficient for this application because the F&O was either resolved by subsequent actions or the resolution of the F&O would not adversely impact the SSC categorization program. The NRC staff finds that the LaSalle IEPRA (includes internal floods) was appropriately peer reviewed consistent with RG 1.200, Revision 2, F&Os closed consistent with Appendix X guidance, as accepted, with conditions by the NRC staff, and the remaining open F&Os have been appropriately assessed for impact on the SSC categorization program.

Internal FPRA Peer Review History

In Section 3.3 of the LAR, the licensee confirmed that the LaSalle FPRA model received a peer review in December 2015 using NEI 07-12, Revision 0, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Draft Version H, November 2008 (Reference 21), the ASME PRA Standard ASME/ANS RA-Sa-2009, and RG 1.200, Revision 2. The licensee further confirmed that the F&Os identified from the 2015 FPRA peer review were addressed as a part of the PRA maintenance process during the subsequent PRA updates, and an Independent Assessment was performed in October 2017 consistent with Appendix X to NEI 05-04, 07-12, and 12-13, as accepted, with conditions by the NRC staff. A focused-scope peer review was performed in parallel with the Independent Assessment to assess the technical element "Fire Risk Quantification (FQ)" of the PRA Standard due to the large change in CDF and LERF. A subsequent independent assessment was performed in September 2019 for review of the changes made to the LaSalle FPRA. The NRC staff performed a review of the remaining open F&Os and open self-assessment items, along with the dispositions provided by the licensee in Attachment 3 of the LAR. In the Division of Risk Assessment (DRA)/PRA Licensing Branch A (APLA) RAI 01 dated September 29, 2020, the NRC staff noted that for F&Os 4-17 and 6-11, the licensee stated the items will be resolved prior to 10 CFR 50.69 program implementation, however, the LAR does not include implementation items to complete these activities prior to implementation of the SSC categorization program. In its October 29, 2020, response to DRA/APLA RAI 01, the licensee provided Table APLA-01.2 that included implementation items to address F&Os 4-17 and 6-11 and provided an updated proposed license condition in Attachment 3, Section 2.3 to the RAI response letter. The NRC staff reviewed the proposed dispositions for F&Os 4-17 and 6-11 provided in Attachment 3 of the LAR and concluded that the proposed resolutions to address the F&Os are sufficient for the SSC categorization program.

In DRA/APLA RAI 02.a, the NRC staff cited two documents that provide NRC-accepted guidance on methods and studies performed by NRC on fire PRAs that were issued after the full-scope peer review of the LaSalle FPRA in December 2015 (i.e., NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures during Fire (RACHELLE-FIRE)," April 2016 (Reference 22), and NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities," June 2015 (Reference 23). The NRC staff requested the licensee to confirm whether the FPRA guidance has been incorporated into the LaSalle FPRA and to provide justification if the guidance was not applied. The NRC staff also requested an explanation of whether incorporation of the guidance represented a PRA upgrade as defined by the ASME/ANS RA-Sa-2009 PRA standard. In October 29, 2020, response to DRA/APLA RAI 02.a, the licensee explained that the guidance in NUREG-2178, Volume 1 was implemented. The licensee indicated that guidance from a draft version of NUREG-2178, Volume 1 was incorporated into the FPRA that was peer reviewed in December 2015. The licensee further confirmed that the peer review did not identify any F&Os related to guidance from the draft and that upon issuance of NUREG-2178, Volume 1, the LaSalle FPRA model was reviewed and updated, as necessary, to ensure that the correct final heat release rates (HRRs) for electrical cabinets were used in the fire modeling calculations. The NRC staff concludes that the licensee's consideration and incorporation of the guidance in NUREG-2178, Volume 1 is appropriate because the guidance in the version of NUREG-2178, Volume 1 has been implemented, and the adjustment of HRR values does not constitute a PRA upgrade in accordance with the PRA standard. The licensee confirmed that LaSalle does not credit incipient fire detection systems, and therefore, the guidance in NUREG-2180 is not applicable.

In DRA/APLA RAI 02.b dated September 29, 2020, the NRC staff requested the licensee to confirm if minimum joint Human Error Probability (HEP) values less than 1E-05 were assumed in the fire PRA, and to provide justification of values less than 1E-05 that were used. NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines-Final Report,” July 2012 (Reference 24), discusses the need to consider a minimum value for the joint HEPs. Table 4-4 of Electrical Power Research Institute (EPRI) 1021081, “Establishing Minimum Acceptable Values for Probabilities of Human Failure Events,” October 2010 (Reference 25), provides a lower limiting value of 1E-6 for sequences with a very low level of dependence. In its October 29, 2020, response to DRA/APLA RAI 02.b, the licensee stated that for the LaSalle FPRA dependency analysis a minimum joint HEP of 1E-06 was used unless the timeframe for completing one or more actions in the combination was longer than 15 hours. In these cases, a lower minimum joint HEP of 5E-07 was used. The NRC staff notes that per the guidance in Figure 6-1 of NUREG-1921 when an operator action is performed by a different crew this leads to low dependency even for high stress scenarios. Therefore, according to Table 6-1 of NUREG-1921, the credit taken by the licensee for applying a lower joint HEP for combinations that include a long-term action is consistent with the guidance. The licensee also provided the results of a sensitivity study that was performed in which a minimum joint HEP of 1E-05 was applied and the results compared to the screening criteria specified in Section 5.1 of NEI 00-04. The results of the sensitivity study demonstrated that no additional components met the screening criteria, thus SSC categorization is not adversely impacted. The NRC staff concludes that the licensee’s application of minimum joint HEP values is acceptable because the licensee demonstrated it meets the intent of applicable guidance, and that the minimum joint HEPs used for the LaSalle FPRA have a minimal impact on the SSC categorization results.

In DRA/APLA RAI 02.c dated September 29, 2020, the NRC staff requested additional information to assess how well-sealed cabinets were treated in the FPRA (i.e., fire propagation outside of well-sealed motor control centers (MCCs) cabinets greater than 440 V was evaluated and consistency with the NRC Fire PRA Frequently Asked Question (FAQ) 14-0009, “Treatment of Well Sealed MCC Electrical Panels Greater than 440V,” October 2014 (Reference 26). In its October 29, 2020, response to DRA/APLA RAI 02.c, the licensee confirmed that all MCCs are assumed not to be well-sealed and, therefore, all fires originating from MCC cabinets are assumed to damage external targets. Accordingly, the refinements using the guidance in FAQ 14-0009 were not used.

In DRA/APLA RAI 02.d dated September 29, 2020, the NRC staff requested a discussion of how the risk contribution of fires originating in one unit is addressed for the other unit given possible impacts due to the physical proximity of equipment and cables. The NRC staff also requested that the licensee discuss how the risk contributions of such scenarios impact the application. In its October 29, 2020, response to DRA/APLA RAI 02.d, the licensee explained that fire scenario development methodology used in the FPRA requires that all targets within the zone of influence, regardless of their location in the reactor units be selected and analyzed for fire-induced failure. The licensee further confirmed that all fires, regardless of location, are assumed to result in, at a minimum, a turbine trip in both units, so a fire in the opposite unit will cause a turbine trip in the analyzed unit. The licensee provided the risk contribution of Unit 1 fires to Unit 2 fire CDF and fire LERF, approximately 5 percent and 3 percent, respectively. The NRC staff finds the licensee’s treatment of fire dependencies between the units is acceptable because the risk contribution of fires originating in one unit is addressed for the other unit.

Based on the above conclusions, the NRC staff finds that the LaSalle FPRA was appropriately peer reviewed consistent with RG 1.200, Revision 2, fire methodologies were appropriately considered and implemented, F&Os were closed consistent with Appendix X, as accepted, with conditions by the NRC staff, and the remaining open F&Os were adequately assessed for their impact on the risk-informed application. Therefore, the LaSalle FPRA is acceptable for use in the SSC categorization program.

Identification and Treatment of Key Assumptions and Sources of Uncertainty

In Attachment 6 of the LAR, the licensee confirmed that NUREG-1855, Revision 1, was used 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[2] models were used in the PRA to evaluate identified model uncertainties. The licensee confirmed that for the LaSalle uncertainty analysis some uncertainties and assumptions were screened out 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.

Attachment 6 of the LAR, as supplemented by letter dated October 29, 2020, identified several key sources of uncertainty that could impact the results of the categorization process. The NRC staff reviewed the dispositions of these key sources along with the responses provided in DRA/APLA RAI 05 and finds that the licensee performed appropriate sensitivity studies to assess for impact on the SSC categorization process and demonstrated that the key assumptions and sources of uncertainty would not adversely impact SSC categorization.

Furthermore, in LAR Section 3.2.7, the licensee confirmed that sensitivity studies will be performed consistent with Section 5 of the NEI 00-04 guidance for the IEPRA and FPRA. In accordance with Section 9 of NEI 00-04, as endorsed by RG 1.201, Revision 1, the licensee's integrated decision-making panel (IDP) will use information and risk insights compiled in the initial categorization process, including awareness of the limitations and assumptions of the PRA, and will combine that with other information from design bases, defense in depth, and safety margins to finalize the categorization of the SSCs. As a result, the NRC staff finds that the licensee will perform a sensitivity study consistent with Section 5 of the NEI 00-04 guidance and the IDP will appropriately consider PRA assumptions and simplifications during the SSC categorization process to address the identified key assumptions and sources of uncertainty in the context of the decision-making under consideration.

In addition, the NRC staff recognizes that the licensee will perform routine PRA changes and updates to assure the PRA continually reflects the as-built, as-operated plant, in addition to changes made to the PRA to support the context of the analysis being performed (i.e., sensitivities). Maintenance and updating of the PRA is also assured because sections 50.69(e) and (f) of 10 CFR stipulate the process for feedback and adjustment to assure configuration control is maintained for these routine changes and updates to the PRA(s).

[2] 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.

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. The PRAs must be acceptable to support the categorization process and must be subjected to a peer-review process assessed against a standard that is endorsed by the NRC. Revision 2 of RG 1.200 provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 PRA Standard using a peer-review process.

The licensee has subjected the IEPRA and FPRA models 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 models 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 models are acceptable and meet the requirements set forth in sections 50.69(c)(1)(i) and (ii).

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

The licensee’s categorization process uses an alternative seismic approach for consideration of seismic risk in the categorization of SSCs and the following non-PRA method(s), respectively:

- Screening analysis performed for non-seismic external hazards (e.g., high winds, external flood);

- Screening analysis performed for other hazards;

- Safe Shutdown Risk Management program consistent with NUMARC 91-06, “Guidelines for Industry Actions to Assessment Shutdown Management,” December 1991 (Reference 27); and

- Passive Components: ANO-2 passive categorization.

