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

July 27, 2021

Mr. James Barstow
Vice President, Nuclear Regulatory
Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 317, 340, AND 300 REGARDING ADOPTION OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS SECTION 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER PLANTS" (EPID L-2020-LLA-0162)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 317, 340, and 300 to Renewed Facility Operating Licenses Nos. DPR-33, DPR-52, and DPR-68 for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively. These amendments are in response to your application dated July 17, 2020, as supplemented by letter dated April 28, 2021.

The amendments revise the Browns Ferry, Units 1, 2, and 3, Renewed Facility Operating Licenses to add a new license condition to allow for the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors."

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

Sincerely,

/RA/

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

Docket Nos. 50-259, 50-260, and 50-296

Enclosures:
1. Amendment No. 317 to DPR-33
2. Amendment No. 340 to DPR-52
3. Amendment No. 300 to DPR-68
4. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 317
Renewed License No. DPR-33

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee) dated July 17, 2020, as supplemented by letter dated April 28, 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 Title 10 of the *Code of Federal Regulations* (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, Renewed Facility Operating License No. DPR-33 is amended by the changes indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2021.07.27 16:31:53 -04'00'

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

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: July 27, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 317

BROWNS FERRY NUCLEAR PLANT, UNIT 1

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace the following pages of Renewed Facility Operating License No. DPR-33 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
6f	6f
7	7
-	8

(23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

(24) TVA shall close all open Facts and Observations (F&Os) listed in Tables 11 and 13 to Attachment 2 of TVA Letter CNL-20-003, “Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (BFN-TS-516),” dated March 27, 2020, prior to implementing any Surveillance Test Interval extensions under the Surveillance Frequency Control Program. The F&O closures will be performed in accordance with the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by Regulatory Guide 1.200.

(25) Adoption of 10 CFR 50.69, “Risk-Informed Categorization and treatment of structures, systems and components for nuclear power plants”

(1) TVA 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, internal fire, and seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; Internal fires and seismic hazards are evaluated with BFN specific PRA models; as specified in License Amendment No. 317.

(2) TVA shall complete the numbered items listed in Enclosure 2, List of Categorization Prerequisites, of TVA letter ML21118B079, dated April 28, 2021, prior to implementation. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than December 20, 2013, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

G. (1) During the power up rate power ascension test program and prior to exceeding 30 days of plant operation above a nominal 3293 megawatts thermal power level (100-percent OLTP) or within 30 days of satisfactory completion of steam dryer monitoring and testing that is necessary for achieving 105-percent OLTP (whichever is longer), with plant conditions stabilized at 105-percent OLTP, TVA shall trip a condensate booster pump, a condensate pump, and a main feedwater pump on an individual basis (i.e., one at a time). Following each pump trip, TVA shall confirm that plant response to the transient is as expected in accordance with previously established acceptance criteria. Evaluation of the test results for each test shall be completed and all discrepancies resolved in accordance with corrective action program requirements and the provisions of the power ascension test program.

(2) Deleted.

H. The licensee must complete the thirteen (13) Unit 1 restart commitments that are discussed in Appendix F of the license renewal application, dated December 31, 2003, as supplemented by letters dated January 31, 2005, March 2, and April 21, 2006. Completion of these activities must be met prior to power operation of Unit 1.

I. This renewed license is effective as of the date of issuance and shall expire midnight on December 20, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
J. E. Dyer

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 1 - Technical Specifications - Appendices A and B

Date of Issuance: May 4, 2006

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 340
Renewed License No. DPR-52

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 17, 2020, as supplemented by letter dated April 28, 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 Title 10 of the *Code of Federal Regulations* (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, Renewed Facility Operating License No. DPR-52 is amended by the changes indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2021.07.27 16:32:37 -04'00'

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

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: July 27, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 340

BROWNS FERRY NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace the following pages of Renewed Facility Operating License No. DPR-52 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
6f	6f
7	7

(23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

(24) TVA shall close all open Facts and Observations (F&Os) listed in Tables 11 and 13 to Attachment 2 of TVA Letter CNL-20-003, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (BFN-TS-516)," dated March 27, 2020, prior to implementing any Surveillance Test Interval extensions under the Surveillance Frequency Control Program. The F&O closures will be performed in accordance with the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by Regulatory Guide 1.200.

(25) Adoption of 10 CFR 50.69, "Risk-Informed Categorization and treatment of structures, systems and components for nuclear power plants"

(1) TVA 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, internal fire, and seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; Internal fires and seismic hazards are evaluated with BFN specific PRA models; as specified in License Amendment No. 340.