The NRC staff’s review of the methods determined to be a deviation from the endorsed methodologies in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1, is discussed below.

3.3.1.2.1 Seismic Approach

Description of Proposed Alternative Seismic Approach

The licensee’s proposed alternative seismic approach is discussed in Section 3.2.3, “Seismic Hazards,” of the enclosure to the LAR, and in the supplements dated October 1, 2020, October 16, 2020, and January 22, 2021. The proposed alternative seismic approach as well as the supporting information are summarized in this section of the SE.

The licensee's alternative seismic approach has two important bases; the assertion that the seismic risk is moderate based on the seismic hazard at the plant and the conclusions from the case studies in EPRI report 3002012988, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization" July 2018 (Reference 28).

In its submittal dated January 31, 2020, the licensee stated that EPRI report 3002012988 applied to LaSalle in its entirety except for the sections 2.2 and 2.4 of the EPRI report. The licensee's alternative seismic approach relied on conclusions from case studies performed in EPRI report 3002012988. The case studies compared HSS SSCs identified from seismic probabilistic risk assessments (SPRAs) against HSS SSCs identified from the IEPRA and, in certain cases, FPRAs for four different U.S. nuclear plants. The case studies were used to identify conclusions on the determination of safety significance of SSCs uniquely from SPRAs (i.e., the SSCs identified as HSS that were not identified as HSS in the IEPRA and, as applicable, FPRAs).

The licensee compared the re-evaluated seismic hazard for LaSalle, developed in response to Near-Term Task Force (NTTF) recommendation 2.1 dated March 31, 2014 (Reference 29), against the site's seismic design basis safe shutdown earthquake (SSE) to demonstrate that the following criterion from the EPRI report is met:

Plants where the GMRS [Ground Motion Response Spectrum] to SSE [Safe Shutdown Earthquake] comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3.

The licensee stated that the above-mentioned criterion was met for LaSalle which demonstrated the moderate seismic hazard at its site and therefore, justified the use of the alternative seismic approach. EPRI report 3002012988 refers to plants that meet the above-mentioned criterion as "Tier 2" plants.

To capture the potential impact of seismic risk in the categorization process the licensee's alternative seismic approach includes both quantitative and qualitative assessments of plant SSC-specific seismic insights and their presentation to the IDP as part of its decision-making. The proposed approach includes focused walkdowns and quantification of PRA importance measures, based on a sensitivity study, for selected SSCs using the licensee's IEPRA. The proposed approach also includes consideration of seismic risk through insights from plant-specific seismic information.

Summary of Case Studies in EPRI Report 3002012988

EPRI report 3002012988 includes the results from case studies performed to determine the extent and type of unique HSS SSCs from SPRAs. The case studies were performed for four plants, designated Plants A through D in EPRI report 3002012988. Information regarding the case study plants and the PRAs used for each plant as part of the corresponding case study is provided in Table 1. The case study plants will be referred to by their designators A through D in the remainder of this SE.

Each case study consisted of comparison of the HSS SSCs categorized using the SPRA for that plant against HSS SSCs categorized using the corresponding IEPRA and, as applicable, FPRA.

The comparison used mapping to assign SSCs and SSC failure modes from the SPRA to the IEPRA and/or FPRA. The purpose of the mapping was to identify SSCs and SSC failure modes from the SPRA that were not captured by the IEPRA and, as applicable, FPRA.

Table 1.
Summary of Case Study Plants in EPRI Report 3002012988

Case Study Plant	Nuclear Steam Supply System and Containment Type	PRAs Exercised for Case Study
A (Peach Bottom Atomic Power Station, Units 1 and 2)	BWR/4, Mark I	SPRA, IEPRA, and FPRA
B (North Anna Power Station, Units 1 and 2)	Westinghouse 3-loop, Large Dry Subatmospheric	SPRA and IEPRA
C (Vogtle Electric Generating Station, Units 1 and 2)	Westinghouse 4-loop, Large Dry Atmospheric	SPRA, IEPRA, and FPRA
D (Watts Bar Nuclear Plant, Units 1 and 2)	Westinghouse 4-loop, Ice Condenser	SPRA and IEPRA

EPRI report 3002012988 concluded that the case studies revealed that the only SSCs identified as HSS in the SPRA that were not also HSS from IEPRA and/or FPRA were from unique seismically induced failure modes.

The NRC staff's review of the licensee's proposed alternative seismic approach focused on: (1) whether it met the requirements for categorization approaches set forth in 10 CFR 50.69(b) and (c) as clarified by the Statement of Consideration (SOC) for 10 CFR 50.69 (69 FR 68047); (2) the technical acceptability of the PRAs (i.e., SPRA as well as IEPRA and, as applicable, FPRA) used for the case studies in EPRI report 3002012988 that support the proposed alternative seismic approach; (3) the acceptability of the conclusions from the case studies in EPRI report 3002012988; (4) the acceptability of the implementation of the conclusions from the case studies by the licensee; and (5) the performance monitoring that supports the proposed alternative seismic approach to meet the requirements of 10 CFR 50.69(e).

The NRC staff's technical evaluation of the licensee's proposed alternative seismic approach is discussed below. The NRC staff's technical evaluation was informed by a regulatory audit, conducted virtually, of the proposed alternative seismic approach on June 18 and 19, 2020. Discussion items were provided to the licensee as part of the audit plan dated June 11, 2020 (Reference 30). The regulatory audit included a demonstration by the licensee, using an example SSC, of the implementation of the proposed approach.

The licensee refers to different EPRI report numbers in its submittal and supplements. While the submittal refers to EPRI report 3002012988, the supplements refer to that EPRI report as well as EPRI report 3002017583, "Alternative Approaches for Addressing Seismic Risk in 10 CFR 50.69 Risk-Informed Categorization," February 2020 (Reference 31). According to the licensee, EPRI report 3002017583 is a technical update to EPRI report 3002012988. The NRC staff determined that EPRI report 3002012988 combined with the information in the licensee's submittal and supplements supported the NRC staff review and decisions. Therefore, the NRC staff did not review or use EPRI report 3002017583 for its decisions. All references to "the EPRI report" in the remainder of this SE refer to EPRI report 3002012988 unless specified otherwise.

The NRC staff notes that Figure 2-2 of EPRI report 3002012988 replicates Figure 2-1 and appears to be a typographical error. Similarly, Section 2.3.1 of EPRI report 3002012988 does not include a Step 9, which appears to be a typographical error. However, these errors did not impact the NRC staff review of LaSalle's proposed alternative seismic approach.

Evaluation of the Information Provided for the Proposed Alternative Seismic Approach

The regulation in 10 CFR 50.69(b)(2)(ii) requires the following information:

> A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific PRA, margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.

The SOC accompanying the publication of 10 CFR 50.69 states that if a licensee wishes to use an approach different from those in NEI 00-04, as endorsed by RG 1.201, Revision 1, the submittal must provide a sufficient description of how the categorization would be conducted. The SOC further states that, as part of the submittal, licensees or applicants must also describe what measures they have used for the methods other than a PRA to determine their adequacy for this application.

The regulation in 10 CFR 50.69(b)(2)(iv) requires the following information:

> A description of, and basis for acceptability of, the evaluations to be conducted to satisfy [section] 50.69(c)(1)(iv).

The SOC on section 50.69(b)(2)(iv) states that the licensee is required to include information about the evaluations it intends to conduct to provide reasonable confidence that the potential increase in risk would be small. The SOC further clarifies that a licensee must provide sufficient information to the NRC, describing the risk sensitivity study and other evaluations and the basis for their acceptability as appropriately representing the potential increase in risk from implementation of the requirements in the rule.

In Section 3.2.3 of the enclosure to the LAR and supplements dated October 1, 2020, October 16, 2020, and January 22, 2021, the licensee provided a description of its proposed alternative seismic approach for considering seismic risk in the categorization process and how the proposed alternative seismic approach would be used in the categorization process. In addition, the licensee based the acceptability of its proposed alternative seismic approach on the conclusions gained from case studies performed in EPRI report 3002012988 and, therefore, indirectly, on the acceptability of the PRAs used for the case studies. The information presented in the submittal and the supplements as well as that in EPRI report 3002012988, taken together, provides sufficient details for LaSalle's proposed alternative seismic approach, how the licensee's proposed alternative seismic approach would be used in the categorization process, and the measures for assuring that the quality and level of detail for the licensee's proposed alternative seismic approach are adequate for the categorization of SSCs. Therefore, the NRC staff finds that the requirements in 10 CFR 50.69 (b)(2)(ii) are met for LaSalle's proposed alternative seismic approach.

Section 3.4.3 of the enclosure to the LAR and supplements dated October 1, and October 16, 2020, and January 22, 2021, provided information on how the proposed alternative seismic approach meets the requirements of section 50.69(c)(1)(iv). The information presented in the submittal and the supplements provides sufficient description of, and basis for acceptability of, the evaluations to be conducted to satisfy 10 CFR 50.69(c)(1)(iv) for LaSalle's alternative seismic approach. Therefore, the NRC staff finds that the requirements in 10 CFR (b)(2)(iv) are met for LaSalle's proposed alternative seismic approach.

As discussed above, 10 CFR 50.69(c)(1)(iv) requires evaluations to be performed to provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small. The NRC staff's review of the information submitted by the licensee against the requirements in 10 CFR 50.69(c)(1)(iv) is discussed in the remainder of this SE.

<u>Evaluation of Technical Acceptability of Case Study PRAs</u>

The NRC staff's review of the technical acceptability of the PRAs used for the case studies in EPRI report 3002012988 focused on determining whether major deficiencies existed in the PRAs used for the case studies, and consequently, in the conclusions developed from those case studies. During its review, the NRC staff determined that the Plant B case study was not essential for making the NRC staff regulatory and technical conclusions on LaSalle's proposed alternative seismic approach. Therefore, the Plant B case study as well as the PRAs used for that case study were not reviewed by the NRC staff for this application. and the Plant B case study is not discussed in the remainder of this SE. A summary of the regulatory activities for which the technical acceptability of the SPRA, IEPRA, and FPRA, as applicable, for each case study plant has been previously reviewed by the staff, is provided in Table 2 of this SE.

Table 2.
Summary of PRA Use for Regulatory Actions by Relevant Case Study Plants

Case Study Plant	PRA Exercised for Case Study	Regulatory Activity Supported by PRA
A (Peach Bottom Atomic Power Station, Units 1 and 2)	SPRA	March 2012 10 CFR 50.54(f) Request Arising from NTTF recommendation 2.1
	IEPRA	10 CFR 50.69 LAR
	FPRA	10 CFR 50.69 LAR
C (Vogtle Electric Generating Station, Units 1 and 2) C (Vogtle Electric Generating Station, Units 1 and 2)	SPRA	10 CFR 50.69 LAR; March 2012 10 CFR 50.54(f) Request Arising from NTTF recommendation 2.1
	IEPRA	Risk-Informed Completion Time (RICT) LAR; 10 CFR 50.69 LAR
	FPRA	RICT LAR; 10 CFR 50.69 LAR
D (Watts Bar Nuclear Plant, Units 1 and 2)	SPRA	10 CFR 50.69 LAR; Technical Specifications Task Force (TSTF)-425 LAR; March 2012 10 CFR 50.54(f) Request Arising from NTTF recommendation 2.1
	IEPRA	10 CFR 50.69 LAR; TSTF-425 LAR

In Section 3.2.3 of the enclosure to the LAR, the licensee incorporated by reference information related to Plant A, Plant C, and Plant D (Reference 32), (Reference 33), (Reference 34), (Reference 35), and (Reference 36). In its submittal and supplement dated October 1, 2020, the licensee incorporated by reference the responses by Calvert Cliffs Nuclear Power Plant to the NRC staff's RAIs related to the technical acceptability of the PRAs used in the case studies. The NRC staff reviewed and evaluated the technical acceptability of the PRAs used in the case studies for plants A, C, and D, in EPRI report 3002012988 and its applicability to LaSalle's proposed alternative seismic approach. The NRC staff also evaluated the peer review process and resolution of peer-review findings, and key assumptions and sources of uncertainty for Plants A, C, and D. Based on the its review, the NRC staff finds that PRAs used for the Plant A, C, and D, case studies in EPRI report 3002012988 supporting LaSalle's proposed alternative seismic approach are the same as those previously reviewed by the NRC staff (Calvert Cliffs dated February 28, 2020) (Reference 37). Therefore, the NRC staff's previous findings on the technical acceptability of PRAs used for the plant A, C, and D, case studies in EPRI report 3002012988 are applicable to this licensee's proposed alternative seismic approach. Consequently, the NRC staff finds that the plants A, C, and D, PRAs used for the corresponding case studies in EPRI report 3002012988, as listed in Table 2 above, are technically acceptable for use in supporting LaSalle's proposed alternative seismic approach.

Evaluation of Changes to the Case Study PRAs

Mapping of HSS SSCs between SPRA and IEPRA as well as FPRA was an important aspect of the categorization conclusions from the plants A, C, and D case studies in EPRI report 3002012988. The mapping performed for plants A, C, and D, is discussed in Sections 3.2, 3.4, and 3.5 of EPRI report 3002012988. The mapping was performed to assign SSCs and SSC failure modes from the SPRA to representative SSCs and SSC failure modes in the IEPRA and/or FPRA. The purpose of the mapping was to identify SSCs and SSC failure modes from the SPRA that were not captured in the IEPRA and/or, as applicable, FPRA.

The case studies in the EPRI report supporting LaSalle's proposed alternative seismic approach as well as the PRAs used for those case studies are identical to those reviewed by the NRC staff for Calvert Cliffs LAR to adopt 10 CFR 50.69 dated November 28, 2018 (Reference 38). As part of its review of the Calvert Cliffs LAR, the NRC staff reviewed the approach, implementation and results of the mapping performed for the plants A, C, and D case studies to ensure that the mapping was technically justified and, therefore, could be used to develop the related categorization conclusions. The NRC staff's review included the details in Section 3.2.5 and the results in Table 3-5 of EPRI report 3002012988 for Plant A, the details in Section 3.4.5 and the results in Table 3-9 of EPRI report 3002012988 for Plant C, and the details in Section 3.5.5 and the results in Table 3-11 of EPRI report 3002012988 for Plant D.

In its supplement dated October 1, 2020, the licensee incorporated by reference the responses by Calvert Cliffs to the NRC staff RAIs related to the mapping performed as part of the case studies in the EPRI report. The NRC staff reviewed and evaluated the mapping of SSCs between the SPRA, IEPRA, and, as applicable, FPRA, for the plants A, C, and D case studies in EPRI report 3002012988 and its applicability to LaSalle's proposed alternative seismic approach. Based on its review, the NRC staff finds that the mapping performed for the plants A, C, and D case studies in EPRI report 3002012988 to support LaSalle's proposed alternative seismic approach is the same as that previously reviewed by the NRC for Calvert Cliffs. Therefore, the NRC staff's previous findings on the mapping performed for the plants A, C, and D case studies in EPRI report 3002012988 are applicable to this licensee's proposed alternative seismic approach. Consequently, the NRC staff finds that the mapping of SSCs between the

SPRA, IEPRA, and, as applicable, FPRA, for the plants A, C, and D case studies was performed in a technically justifiable manner because: (1) the mapping of explicitly modeled components was performed by identifying representative and/or logically related SSCs, including so-called ‘super components’ among the PRAs, (2) the mapping of passive and implicitly modeled components was performed by identifying appropriate functions for such components, and (3) the building and containment penetration failures were included in the categorization conclusions derived from the case studies in EPRI report 3002012988 and, therefore, in LaSalle’s proposed alternative seismic approach.

Evaluation of the Conclusions from the Case Studies in EPRI Report 3002012988

Section 3.6 of EPRI report 3002012988 discusses the conclusions on the determination of unique HSS SSCs from SPRAs. Those conclusions were identified by the licensee in support of its proposed alternative seismic approach.

The key categorization conclusion from the plants A, C, and D case studies is that the only SSCs identified as HSS in the SPRA that were not also HSS from IEPRA and/or FPRA were from unique seismically induced failure modes. The remainder of HSS SSCs from SPRA are captured by the corresponding IEPRA and/or FPRA or other aspects of the NEI 00-04, Revision 0, categorization process. Additional details about the categorization conclusions is provided in Sections 3.6.2 through 3.6.4 of EPRI report 3002012988.

Section 3.6.2 of EPRI report 3002012988 states that the case studies identified HSS SSCs unique to SPRAs due to the correlated failure mode of those SSCs during a seismic event (e.g., if two pumps performing the same function are located side by side in a plant, they are both assumed to fail with the same conditional probability of failure for a given seismic acceleration). Section 3.6.3 of EPRI report 3002012988 discusses the conclusions from the case studies related to relays and states that relays are explicitly modeled in SPRAs but are usually not included in IEPRA. EPRI report 3002012988 states that relays would be implicitly modeled in the IEPRA and their function within the system would need to be evaluated to perform the 10 CFR 50.69 categorization down to the component level based on the guidance in Section 5 of NEI 00-04, Revision 0, for implicitly modeled components.

The NRC staff reviewed the results of the case studies presented in Sections 3.2.5, 3.4.5, and 3.5.5, of EPRI report 3002012988 for plants A, C, and D, respectively. The NRC staff evaluated unique aspects of SPRAs as part of its review of the conclusions on the determination of unique HSS SSCs from SPRAs to support the proposed alternative seismic approach. Appendix 5-A of the 2009 ASME/ANS PRA Standard, endorsed in RG 1.200, Revision 2, respectively), discusses the important differences between SPRA and IEPRA including:

- Initiating events caused by earthquakes that can be different from those in IEPRA;
- Consideration of the entire seismic hazard curve (i.e., all possible levels of earthquakes along with their frequencies of occurrence and consequences);
- Simultaneous damage to multiple redundant components (also known as correlated failures) which represents a major common-cause effect.

Additional discussion in the 2009 ASME/ANS PRA Standard notes that earthquakes can cause failures that are not explicitly represented in the IEPRA such as damage to structures and other passive items such as large tanks, and anchorage. Other categories of seismically induced failures that are typically not modeled in the IEPRA are seismically induced relay-chatter. In addition, SPRAs need to consider the physical locations and proximity of SSCs because secondary failures such as spatial interactions must be considered. These differences and special considerations for SPRA as compared to IEPRA are consistent with the discussion in NUREG/CR-2300, Volume 2, “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants: Chapters 9–13 and Appendices A-G,” January 1983 (Reference 39).

The NRC staff’s review of the results of the case studies for plants A, C, and D noted that the SSCs that were identified as HSS uniquely from the SPRAs are primarily due to correlated and interaction failure modes of SSCs, failure of passive structures, and the relay-chatter failure mode. The NRC staff finds that SSCs experiencing such failure modes are expected to be identified as HSS from SPRAs because these failure modes are unique to and dominant in SPRAs. The NRC staff further finds that the identification of SSCs as HSS due to such failure modes uniquely from SPRAs is consistent with the understanding of the impact of seismic events on nuclear power plants as well as the development of SPRAs.

The NRC staff’s evaluation of the categorization conclusion related to relays, due to relay-chatter failure mode, determined that subcomponents such as relays that are not directly modeled in other PRAs could be treated as another failure mode for the SSCs to which they are associated. Therefore, the importance of relays would be accounted for in the importance calculation for the corresponding SSC using the NEI 00-04 formulae for the integral assessment. The NRC staff notes that although EPRI report 3002012988 appears to focus on emergency power system relays, the categorization conclusion for relays is applicable to any relay modeled in the SPRA because the fact that the relays are modeled in the SPRA implies that those relays are important for the function of an SSC.

In its supplement dated October 1, 2020, the licensee incorporated by reference the responses by Calvert Cliffs to the NRC staff’s RAIs related to the technical acceptability of the PRAs used in the case studies. These responses included the addition of Section 3.6.6 to EPRI report 3002012988 to discuss categorization of civil structures. Section 3.6.6 to EPRI report 3002012988 states that civil structures containing PRA credited equipment (e.g., reactor building) are likely important to safety because their failure can fail the credited equipment functions, and recommends considering civil structures housing HSS SSCs to be HSS themselves, unless otherwise justified, in the event a licensee chooses to categorize structures under the 10 CFR 50.69 program. The NRC staff finds that the building failures were appropriately considered by recommending the structures housing HSS SSCs to be HSS themselves. The NRC staff notes that it did not review and is not providing any conclusions on the discussion related to the appropriateness of the use of the risk achievement worth (RAW) importance measure for SPRAs in Section 3.6.6, which was provided as an addition to EPRI report 3002012988 as part of the Calvert Cliffs LAR and is incorporated by reference for LaSalle.