(2) TVA shall complete the numbered items listed in Enclosure 2, List of Categorization Prerequisites, of TVA letter ML21118B079, dated April 28, 2021, prior to implementation. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than June 28, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

G. This renewed license is effective as of the date of issuance and shall expire midnight on June 28, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
J. E. Dyer
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 2 - Technical Specifications - Appendices A and B

Date of Issuance: May 4, 2006

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 300
Renewed License No. DPR-68

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 17, 2020, as supplemented by letter dated April 28, 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 Title 10 of the *Code of Federal Regulations* (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, Renewed Facility Operating License No. DPR-68 is amended by the changes indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2021.07.27 16:33:06 -04'00'

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

Attachment:
Changes to the Renewed Facility Operating License

Date of Issuance: July 27, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 300

BROWNS FERRY NUCLEAR PLANT, UNIT 3

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace the following pages of Renewed Facility Operating License No. DPR-68 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
6f	6f
7	7
-	8

(16) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be complete prior to initial power ascension above 3458.

(17) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, Plant Modifications Committed, of Tennessee Valley Authority letter CNL-18-100, dated October 18, 2018; as supplemented by letter CNL-19-027, dated February 13, 2019.

(18) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

(19) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

(20) TVA shall close all open Facts and Observations (F&Os) listed in Tables 11 and 13 to Attachment 2 of TVA Letter CNL-20-003, “Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (BFN-TS-516),” dated March 27, 2020, prior to implementing any Surveillance Test Interval extensions under the Surveillance Frequency Control Program. The F&O closures will be performed in accordance with the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by Regulatory Guide 1.200.

(21) Adoption of 10 CFR 50.69, “Risk-Informed Categorization and treatment of structures, systems and components for nuclear power plants”

(1) TVA 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, internal fire, and seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening

Assessment for External Hazards, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; Internal fires and seismic hazards are evaluated with BFN specific PRA models; as specified in License Amendment No. 300.

(2) TVA shall complete the numbered items listed in Enclosure 2, List of Categorization Prerequisites, of TVA letter ML21118B079, dated April 28, 2021, prior to implementation. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than July 2, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

G. This renewed license is effective as of the date of issuance and shall expire midnight on July 2, 2036.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By
J. E. Dyer
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 3 - Technical Specifications - Appendices A and B

Date of Issuance: May 4, 2006

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 317, 340, AND 300

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-33, DPR-52, AND DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter dated July 17, 2020 (Reference 1), as supplemented by letter dated April 28, 2021 (Reference 2), the Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request for Browns Ferry Nuclear Plant, Units 1, 2, and 3 (Browns Ferry). The requested changes would add a new license condition to the Renewed Facility Operating Licenses (RFOLs) to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk informed categorization and treatment of structures, systems, and components for nuclear power reactors." The provisions of 10 CFR 50.69 allow adjustment of the scope of structures, systems, and components (SSCs) subject to special treatment requirements (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation) based on an integrated and systematic risk-informed process that includes several approaches and methods for categorizing SSCs according to their safety significance.

The supplement dated April 28, 2021, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 8, 2020 (85 FR 55507).

2.0 REGULATORY EVALUATION

2.1 Risk-Informed Categorization and Treatment of Structures, Systems, and Components

The risk-informed approach to regulation enhances and extends the traditional deterministic regulation by considering risk in a comprehensive manner. Specifically, a probabilistic approach allows consideration of a broader set of potential challenges to safety, thus providing a logical means for prioritizing these challenges based on safety significance and allowing consideration of a broader set of resources to defend against these challenges. Probabilistic risk assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures.

To take advantage of the safety enhancements available using PRA, the NRC promulgated a new regulation, 10 CFR 50.69, in the *Federal Register* on November 22, 2004 (69 FR 68008), which became effective on December 22, 2004. The provisions of 10 CFR 50.69 allow adjustment of the scope of SSCs subject to special treatment requirements. Special treatment refers to those requirements that provide increased assurance beyond normal industry practices that SSCs perform their design-basis functions. For SSCs categorized as low safety-significance (LSS), alternative treatment requirements may be implemented in accordance with the regulation. For SSCs determined to be of high safety-significance (HSS), requirements may not be changed.