Section 3.6.4 of EPRI report 3002012988 discusses the categorization conclusion related to the identification of FLEX equipment from the case studies. The discussion in Section 3.6.4 of EPRI report 3002012988 states that meeting PRA technical acceptability guidance in RG 1.200, Revision 2, ensures that the performance assumed in the PRA for FLEX equipment is consistent with plant practices. The NRC staff disagrees with the EPRI report’s interpretation of the guidance in RG 1.200, Revision 2, because the guidance in RG 1.200, Revision 2, ensures that SSC reliability data used in PRAs reflect the as-built, as-operated plants, and the guidance does not support maintaining the assumed performance of such SSCs, which is required by

10 CFR 50.69(e). Nonetheless, the interpretation in EPRI report 3002012988 on the use of PRA technical acceptability guidance in RG 1.200, Revision 2, for performance monitoring for FLEX equipment does not change the categorization conclusions about unique HSS SSCs from SPRAs, or the licensee's implementation in the proposed alternative seismic approach for LaSalle, and therefore, does not impact the NRC staff's evaluation of LaSalle's LAR.

Based on its review, the NRC staff finds that the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI report 3002012988 from the plants A, C, and D case studies and licensee's submittal as well as supplements are valid because: (1) they were developed following a systematic process that used technically acceptable SPRAs, (2) the HSS SSCs from the SPRA, IEPRA, and, as applicable, FPRA were identified consistent with the guidance in NEI 00-04, (3) mapping of SSCs between the PRAs used in each case study was performed in a technically justifiable manner to identify conclusions, (4) the categorization conclusions were consistent with the SSCs identified as uniquely HSS from SPRAs in the plants A, C, and D case studies, and (5) the categorization conclusions are consistent with the unique aspects and key differences of SPRAs compared to other PRAs.

Evaluation of the Criteria for the Proposed Alternative Seismic Approach

The licensee proposed the following criteria for the applicability and use of the proposed alternative seismic approach:

> Plants where the GMRS [Ground Motion Response Spectrum] to SSE [Safe Shutdown Earthquake] comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3.

"Tier 1" in the EPRI report is defined as:

> Plants where the GMRS peak acceleration is at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz.

"Tier 3" in the EPRI report is defined as:

> Plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 50.54(f) letter.

As noted above, EPRI report 3002012988 refers to plants as "Tier 2" plants where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3.

For the currently operating plants, the SSE was developed to envelope the deterministic hazard at the site. Therefore, from a seismic hazard perspective, the site-specific SSE, derived using a deterministic approach, can be compared to the corresponding GMRS, which is derived using a probabilistic approach. The comparison of the site-specific GMRS and SSE provides information about any seismic risk that would be unaccounted for in the current plant LB.

The seismic fragility of an SSC is based on the site-specific load demand on the SSC during a seismic event and the SSC's capacity to accommodate that load. SSCs have inherent capacities to accommodate seismic loading based on the non-seismic design loads, such as those from pressure, temperature, and dead weight, as well as the required functions for the

SSC, irrespective of the site-specific seismic load. In addition, conservatisms in the design process result in margins above the design basis for such SSCs. Certain features such as equipment anchorage are designed against the site-specific seismic demand and therefore, are more closely associated with the site-specific seismic loading. However, the design basis criteria for such features include conservatisms that introduce margins to failure of such features, and consequently, of the corresponding SSCs to perform their function.

For nuclear power plants, a wide range of SSCs are susceptible to seismically induced damage due to resonance in the 1 to 10 Hz frequency range because of the configuration of SSCs. Empirical evidence, including earthquake data at nuclear power plants, demonstrates that high frequency vibratory motion (greater than 30 - 40 Hz) caused during earthquakes is not damaging to nuclear power plant SSCs. The criteria for determination of exceedance of the operating basis earthquake at nuclear power plants in RG 1.166, “Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post-Earthquake Actions,” March 1997 (Reference 40) focuses on the less than 10 Hz range. Exceptions to the susceptibility of SSCs in nuclear power plants in the 1 to 10 Hz range are the functional performance of vibration sensitive components, such as relays and other electrical and instrumentation devices. As stated in EPRI report 1025287, “Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic,” November 2012 (Reference 41), experience has shown a low likelihood of a seismically designed SSCs being damaged by ground motions with GMRS peak below 0.4g in the 1 to 10 Hz range. Therefore, plants exceeding the “Tier 1” threshold in the criteria for the proposed alternative seismic approach would have a higher likelihood of SSCs being impacted by seismic events.

EPRI report 3002012988 classifies as “Tier 3” those plants that were requested by the NRC to perform a plant-specific seismic risk evaluation to respond to the March 12, 2012, letter pursuant to 10 CFR 50.54(f) (Reference 42); hereafter, referred to as the 50.54(f) letter). All the plants that were requested to perform a seismic risk evaluation developed and submitted SPRAs. Therefore, the GMRS to SSE ratios for the “Tier 3” plants were such that a plant specific SPRA was deemed appropriate to understand the impact of seismic events on SSCs and plant response. The GMRS to SSE ratio for the plants that were requested to perform SPRAs was higher than the GMRS to SSE ratio at LaSalle.

The case studies supporting LaSalle’s application are based on plants that are classified by EPRI report 3002012988 as “Tier 3” (i.e., where the GMRS exceedance of the SSE was higher than that for LaSalle, and the NRC requested plant specific SPRAs). The case studies used SPRAs where the full range of the seismic hazard, including the high frequency portion, is included. The categorization conclusions from the case studies evaluated components susceptible of high frequency excitation, such as relays. LaSalle’s proposed alternative seismic approach includes explicit consideration of such components.

Based on its evaluation, as discussed above, the NRC staff finds that the proposed criteria for the alternative seismic approach to determine the applicability and use of the approach is acceptable because the criteria: (1) provide information about any seismic risk that would be unaccounted for in the current plant LB, (2) include exceedance of the plant-specific SSE by the GMRS from the re-evaluated hazard which can impact SSCs, (3) include seismic acceleration in the frequency range of 1 to 10 Hz where a wide range of nuclear power plant SSCs are susceptible to seismically induced damage, and (4) exclude plants where the so-called Tier 2 approach would be inappropriate for SSC categorization because the exceedance of the SSE by the GMRS from the re-evaluated hazard was high enough that a SPRA was deemed necessary by the NRC.

<u>Evaluation of Applicability of Criteria for the Proposed Alternative Seismic Approach to LaSalle</u>

The licensee compared the GMRS from the re-evaluated seismic hazard for LaSalle, developed in response to NTTF recommendation 2.1 dated March 31, 2014, against the site's design basis SSE to demonstrate that LaSalle meets the criteria for application of the proposed alternative seismic approach. The NRC staff previously evaluated the licensee's response to the 10 CFR 50.54(f) letter associated with NTTF recommendation 2.1, in which the licensee submitted its re-evaluated seismic hazard. The NRC staff's previous evaluation also includes confirmatory analysis of the seismic hazard. The NRC staff's previous assessment of the licensee's re-evaluated seismic hazard dated April 21, 2015 (Reference 43), states that the licensee's methodology was acceptable and that the GMRS, which was determined using the re-evaluated hazard, adequately characterized the site. Since the same re-evaluated hazard is used for comparison against the criteria for use of the proposed alternative seismic approach, the NRC staff's previous assessment on the re-evaluated hazard is applicable to this review. The NRC staff finds that the LaSalle GMRS is above the so-called Tier 1 criteria. Further, the NRC staff's review of the 10 CFR 50.54(f) letter confirmed that LaSalle was not required to perform a seismic PRA as part of the NRC's post-Fukushima actions.

Section 2.1 of EPRI report 3002012988 lists the "Tier 3" plants. The EPRI report states that several plants that have received extensions of their SPRA submittal dates are not included in the list in Section 2.1 of the EPRI report. Therefore, the list in Section 2.1 of the EPRI report is not comprehensive, and the NRC staff takes no position on the completeness or accuracy of that list; however, the completeness or accuracy of the list in Section 2.1 of the EPRI report does not impact the NRC staff's evaluation and conclusion on this application, because the staff's review is specific to LaSalle.

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

<u>Evaluation of the Implementation of the Proposed Alternative Seismic Approach</u>

The categorization conclusions from the case studies for plants A, C and D indicated that seismic-specific failure modes resulted in HSS categorization uniquely from SPRAs. Therefore, such seismic-specific failure modes, such as correlated failures, interaction failures, relay-chatter, and passive component structural failure mode, can influence the categorization process. The NRC staff reviewed the licensee's proposed alternative seismic approach to evaluate whether the categorization related conclusions from EPRI report 3002012988 were appropriately included and implemented.

In Section 3.2.3 of the enclosure to the LAR, the licensee discussed its proposed alternative seismic approach. The proposed alternative seismic approach includes a combination of qualitative and quantitative considerations of the mitigation capabilities as well as seismic failure modes of SSCs in the categorization process. These considerations are based on plant-specific walkdowns for the SSCs undergoing categorization, quantification of the impact of seismic failure of SSCs subject to correlated or interaction failures, and insights obtained from prior seismic evaluations performed for LaSalle.

Qualitative Evaluation for the Alternative Seismic Approach

The licensee stated that as part of the categorization team's preparation of the System Categorization Document (SCD) that is presented to the IDP, a section will be included which summarizes the identified plant specific seismic insights pertinent to the SSC being categorized. The licensee further explained that at several steps of the categorization process the categorization team will consider the available seismic insights relative to the system being categorized and document their conclusions in the SCD. In addition, the IDP would be provided with the basis for the proposed alternative seismic approach including the seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

Section 3.1.1 of the enclosure to the LAR included Figure 3-1 which identified two steps, represented by four blocks on the figure, that highlighted the review and consideration of seismic insights, including those from prior seismic evaluations, in the categorization of SSCs. In addition, Table 3-1 in the enclosure to the LAR included an explicit mention of the categorization evaluation for seismic hazard which would be performed at either the functional and/or component level.

The licensee explained that the categorization team would review available LaSalle specific seismic information and other resources to identify plant-specific seismic insights relevant to the SSCs being categorized such as:

- Impact of relay-chatter;
- Implications related to potential seismic interactions such as with block walls;
- Seismic failures of passive SSCs such as tanks and heat exchangers;
- Any known structural or anchorage issues with a particular SSC;
- Components that are implicitly part of PRA-modeled functions (including relays);
- Components that may be subject to correlated failures.

The licensee further explained that these insights would provide the IDP a means to consider potential impacts of seismic events in the categorization process. The licensee stated that the IDP could challenge, from a seismic perspective, any candidate LSS recommendation for any SSC if they believed there was basis for doing so and that any decision by the IDP to downgrade preliminary HSS components to LSS would also consider the applicable seismic insights.

The licensee explained that sources of the insights related to seismic events would be prior plant specific seismic evaluations such as the seismic hazard screening, spent fuel pool assessment, expedited seismic evaluation process as well as the seismic high frequency evaluation performed for NTTF recommendation 2.1, seismic walkdowns performed for NTTF Recommendation 2.3, and seismic mitigation strategy assessment performed for NTTF recommendation 4.2.

In Section 3.2.3 of the enclosure to the LAR, the licensee stated that for SSCs that were uniquely HSS from the FPRA but not HSS from IEPRA, the categorization team would review design-basis functions of the SSC(s) during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events. The results of the review would be presented to the IDP as additional qualitative inputs and would be described in the SCD. The licensee further clarified that the discussion with the IDP will focus on SSCs that are uniquely HSS from FPRA because such SSCs may not be categorized as HSS following the integrated importance measure determination.

The NRC staff concludes, based on its review of the qualitative evaluations for seismic risk in the licensee's proposed alternative seismic approach, that: (1) the evaluations will include potentially important seismically induced failure modes, as well as mitigation capabilities of SSCs during seismically induced design basis and severe accident events consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI report 3002012988, (2) the licensee will provide system-specific qualitative seismic insights to the IDP for consideration as part of the IDP review process as each system is categorized, (3) the insights will use plant-specific prior seismic evaluations, which, in conjunction with the performance monitoring for the proposed alternative seismic approach, reasonably reflect the current plant configuration, and (4) the qualitative evaluation will complement focused walkdowns and quantitative evaluations identified for the SSCs. Further, the recommendation for categorizing civil structures in the alternative seismic approach provides appropriate consideration of such failures from a seismic event.

<u>Focused Walkdowns for the Alternative Seismic Approach</u>

In Section 3.2.3 of the enclosure to the LAR, the licensee explained that the proposed alternative seismic approach would include focused walkdowns of SSCs undergoing categorization. The purpose of the walkdowns is to identify, for the SSCs that are being categorized, the conditions for occurrence of correlated failures, failure of more than one SSC due to interactions with other SSCs, and single component failures. In its supplement dated January 22, 2021, the licensee included, in its proposed alternative seismic approach, single component failures in a system, due to either seismic interaction or direct component failure modes, that result in total loss of a multi-train system where there is no other system that independently provides the same function. The licensee also provided a corresponding markup to the EPRI report in Attachment 2 of the January 22, 2021, supplement.

In its supplement dated October 16, 2020, the licensee provided additional details of and modifications to its proposed alternative seismic approach. The licensee stated that steps 3a, 3b, and 3c in the alternative seismic approach were elective and were intended to be applied similar to the screening processes employed in SPRAs. The licensee provided a mark-up of changes to the EPRI report in Attachment 2 of supplement dated October 16, 2020, that focused on the degree that the effort to screen out any SSC is less than the effort to keep the SSC on the list for further evaluation in the proposed approach. The licensee explained that the criteria in steps 3a, 3b and 3c were not intended to screen the SSCs being categorized from consideration as the “source” of a seismically induced interaction failure of other components. All SSCs in the system being categorized would have their interaction evaluated regardless of the answer to steps 3a, 3b and 3c. During the walkdowns, SSCs in the system being categorized will be evaluated to determine if they could physically impact one or more other components, either within the system being categorized or within another system which has a core damage mitigation or containment performance function. The licensee stated that the criteria in step 3a were not intended to screen these SSCs from being the “target” of a seismic interaction failure. The licensee provided a mark-up of changes to Figure 2-3 of the EPRI report in Attachment 2 of the supplement dated October 16, 2020, to clarify that SSCs can be screened out in Step 3 from consideration of a correlated functional failure of the SSC itself but still need to be considered for potential seismic interaction failures that could affect multiple SSCs. This was accomplished by revising the text in Step 4 and by adding a line from Step 1 into Step 5b to show that all system SSCs need to be considered in the seismic interaction review. The licensee also clarified that mitigation functions that are supported by an SSC are not hazard specific and apply to all evaluated hazards including seismic events.

The NRC staff reviewed the licensee's description of the Steps 3a, 3b, and 3c, as well as the impact on the decisions in those steps on the walkdowns. The NRC staff review was informed by a regulatory audit of the approach, which included an example demonstration of the proposed alternative seismic approach by the licensee. The NRC staff determined that all SSCs in the system being categorized would be evaluated. during the licensee's seismic walkdown. The staff also found that the licensee's evaluation would identify instances where the failure of other SSCs could cause interactions with SSCs in the system being categorized or vice versa. Based on its review, the NRC staff finds that the licensee's focused walkdown in the proposed alternative seismic approach: (1) includes consideration of seismically induced correlated and interaction failures that fail more than one SSC as well as single component failures, (2) includes evaluations of the direct and indirect impacts of seismically induced correlated and interaction failure of an SSC, (3) these failure modes reflect the insights from the case studies in the EPRI report, and (4) the modifications to the proposed alternative seismic approach through changes to the EPRI report appropriately reflect the evaluation of such direct and indirect impacts.

In its supplement dated October 1, 2020, the licensee stated that seismic walkdown methods have been documented in past SPRA and seismic margin assessment (SMA) methodology reports and identified relevant references as well as a training course. The licensee explained that the qualifications for seismic walkdowns for its proposed alternative seismic approach are similar to those for the SPRAs, including a recommendation that the walkdown team members take the above-mentioned EPRI training course or equivalent. In its supplement dated October 1, 2020, the licensee stated that the results of the walkdowns performed as part of the proposed alternative seismic approach will be documented similar to the documentation of walkdowns for SPRAs and subsequently conveyed to the 10 CFR 50.69 categorization team. The licensee identified information that would be included in the walkdown documentation and explained that documentation of the walkdown results will be developed and retained in accordance with NEI 00-04, Section 11, "Documentation and Change Control." The NRC staff did not review the details in the references cited by the licensee on walkdown methods and training. Therefore, the NRC staff does not make a finding on those documents and the training. The NRC staff's review identified several peer-reviewed SPRAs submitted by various licensees for risk-informed licensing and non-licensing actions that are supported by the references identified by the licensee. The NRC staff also noted that the SPID, Part 5 "Seismic PRAs" of the 2013 PRA Standard endorsed by the SPID, and the Code Case for Seismic PRAs (endorsed in RG 1.200, Revision 3); include the references identified by the licensee. Based on its review, the NRC staff finds that the qualification of personnel performing the walkdowns for the proposed alternative seismic approach is consistent with the state-of-practice for development and peer review of contemporary SPRAs. The NRC staff review also finds that the documentation and retention of walkdown information for the proposed alternative seismic approach is consistent with state-of-practice SPRAs and that the guidance in NEI 00-04 will result in appropriate information being presented to the IDP for categorization decisions. Therefore, the NRC staff concludes that the qualification of personnel performing the walkdowns and the documentation as well as retention of the walkdown results is acceptable for the proposed alternative seismic approach.

Appendix B of EPRI report 3002012988 provides guidance on “capacity-based screening for high capacity SSCs.” The guidance includes a screening value of the seismic CDF and if a single SSC were to contribute that value or less, it could be screened from evaluation under the proposed alternative seismic approach. In its supplement dated October 16, 2020, the licensee provided the basis for the selected capacity-based screening criteria of 2.5E-7 per year (i.e., a high confidence in low probability of failure (HCLPF) value that, when convolved with the site-specific hazard, results in a CDF of 2.5E-7 or less). The licensee stated that the option to choose another site-specific screening criterion will be removed. A mark-up of changes to the EPRI report provided in Attachment 2 to the supplement dated October 16, 2020, reflect the licensee’s response. Based on its review of the capacity-based screening criterion for the proposed alternative seismic approach, the NRC staff determined that the licensee’s approach for selecting the screening criterion is consistent with that for state-of-practice SPRAs and that SSCs screened out based on the criterion are not expected to result in HSS components within the 50.69 categorization process.

Appendix B of the EPRI report additionally provides guidance on the development of fragility values for SSCs to support the capacity-based screening. In its supplement dated October 16, 2020, the licensee identified Section 3.3, “Separation of Variables Fragility Approach”, and Section 3.4, “Hybrid Fragility Approach,” of EPRI 3002012994, “Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments,” September 2018 (Reference 44), for the appropriate fragility methods. Further, the licensee identified Sections 6.4, “Special Considerations for High-Frequency Seismic Demands,” and Section 6.7, “Response Analysis Scaling Methods for scaling methods.” The licensee stated that options other than those identified above (e.g., the American Society of Civil Engineers [ASCE] 7 criteria) will not be used. Attachment 2 of the supplement dated October 16, 2020, included a mark-up of conforming changes to the EPRI report. The NRC staff reviewed the licensee’s supplement as well as the changes to the EPRI report considering the fragility methods used in multiple state-of-practice SPRAs as well as corresponding guidance documents such as the SPID. Based on its review, the staff finds that the separation of variables approach, and hybrid fragility approach represent the state-of-practice approaches used in contemporary SPRAs. The NRC staff review also finds that the scaling approaches identified by the licensee are state-of-practice approaches for scaling in-structure response spectra. Consequently, the NRC staff finds that the fragility approaches proposed for development of fragility values in Step 5b of the proposed alternative seismic approach are acceptable for the proposed alternative seismic approach because (1) they represent state-of-practice approaches consistent with those used in contemporary SPRAs reviewed by the NRC staff, and (2) no unreviewed methods would be used for fragility calculations.

In its supplement dated October 16, 2020, the licensee explained that, as with an SPRA, the personnel performing these fragility calculations would need to have the background or experience with the state-of-the-practice fragility methods (i.e., those identified in the previous paragraph) and that these fragility calculations would be independently checked by another engineer with this same fragility background or experience. In addition, the licensee stated that one of the quality measures listed in Section 3.3 of NEI 00-04 is the “use of personnel qualified for the analysis” and that applied to the proposed alternative seismic approach. The licensee further stated that Section 3.3 and Section 11 of NEI 00-04 apply to the evaluations performed in the proposed alternative seismic approach. The NRC staff finds that personnel performing fragility evaluations for the proposed alternative seismic approach will have experience or background consistent with that used for state-of-practice SPRAs as well as the guidance in NEI 00-04 on personnel qualifications and the use of such personnel is therefore, acceptable for the proposed alternative seismic approach. In addition, the NRC staff review determined that the documentation of the fragility evaluations will be consistent with documentation used for other categorization processes and is therefore, acceptable for the proposed alternative seismic approach.