Section 50.69 of 10 CFR contains requirements regarding how a licensee categorizes SSCs using a risk-informed process, adjusts treatment requirements consistent with the relative significance of the SSC, and manages the process over the lifetime of the plant. A risk-informed categorization process is employed to determine the safety-significance of SSCs and place the SSCs into one of four risk-informed safety class (RISC) categories. The determination of safety-significance is performed by an integrated decision-making process, which uses both risk insights and traditional engineering insights. The safety functions include the design-basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability and is a function of the SSC categorization results and associated bases. Finally, periodic assessment activities are conducted to adjust the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable functional requirements.

Section 50.69 of 10 CFR 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. In 2004, when promulgating the 10 CFR 50.69 rule[1], the Commission stated:

> It is important to note that this rulemaking effort, while intended to ensure that the scope of special treatment requirements imposed on SSCs is risk-informed, is not intended to allow for the elimination of SSC functional requirements or to allow equipment that is required by the deterministic design basis to be removed from the facility (i.e., changes to the design of the facility must continue to meet the current requirements governing design change; most notably § 50.59). Instead, this rulemaking should enable licensees and the staff to focus their resources on SSCs that make a significant contribution to plant safety by restructuring the regulations to allow an alternative risk-informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, this approach should allow an acceptable, though reduced, level of confidence (i.e., “reasonable confidence”) that these SSCs will satisfy functional requirements. However, continued maintenance of the health and safety of the public will depend on effective implementation of §50.69 by the licensee or applicant applying the rule at its nuclear power plant.

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

[1] Final Rule, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors, 69 FR 68008, 68011 (November 22, 2004).

risk-informed approach to treatment that provides a reasonable level of confidence that these SSCs will satisfy functional requirements. Implementation of 10 CFR 50.69 allows licensees to improve their focus on equipment that is categorized as HSS.

2.2 Licensee's Proposed Changes

The licensee proposed to amend the Browns Ferry RFOLs by adding the following license condition that would allow for the implementation of 10 CFR 50.69 (example for Unit 1):

> (25) Adoption of 10 CFR 50.69, "Risk-Informed Categorization and treatment of structures, systems and components for nuclear power plants"
>
> (1) TVA 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, internal fire, and seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the Individual Plant Examinations of External Events (IPEEE) Screening Assessment for External Hazards, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; Internal fires and seismic hazards are evaluated with BFN specific PRA models; as specified in License Amendment [XXX].
>
> (2) TVA shall complete the numbered items listed in Enclosure 2, List of Categorization Prerequisites, of TVA letter ML21118B079, dated April 28, 2021, prior to implementation. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.
>
> (3) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a shutdown defense in depth approach to a shutdown probabilistic risk assessment approach).

2.3 Regulatory Requirements and Guidance

The NRC staff considered the following regulatory requirements and guidance during its review of the proposed changes.

2.3.1 Regulatory Requirements

Section 50.69 of 10 CFR provides an alternative approach for establishing requirements for treatment of SSCs for nuclear power reactors using an integrated and systematic risk-informed approach of categorizing SSCs according to their safety significance. Specifically, 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.

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[2]

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 (i.e., it does not remove any requirements from these SSCs) for special treatment. 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, and RISC-4 SSCs are removed from the scope of any applicable special treatment requirements identified in 10 CFR 50.69(b)(1).

Section 50.69(c)(1) of 10 CFR states that SSCs must be categorized as RISC-1, RISC-2, RISC-3, or RISC-4 SSCs using a categorization process that determines if an SSC performs one or more safety-significant functions and identifies those functions. The process must:

(i) Consider results and insights from the plant-specific PRA. This PRA must, at a minimum, model severe accident scenarios resulting from internal initiating events occurring at full power operation. The PRA must be of sufficient quality and level of detail to support the categorization process and must be subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC.

(ii) Determine SSC functional importance using an integrated, systematic process for addressing initiating events (internal and external), SSCs, and plant operating modes, including those not modeled in the plant-specific PRA. The functions to be identified and considered include design bases functions and functions credited for mitigation and prevention of severe

[2] Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 9), uses the term "high-safety-significant" to refer to SSCs that perform safety-significant functions. The NRC understands HSS to have the same meaning as "safety-significant" (i.e., SSCs that are categorized as RISC-1 or RISC-2), as used in 10 CFR 50.69.

accidents. All aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience.

(iii) Maintain defense-in-depth (DID).

(iv) Include evaluations that 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 permitted by implementation of 10 CFR 50.69(b)(1) and (d)(2) are small.

(v) Be performed for entire systems and structures, not for selected components within a system or structure.