In its supplement dated October 1, 2020, the licensee stated that Section 5 of NEI 00-04 provides guidance for SSCs that are implicitly modeled in a PRA which would address categorization of subcomponents such as relays. The licensee explained that EPRI report 3002012988 identified seismically correlated SSCs, and seismic-induced interactions, which could similarly lead to seismic vulnerabilities of multiple SSCs, to be unique to seismic risk considerations compared to other parts of the 10 CFR 50.69 process.

The NRC staff determined that LaSalle's proposed alternative seismic approach will result in consideration of relays as implicitly modeled components and of insights related to the impact of seismically induced relay-chatter for the function achieved by the SSC during the categorization. Section 3.2.3 of the enclosure to the LAR explicitly includes such insights in the information provided to the IDP as part of the categorization process. The NRC staff also determined that the focused walkdowns of SSCs undergoing categorization will identify seismic interaction and correlated failures including those resulting from potential failures of passive components as well as structural and anchorage issues. Further, the NRC staff concludes that insights from available plant-specific seismic reviews will also provide categorization related insights from a seismic failure modes perspective.

Quantitative Evaluation for the Alternative Seismic Approach

In Section 3.2.3 of the enclosure to the LAR, the licensee explained that SSCs identified as being vulnerable to correlated or interaction failure modes based on the walkdown would be subjected to a quantitative evaluation using on the licensee's IEPRA to determine the impact of seismic events on the categorization. The quantitative evaluation would be performed through a sensitivity study, termed the surrogate sensitivity, using the licensee's IEPRA. The surrogate sensitivity would be performed by introducing PRA basic events, termed surrogate events, in the licensee's IEPRA at appropriate locations to reflect seismically induced correlated failure or interaction failure of single or multiple SSCs. Subsequently, the modified IEPRA with the surrogate events would be quantified for the loss-of-offsite power (LOOP) and small break loss-of-coolant accident (hereafter referred to as small loss-of-coolant accident (LOCA)) initiators and importance measures would be derived. The importance measures for the surrogate events derived from this sensitivity study would be used to identify the SSCs that should be HSS due to seismically correlated failures or seismic interaction related failures. The licensee further stated that the quantitative evaluation to determine the importance of SSCs on a system basis in the proposed alternative seismic approach was detailed in Section 2.3.1 of EPRI report 3002012988. The guidance in Section 2.3.1 of the EPRI report provides details of the sensitivity evaluation performed using the licensee's IEPRA.

Item 8 in Section 2.3.1 of EPRI report 3002012988 states that the LOOP and small LOCA initiators in the IEPRA will be used, in conjunction with surrogate events, to determine the impact of seismic-specific failures of SSCs following the walkdown. In its supplement dated October 16, 2020, the licensee justified the selection of the LOOP and small LOCA initiators for the surrogate sensitivity stating that insights from the SPRAs used for the case studies in the EPRI report as well as other industry analyses were used to determine which seismic accident sequences would capture the important insights of seismic impacts on SSC categorization. The licensee also provided a comparison of critical safety functions between LOOP, small LOCA, medium break LOCA (hereafter referred to as medium LOCA), and large break LOCA (hereafter referred to as large LOCA) sequences for boiling-water reactors and pressurized-water reactors to show that LOOP and small LOCA sequences challenge the majority of critical safety functions. The NRC staff reviewed the information in the supplement and considered insights from several contemporary SPRAs reviewed by the staff for regulatory decisions. The NRC staff review was informed by a regulatory audit. Based on its review, the staff determined that offsite power fragility is one of the lowest fragilities and that seismically induced LOOP is expected to be a dominant sequence from a seismic risk perspective. In addition, the fragility for medium and large LOCAs is either same as or higher than that for small LOCA. Therefore, seismically induced small LOCA would have the same or higher occurrence frequency compared to seismically induced medium and large LOCAs. Furthermore, LOOP and small LOCA sequences challenge the same, if not more, critical safety functions as initiators, including medium and large LOCAs. The NRC staff determined that the seismically induced small LOCA accident sequences in the surrogate sensitivity analysis include failure of offsite alternating current (AC) power and that the surrogate sensitivity analysis will include accident sequences resulting from probabilistic failures in conjunction with LOOP and small LOCA (i.e., Station Blackout [SBO], LOOP/SBO with Anticipated Transient Without Scram [ATWS], and small LOCA with ATWS). Based on its review, the NRC staff concludes that the LOOP and small LOCA initiators are acceptable for the proposed alternative seismic approach because: (1) seismically induced LOOP is expected to be a dominant sequence, (2) seismically induced small LOCA would have the same or higher occurrence frequency as seismically induced medium and large LOCAs, (3) LOOP and small LOCA sequences challenge the same, if not more, critical safety functions as other LOCA sizes and initiators, and (4) the surrogate sensitivity includes probabilistic failures in conjunction with LOOP and small LOCA including failure of offsite AC power.

The licensee proposed the use of 1.0E-4 as the failure probability for surrogate events introduced in the licensee's IEPRA for the surrogate sensitivity study. The licensee explained that the failure probability of 1.0E-4 was based on the occurrence frequency for a seismically induced LOOP and the failure probability of the surrogate event at that annual exceedance probability of 1.0. The licensee stated that these values cannot be used directly in the proposed approach because they challenge the calculation of the RAW importance measure. To avoid this issue, the proposed approach switches the seismically induced LOOP frequency of 1.0E-4 and the surrogate failure probability of 1.0. The licensee provided justification for the seismically induced LOOP frequency based on three SPRAs used for the case studies in the EPRI report. Similarly, the licensee provided justification for the selected seismically induced small LOCA frequency of 1.0E-2 per year. The NRC staff notes that selected seismically induced small LOCA frequency of 1.0E-2 per year, when multiplied by the 1.0E-4 failure probability for the surrogate event, is equivalent to an occurrence frequency of 1.0E-6 per year and a failure probability of the surrogate event at that annual exceedance probability of 1.0.

In its supplement dated October 16, 2020, the licensee stated that its proposed alternative seismic approach will use an initiating frequency of 1.0 per year for LOOP and 1.0E-2 per year for small LOCA and that other values will not be used. Further, in its supplement dated January 22, 2021, the licensee stated that a failure probability of 1.0E-4 will be the only value used for the surrogate event and no alternate values will be used. The licensee clarified that use of "more realistic seismic-induced failure probabilities" no longer applied to the proposed alternative seismic approach. The licensee also provided a markup of changes to the EPRI report in Attachment 2 of the January 22, 2021 supplement that deleted the use of "other appropriately justified values" for surrogate event failure probability.

The NRC staff determined that seismically induced LOOP and small LOCA occurrence frequencies are representative for LaSalle based on the three SPRAs in the case studies in the EPRI report and the fact that the seismic hazard at the licensee's site is lower than the hazard for those SPRAs. Therefore, the NRC staff concludes that the proposed occurrence frequency for the seismically induced LOOP event of 1.0 per year, the proposed occurrence frequency for the seismically induced small LOCA event of 1.0E-2 per year, and the proposed surrogate event failure probability of 1.0E-4 are acceptable for use in the licensee's alternative seismic approach. Further, the NRC staff determined that the occurrence frequency and failure probability switch in the surrogate sensitivity is acceptable for the licensee's alternative seismic approach because: (1) it is necessary for developing the importance measures for comparison against the corresponding thresholds in NEI 00-04, and (2) it does not alter the basis for the proposed values.

The licensee, in its supplement dated October 16, 2020, provided details on the modification of the IEPRA for the surrogate sensitivity study to reflect typical seismic impacts included in probabilistic seismic risk assessments. In addition, the licensee stated that direct current (DC) power restoration credit would not be applied for the surrogate sensitivity study. The licensee provided a mark-up of changes to the EPRI report to reflect the details of the modification of the IEPRA for the surrogate sensitivity study. Based on its review of the information in the supplement and the changes to the EPRI report, the NRC staff finds that the modifications to the IEPRA for the surrogate sensitivity study reflect the primary impacts of a seismic event on plant response and are therefore, acceptable for the proposed alternative seismic approach.

The LAR stated that the proposed alternative seismic approach will use the Fussell-Vesely (F-V) and RAW thresholds for common-cause events in NEI 00-04, Revision 0, to determine the categorization outcome of the surrogate sensitivity study. In its supplement dated October 16, 2020, the licensee justified the use of the RAW threshold for common-cause events by explaining that the surrogate sensitivity in the proposed alternative seismic approach accounts for correlated events and interaction events, which fail more than one component. The licensee stated that such a failure mode is a common-cause failure and should be treated as such. The licensee further stated that, where an SPRA is used, correlated events and interaction events would use the HSS RAW threshold of 20 based on the discussion of such failure modes in Section 5.1 of the NEI 00-04. In its supplement dated January 22, 2021, the licensee stated that single component failure in a system due to either seismic interaction or direct component failure modes, that result in total loss of a multi-train system and where there is not another system that independently provides the same function, will be included in the proposed approach. The licensee further confirmed that the F-V and RAW thresholds for such single component failures will be consistent with the guidance in NEI 00-04 (i.e., RAW threshold of 2). The licensee also provided a markup of changes to the EPRI report in Attachment 2 of the January 22, 2021, supplement that reflected the use of F-V and RAW thresholds consistent with the guidance in NEI 00-04 for such single component failures. The NRC staff finds that the use of the importance measure thresholds for the surrogate sensitivity is acceptable for the proposed alternative seismic approach, because the F-V and RAW thresholds for the correlated and interaction failures that fail more than one SSC as well as single component failures are consistent with the values in NEI 00-04, Revision 0, as endorsed by the NRC.

In its supplement dated January 22, 2021, the licensee stated that "EGC has implemented 50.69 at multiple sites and has confirmed that categorization is a robust process and no single consideration (e.g., seismic risk) dominates the results for a single SSC." The NRC staff did not rely on this statement in making its finding on any aspect of the proposed alternative seismic approach including the use of the importance measure thresholds for different seismically induced failure modes in this approach. Therefore, the NRC staff is not making a finding on the information in the cited sentence related to the implementation of 10 CFR 50.69 at multiple EGC sites.

In its supplement dated October 16, 2020, the licensee stated that surrogate events incorporated in the licensee's IEPRA for the categorization of system will not be retained and included in the sensitivity studies for subsequent and distinct system categorizations. The licensee explained that the categorization is performed on a system by system basis and that its not retaining the surrogate events will be consistent with the overall categorization process. The licensee explained that programs using importance measures (e.g. Mitigating Systems Performance Index [MSPI]) have shown that dominant contributors can reduce the importance of components, therefore, retaining surrogates from system to system may reduce the importance of components in one system. The NRC staff finds that not retaining surrogate events from the categorization of one system in the categorization of another distinct system is acceptable for the alternative seismic approach because: (1) it is consistent with the overall categorization process, and (2) it provides categorization results that do not keep changing based on the number of systems categorized.

The licensee's supplement dated October 16, 2020, included an addition to Step 5c in the markup of the EPRI report. The addition provided guidance on the inclusion and retention of basic events for SSCs that are screened using Step 5c in the IEPRA model. In its supplement dated January 22, 2021, the licensee clarified that the additional guidance about inclusion and retention of basic events for SSCs that are screened using Step 5c in the IEPRA model will not be utilized in its proposed alternative seismic approach. The licensee also provided a markup to the EPRI report in Attachment 2 of the January 22, 2021 supplement that deleted the discussion of the addition from Step 5c and Appendix B (Introduction and Section B.1) of the EPRI report. The NRC staff finds that the non-inclusion of the new addition to Step 5c in the licensee's proposed alternative seismic approach is acceptable because the non-inclusion does not change or negatively impact the primary purpose of Step 5c as well as the overall proposed approach.

The licensee explained the consideration of the results from the surrogate sensitivity in the IDP decision in the supplement dated October 16, 2020. The licensee explained that because the proposed alternative seismic approach is a pseudo-deterministic evaluation process, HSS designations arising from implementation of the approach would not be subject to reconsideration by the IDP. The licensee stated that the "one exception to the rule" was an HSS determination that was solely due to a seismic interaction concern, and it could be shown that special treatment of the SSC would not change the likelihood of failure or its consequences. For such cases, the IDP may consider categorizing the SSC as LSS. The licensee provided an example where the HSS designation was due to the interaction failure from a block wall and it could be shown that the failure of the block wall would fail the equipment regardless if the equipment was purchased, installed, and tested with special treatment or not. The NRC staff finds that the consideration of the categorization results from the proposed approach by the IDP will reflect the pseudo-deterministic evaluation of the impacts of seismic events and therefore, acceptable for the proposed alternative seismic approach. The NRC staff also finds that it is appropriate for the IDP to consider changing the categorization of cases with HSS determination from the proposed alternative seismic approach solely due to a seismic interaction concern where justification can be provided to the IDP that special treatment would not change likelihood of failure or the consequences for such cases.

In its October 16, 2020, supplement, the licensee provided an example comparison between the categorization results for different SSCs in a system using the surrogate sensitivity, considering the IEPRA for Plant A from the EPRI case study and the Plant A SPRA. The licensee discussed the approach for performing the example comparison. The surrogate sensitivity in the example comparison used a failure probability of 1.0E-4 for the surrogate events and initiating frequencies of 1.0 per year and 1.0E-2 per year for seismically induced LOOP and small LOCA, respectively. The comparison showed that the categorization outcome from the surrogate sensitivity is comparable to that from a SPRA. Based on its review, the NRC staff finds reasonable confidence that the categorization outcome from the licensee's proposed alternative seismic approach will be comparable to those from SPRAs.

<u>Conclusions on the Implementation of the Alternative Seismic Approach</u>

Based on its review of LaSalle's proposed alternative seismic approach, in conjunction with requirements in 10 CFR 50.69 and the corresponding SOC, the NRC staff finds that the proposed alternative seismic approach provides reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(ii) and (iv) and meets the intent of the SOC because:

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

2. It includes focused walkdown(s) which evaluate(s) the direct and indirect impacts of seismically induced correlated failures, interaction failures, and single component failures in a system under categorization.

3. It includes a quantitative evaluation, with justified failure probability and initiating event frequencies, that provides reasonable confidence that the categorization results from the licensee's proposed alternative seismic approach will be similar to those from SPRAs.

4. Personnel performing necessary walkdowns and analyses will have qualifications consistent with the state-of-practice SPRAs and the guidance in NEI 00-04. The documentation of these walkdowns and analyses will be consistent with state-of-practice SPRAs and the guidance in NEI 00-04.

5. The quantitative and qualitative insights presented to the IDP include potentially important seismically induced failure modes as well as mitigation capabilities of SSCs during seismically induced design basis and severe accident events, consistent with the conclusions on the determination of unique HSS SSCs from SPRAs in EPRI report 3002012988. The quantification will use the licensee's IEPRA and the insights will use prior plant specific seismic evaluations. Therefore, in conjunction with performance monitoring for the proposed alternative seismic approach, the proposed alternative seismic approach will reasonably reflect the current plant configuration.

6. It presents system-specific insights and categorization results from a seismic risk perspective to the IDP for consideration as part of the IDP review process, thereby providing the IDP with a means to consider potential impacts of seismic events in the categorization process.

7. It presents the IDP with the basis for the proposed alternative seismic approach including the moderate seismic hazard for the plant and the criteria for use of the proposed alternative seismic approach.

<u>Evaluation for Performance Monitoring for the Alternative Seismic Approach</u>

In Section 3.5 of the enclosure to the LAR, the licensee stated that its configuration control process ensured that changes to the plant, including a physical change and changes to documents, are evaluated to determine the impact on design bases, licensing documents, programs, procedures, and training.

The NRC staff evaluated the licensee's discussion of its performance monitoring program for the proposed alternative seismic approach to ensure: (1) the continued validity of the plant-specific information that were developed for each SSC that is categorized, (2) that any changes to the plant, including the seismic hazard, are captured and appropriately addressed as part of the 10 CFR 50.69 program, and (3) that the requirements in 10 CFR 50.69(e) were met for the proposed alternative seismic approach.

In its LAR, the licensee stated that its performance monitoring process requires periodic review to assess changes that could impact the categorization results and to provide the IDP with an opportunity to recommend categorization and treatment adjustments due to such changes. The licensee explained that its configuration control program had been updated to have a checklist which would include:

- A review of the impact on the SCD for configuration changes that may impact a categorized system under 10 CFR 50.69.

- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected including identification of any potential seismic interaction between added or modified components and new or existing safety-related or safe shutdown components or structures. Review of impact to seismic loading, SSE seismic requirements, as well as the method of combining seismic components.

- Review of seismic dynamic qualification of components if the configuration change adds, relocates, or alters Seismic Category I mechanical or electrical components.

The licensee stated that its performance monitoring program required that SCDs could not be approved by the IDP until the panel's comments on issues, including system-specific seismic insights, had been resolved to the satisfaction of the IDP.

The licensee explained that its scheduled periodic reviews would occur no longer than once every two refueling outages and would evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it was determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process would be updated. The licensee explained that if a PRA model or other risk information is updated, a review of the SSC categorization would be performed in addition to the periodic review.

The NRC staff recognizes that the seismic hazard at any site could potentially increase such that categorization process may be impacted from a seismic risk perspective, either solely due to the seismic risk or via the integrated importance measure determination. In this regard, the licensee stated that if the LaSalle seismic hazard changed at some future time, the licensee would follow its categorization review and adjustment process procedures and would update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e). In its supplement dated January 22, 2021, the licensee stated that if its categorization review and adjustment process determines that an approach different from the proposed alternative seismic approach is necessary for consideration of seismic risk in its categorization program, it will seek prior NRC approval for use of such an approach consistent with its proposed license condition.

Based on its review, the NRC staff finds that the licensee's configuration control program includes consideration of seismic issues as well as failure modes such as interaction between components and review of seismic loading and seismic dynamic qualification. Further, the licensee's performance monitoring program assesses changes that impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments due to such changes. Therefore, the NRC staff finds that the licensee's performance monitoring and configuration control process addresses plant-specific seismic evaluation, thereby ensuring that the corresponding impacts on SSC categorization continues to remain valid and if necessary, are presented to the IDP for consideration of categorization changes.

The licensee stated that if significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can or did prevent a safety significant function from being satisfied, an immediate evaluation and review would be performed prior to the normally scheduled periodic review. During its review, the NRC staff noted that the licensee's performance monitoring program for 10 CFR 50.69 has the capability to identify significant changes to the plant risk profile as well as instances in which a RISC-3 or RISC-4 SSC may fail to perform a safety significant function, resulting in an immediate evaluation and review for such instances. The NRC staff further finds consideration of changes to the seismic hazard at the plant and its impacts are included in the feedback mechanism for the proposed alternative seismic approach at LaSalle. Based on its review, the NRC staff finds that the requirements in 10 CFR 50.69(e) are met for the proposed alternative seismic approach.

In sum, the NRC staff determined that: (1) the licensee's programs provide reasonable assurance that the existing seismic capacity of LSS components would not be significantly impacted by alternative treatments permitted by 10 CFR 50.69, and (2) the monitoring and configuration control program ensures that potential degradation of the seismic capacity would be detected and addressed before it significantly impacts the plant risk profile. Therefore, the NRC staff finds reasonable confidence that the licensee's proposed alternative seismic approach is a systematic process for evaluating seismic hazards for impact on SCC categorization which meets the requirements in 10 CFR 50.69(c)(2)(iv).

Conclusion for Proposed Alternative Seismic Approach

Based on its review, the NRC staff concludes that the licensee's proposed alternative seismic approach, including the criteria for its use, is acceptable for considering seismic risk in the licensee's categorization process under 10 CFR 50.69.

3.3.1.2.2 External Hazards and Other Hazards (Non-Seismic)

The NRC staff reviewed the licensee's discussion of its consideration of non-seismic external hazards and other hazards provided in Section 3.2.4 of the enclosure to the LAR and Attachment 4 of that enclosure. Non-seismic external hazards include high winds, external flood hazards, transportation, nearby facility accidents, and other hazards listed in Appendix 6-A of the 2009 ASME/ANS PRA Standard (RA-Sa-2009). The licensee evaluated all non-seismic external hazards and other hazards for the 10 CFR 50.69 application using a plant-specific evaluation in accordance with GL 88-20 and the criteria in the 2009 ASME/ANS PRA Standard.

In its supplement dated October 1, 2020, the licensee stated that four non-seismic external hazards credited SSCs per Figure 5-6 of NEI 00-04 to allow the hazard to screen and that all other non-seismic external hazards listed in Attachment 4 of the enclosure to the LAR were screened without crediting SSCs. The licensee identified the four non-seismic external hazards that credited SSCs in their screening as transportation accident; external flooding; extreme wind or tornado; and turbine missiles. The NRC staff finds that the licensee's approach for categorizing SSCs credited for screening non-seismic external hazards is consistent with the guidance in NEI 00-04.