Section 50.69(c)(2) of 10 CFR states:

> The SSCs must be categorized by an Integrated Decision-Making Panel (IDP) staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering, and system engineering.

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). As stated in 10 CFR 50.69(b)(1), a licensee may voluntarily comply with 10 CFR 50.69 as an alternative to compliance with the following requirements for LSS SSCs:

(i) 10 CFR Part 21,
(ii) A portion of 10 CFR 50.46a(b),
(iii) 10 CFR 50.49,
(iv) 10 CFR 50.55(e),
(v) Specified requirements of 10 CFR 50.55a,
(vi) 10 CFR 50.65, except for section (a)(4),
(vii) 10 CFR 50.72,
(viii) 10 CFR 50.73,
(ix) Appendix B to 10 CFR Part 50,
(x) Specified requirements for containment leakage testing requirements, and
(xi) Specified requirements of Appendix A to 10 CFR Part 100.

2.3.2 Guidance

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

- RG 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," (Reference 3)

- RG 1.200, Revision 2 and 3, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Reference 4 and 5)

- RG 1.174, Revision 3, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (Reference 5)

- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision making” (Reference 7)

- 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” (Reference 8)

The Nuclear Energy Institute (NEI) issued NEI 00-04, Revision 0, “10 CFR 50.69 SSC Categorization Guideline” (Reference 9), 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 confidence that all aspects of the program reasonably reflect the current as-built, as-operated plant configuration and applicable plant and industry operational experience as required by 10 CFR 50.69(c)(1)(ii).

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed licensing basis changes meet the five key principles stated in Section C of RG 1.174, Revision 3 (Reference 5). These key principles are:

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

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

Principle 3: The proposed licensing basis change maintains sufficient safety margins.

Principle 4: When the proposed licensing basis 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 licensing basis change should be monitored using performance measurement strategies.

3.2 Traditional Engineering Evaluation

The traditional engineering evaluation below addresses the first three key principles of RG 1.174, Revision 3, which 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 Bases Change Meets the Current Regulations

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 (References 8 and 4), and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process. The NRC staff's review, as documented in this safety evaluation, used the framework provided in RG 1.174, and NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (Reference 3).

Section 2 of NEI 00-04, Revision 0, (Reference 9) 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 (Reference 1), the licensee stated that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0 (Reference 9), as endorsed in RG 1.201, Revision 1 (Reference 3).

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process that are delineated in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 0 (References 8 and 3, respectively).

Concerning the categorization of any SSCs and functions that serve as the interface between two or more systems, the licensee stated in LAR Section 3.1.14 that one of the initial steps of 10 CFR 50.69 categorization is to develop a list of systems functions. If a system is determined to include components that support functions of other systems, then support functions are created to identify the supported systems. The licensee stated that in most cases a component that supports one of these support functions cannot be fully categorized until the supporting system is entirely categorized. The licensee stated, however, that in some cases the impacts of the component on an interfacing system can be fully determined without categorizing the entire interfacing system as discussed in the NRC staff's safety evaluation supporting the issuance of license amendments Calvert Cliffs Nuclear Power Plant, Units 1 and 2 (Calvert Cliffs) that allowed for the adoption of 10 CFR 50.69 (Reference 10). LAR Section 3.1.14 (Reference 1) states that these cases occur when failure of an interfacing component cannot prevent performance of interfacing system functions or when the risk contribution from the component is limited to passive failures assessed as low safety-significant following the passive categorization process for the applicable pressure boundary segments. In Request for Additional Information (RAI) 05 (Reference 11), however, the NRC staff pointed out that the cited Calvert Cliffs approach accepted by NRC was based on meeting both limitations presented in the Browns Ferry 10 CFR 50.69 LAR statement above before the interfacing SSC could be categorized without waiting for categorization of both interfacing systems. Accordingly, the NRC staff requested clarification of this apparent inconsistency. In the response to RAI 05 (Reference 2), the licensee clarified that both of the following limitations must be true before an interfacing SSC can be categorized without categorizing the entire interfacing system:

1. Cases where an interfacing component failure cannot prevent performance of interfacing system functions, and

2. The risk is limited to passive failures assessed as low safety-significant following the passive categorization process for the applicable pressure boundary segments.

As stated by NRC staff in the Calvert Cliffs 10 CFR 50.69 safety evaluation, the passive failure classification proposed by the licensee only affects treatment programs for Class 2 and Class 3 pressure-retaining items and their associated supports. Passive failures are not normally modeled in PRAs, and the licensee's proposed passive categorization process relies on the conditional core damage and large early release probabilities following a passive failure, which are determined by imposing the impact of the passive failure on all components modeled in the PRA. This passive categorization method requires the full impact of the passive failure on safety significance to be evaluated, regardless of which system the component is assigned to.