In Attachment 4 of the enclosure to the LAR, the licensee explained that the results of its flood hazard reevaluation report (FHRR dated March 12, 2014, (Reference 45)) indicated that external flooding from all mechanisms except local intense precipitation (LIP) and probable maximum storm surge (PMSS) were bounded by the plant's current licensing basis (CLB). The licensee further explained that its focused evaluation (FE dated March 8, 2017; (Reference 46)) determined that there were no impacts to safety-related SSCs from the LIP and PMSS events, and the design basis of the plant was adequate to mitigate the effects from those external flood-causing mechanisms with sufficient margin.

The NRC staff previously reviewed the licensee's FHRR and determined that the licensee conducted the hazard reevaluation using present-day methodologies and regulatory guidance used by the NRC staff dated January 10, 2017 (Reference 47). The NRC staff's previous review also confirmed that the re-evaluated flood hazard results for LIP and PMSS flood-causing mechanisms are not bounded by the plant's CLB. The NRC staff previously reviewed the licensee's FE on August 12, 2017 (Reference 48), and concluded that effective flood protection existed from the re-evaluated flood hazards. The NRC staff's previous review of the licensee's FE also noted that the licensee did not rely on any personnel actions or new modifications to the plant in order to respond to the LIP and PMSS events. In Attachment 4 of the enclosure to the LAR, the licensee stated that in accordance with the external hazard screening process per Figure 5-6 of NEI 00-04, several flood doors integral to flood protection at LaSalle were identified for categorization as HSS SSCs should their associated systems be categorized. The NRC staff finds that the endorsed guidance in Figure 5-6 in NEI 00-04 provides the appropriate approach for categorizing SSCs that are credited for screening external flooding.

Based on the NRC staff's review of the information provided by the licensee in Attachment 4 of the enclosure to the LAR, the supplement dated October 1, 2020, as well as the NRC staff's previous assessment of the licensee's focused evaluation for the re-evaluated external flood hazard for the site, the staff finds that the licensee's SSC categorization process will evaluate the safety significance of any SSCs for the external flooding hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

In its supplement dated October 1, 2020, the licensee stated that details of the high winds and tornado missile hazard screening analysis were not included in the 10 CFR 50.69 LAR, but were included in Enclosure 4, Section 4 of the licensee's TSTF-505 LAR dated January 31, 2020 (Reference 49). In that LAR, the licensee stated that Section 3.3.2.1 of its UFSAR (Reference 50) provided the design basis for tornado hazard. Based on its review of the design basis in Section 3.3.2.1 of the UFSAR and NUREG/CR-4461, "Tornado Climatology of the Contiguous United States," February 2007 (Reference 51), the NRC staff finds that the occurrence frequency of the design basis wind speeds for LaSalle is less 1.0E-6 per year. The staff also notes that the primary concern for high winds is LOOP caused by the winds and that the internal events PRA already includes LOOP events due to severe weather including high winds and tornados. Therefore, the NRC staff finds that the impact of high winds and tornado missiles on plant response and the resulting categorization of SSCs is included in the categorization process.

In Enclosure 4, Section 4 of its TSTF-505 LAR, the licensee explained that two systems were determined to be vulnerable to tornado missiles and not in conformance with the LaSalle design and licensing bases. These systems were identified as the main control room (MCR) heating ventilation and air conditioning system (VC) and the auxiliary electrical equipment room (AEER) heating ventilation and air conditioning system (VE). The licensee explained that the floor above the level where systems VC and VE are located is 6 inches thick reinforced concrete and was not evaluated to protect against design basis vertical missiles. The licensee stated that only a few missiles types are capable of perforating or failing a 6 inch concrete floor and that the occurrence frequency of tornado wind speeds needed to develop missiles (design basis and beyond) capable of penetrating or failing the 6 inch concrete floor above the vulnerable SSCs is much less than 1E-6 per year. The licensee further explained that the VC and VE systems are each spatially separated by train, with at least 20 feet between components from different trains. The licensee stated that a loss of ventilation to the control room and/or the AEER would not result in immediate failure to any safety-related or risk significant components because several hours would elapse before the affected rooms would reach temperatures that could potentially result in higher failure rates for components. The licensee further stated that procedures were available to mitigate such failures and that based on room heat-up calculations and the LaSalle UFSAR, it was assumed that at least 2 hours are available to perform the above actions before design temperatures in the MCR or AEER are reached. In its supplement dated October 1, 2020, the licensee stated that the procedures and applicable equipment associated with mitigation were not required for screening and only provided additional defense-in-depth. Further, the licensee stated that the only SSCs credited for screening high winds and tornado-generated missiles are Seismic Category I structures, which are already considered HSS for 10 CFR 50.69 categorization. As noted above, in its supplement dated October 1, 2020, the licensee stated that extreme winds or tornado hazard credited SSCs per Figure 5-6 of NEI 00-04 to allow the hazard to screen. Based on its review, the NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for the extreme winds or tornado hazard consistent with the guidance provided in NEI 00-04, as endorsed by the NRC.

In its supplement dated October 1, 2020, the licensee explained that the basis for screening the turbine missile hazard is the probability of the turbine missile bounding mean CDF being less than 1E-6 per year. As noted above, in its supplement dated October 1, 2020, the licensee stated that the turbine missile hazard credited SSCs to allow the hazard to screen. Therefore, per Figure 5-6 of NEI 00-04, a failure of those SSCs would make the hazard unscreened and consequently, those SSCs, if categorized, would be HSS.

In summary, the NRC staff finds that the licensee's SSC categorization process will evaluate the safety significance of SSCs for non-seismic external hazards and other hazards consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in NEI 00-04, Revision 0, and RG 1.201, Revision 1. The NRC staff concludes that the licensee's consideration of non-seismic and other external hazards is acceptable for this application.

3.3.1.2.3 Component Safety Significance Assessment for Passive Components

In Section 3.1.2 of the LAR, the licensee proposed using a categorization method for passive components that was not cited in NEI 00-04, Revision 0, or RG 1.201, Revision 1, but was approved by the NRC for 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 (Reference 52). 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 SSCs solely based on consequences, which measures the safety significance of the pipe assuming 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 repair/replacement methodology is acceptable and appropriate for passive component categorization of Class 2 and Class 3 SSCs.

In Section 3.1.2 of the LAR, the licensee stated, "[t]he passive categorization process is intended to apply the same risk-informed process accepted in the AN0-2-R&R-004 for the passive categorization of Class 2, 3, and non-class components. Consistent with AN0-2-R&R-004, Class 1 pressure retaining SSCs in the scope of the system being categorized will be assigned HSS and cannot be changed by the IDP." The NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.3.1.3 Key Principle 4: Conclusions

As discussed above, Key Principle 4 states that 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. Based on the NRC staff's review for IEPRA (includes internal floods) and FPRA model acceptability, the evaluation of the proposed alternative seismic approach, and the use of non-PRA methods, set forth above, the NRC staff concludes that the proposed change satisfies the fourth key principle for risk-informed decision-making prescribed in RG 1.174, Revision 3 (Reference 8).

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.

Sections 11 and 12 of NEI 00-04, Revision 0, includes a 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 LaSalle station as-built, as-operated plant.

The NRC staff's evaluation of and findings on the licensee's performance monitoring as part of the alternative seismic approach is provided in Section 3.3.1.2.1 of this SE.

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

4.0 CHANGES TO THE OPERATING LICENSES

The NRC staff finds the categorization approaches proposed in the LAR are acceptable to meet 10 CFR 50.69. Use of an approach other than those proposed in the LAR would necessitate prior NRC approval to ensure that the approach meets the requirements in 10 CFR 50.69.

Based on the NRC staff's review of the LAR and the licensee's responses to the staff RAIs, the NRC staff identified specific actions, as described below, that support the NRC staff's conclusion that the proposed program meets the requirements in 10 CFR 50.69 and the guidance in RG 1.201, Revision 1 and NEI 00-04, Revision 0.

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 program is conditioned upon the License Condition provided below that delineates completion of the implementation of listed items and prerequisites to address changes to the PRA models or documentation. These implementation items are identified in the licensee's letter dated October 29, 2020 (Reference [4]).

The licensee proposed the following amendment to the RFOLs for LaSalle, Units 1 and 2. The proposed license condition states:

> Exelon Generation Company, LLC (EGC) 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 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 the EGC submittal letter dated

> January 31, 2020, and all its subsequent associated supplements, as specified in License Amendment No. [249/235] dated May 27, 2021.
>
> EGC will complete the implementation items listed in Table APLA-01.2 in Attachment 1 of EGC letter to NRC dated October 29, 2020, prior to implementation of 10 CFR 50.69 program. All issues identified will be addressed and any associated changes will be made, focused-scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA Standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> 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 PRA approach).

The NRC staff finds that the proposed license condition is acceptable, because: (1) the license condition adequately implements 10 CFR 50.69 using models, methods, and approaches that are consistent with applicable guidance that has previously been endorsed by the NRC; (2) as discussed in SE Section 3.3.1.2.1, the proposed alternative seismic approach for assessing seismic risk for this application is acceptable; and (3) as discussed in SE Section 3.3.1.2.3, the use of the ANO-2 methodology for passive categorization is acceptable for use at LaSalle. Accordingly, the NRC staff approves the licensee's proposed license condition.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Illinois official was notified of the proposed issuance of the amendment on March 18, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding April 7, 2020 (85 FR 19511). Accordingly, the amendment meets 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 amendment.

7.0 CONCLUSION

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

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SUBJECT: LASALLE COUNTY STATION, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 249 AND 235 RELATED TO APPLICATION TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2020-LLA-0017) DATED MAY 27, 2021

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DATE	03/29/2021	03/29/2021	04/05/2021	03/23/2021
OFFICE	NRR/DEX/EMIB/BC*	NRR/DEX/EEEB/BC*	NRR/DEX/EEOB-EICB/BC*	NRR/DRA/APLA/BC
NAME	ABuford	BTitus	MWaters	RPascarelli
DATE	04/07/2021	03/30/2021	04/09/2021	03/17/2021
OFFICE	NRR/DRA/APLB/BC	NRR/DRA/APLC/BC	NRR/DSS/SCPB/BC*	OGC *
NAME	JBorromeo(A)	SRosenberg	BWittick	STurk
DATE	03/17/2021	03/17/2021	03/25/2021	05/17/2021
OFFICE	NRR/DORL/LPL3/BC*	NRR/DORL/LPL3/PM*		
NAME	NSalgado	BVaidya		
DATE	05/27/2021	05/27/2021		

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