In addition to the passive categorization method, the licensee also stated that it will perform the categorization only if it can confirm that a failure of the interface component cannot prevent performance of an interfacing system function. Finally, the licensee added these two limitations to the commitments made by the proposed licensee condition in the LAR Attachment 1, "List of Categorization Prerequisites," and provided the revised list in Enclosure 1 of its letter dated April 28, 2021 (Reference 2). The licensee commitment reads as follows:

> The following two limitations will both be true before an interfacing SSC can be categorized without categorizing the entire interfacing system.
>
> 1. Cases where an interface component failure cannot prevent performance of interface system functions, and
> 2. The risk is limited to passive failures assessed as low safety-significant following the passive categorization process for the applicable pressure boundary segments.

The NRC staff finds that the licensee's proposal, as described above, will yield the same or more conservative results when (and if) the uncategorized system is categorized, and, therefore, finds that the proposed categorization of any SSCs and functions that serve as the interface between two or more systems is acceptable.

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 (Reference 9) and clarifications in RG 1.201, Revision 1 (Reference 3), 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 the requirements in 10 CFR 50.69. As described in the subsequent sections, the NRC staff performed a more detailed review of specific steps/elements of the licensee's SSC categorization process where necessary to confirm consistency with the NEI 00-04 guidance, as endorsed. 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 (Reference 5).

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

In RG 1.174, Revision 3, the NRC identified the following considerations used for evaluating how the licensing basis change is maintained for the defense-in-depth (DID) philosophy:

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

RG 1.201, Revision 1 (Reference 3), endorses the guidance in Section 6 of NEI 00-04 (Reference 9), but notes that the containment isolation criteria in this section of the guidance are separate and distinct from those set forth in 10 CFR 50.69(b)(1)(x). The criteria in 10 CFR 50.69(b)(1)(x) are to be used in determining which containment penetrations and valves may be exempted from the Type B and Type C leakage testing requirements in both Options A and B of Appendix J to 10 CFR Part 50. The criteria provided in paragraph 50.69(b)(1)(x) of 10 CFR are not to determine the proper RISC category for containment isolation valves or penetrations.

In Section 3.1.1 of the LAR (Reference 1), the licensee clarified that it will require an SSC to be categorized as HSS based on the DID assessment performed in accordance with NEI 00-04, Revision 0 (Reference 9). Based on the above, the NRC staff concludes that the proposed change is consistent with the DID philosophy described in key principle 2 of RG 1.174, Revision 3 (Reference 6), 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 would meet the 10 CFR 50.69(c)(1)(iii) criterion that requires DID to be maintained.

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

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

Consistent with the guidance provided in NEI 00-04 for review of safety margins, and in accordance with the implementation of the SSC categorization program, the only requirements that are relaxed for LSS SSCs (includes RISC-3) are those related to treatment. The SSC's design basis functions, as described in the plants' licensing basis, including the Updated Final Safety Analysis Report and Technical Specifications Bases, do not change and should continue to be met. Similarly, there is no impact to safety analysis acceptance criteria as described in the plant licensing basis. 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 (Reference 5) and would therefore meet 10 CFR 50.69(c)(1)(iv).

3.3 Risk-Informed Assessment

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

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

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

In Sections 3.1.4, 3.1.5, and 3.1.6 of the LAR (Reference 1), the licensee described that the Browns Ferry categorization process uses PRA modeled hazards to assess risks for the internal events (includes internal flood), internal fires, and seismic events. For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Non-seismic External Hazards: Screening analysis performed for IPEEE (Reference 12) updated using criteria from Part 6 of the ASME/ANS RA Sa–2009, "Addendum A to RA S–2008, Standard for Level 1/LERF Probabilistic Risk Assessment for Nuclear Power Plant Applications," (the PRA Standard), as endorsed by the NRC.

- Other Hazards: Screening analysis performed for the IPEEE (Reference 12) updated using criteria from Part 6 of the PRA Standard, as endorsed by the NRC.

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06 (Reference 13).

- Passive Components: ANO 2 passive categorization methodology (Reference 14).

The approaches and methods proposed by the licensee to address internal events, fire, seismic, external events, other hazards, DID, and shutdown events are consistent with the approaches and methods included in the guidance in NEI 00-04, Revision 0 (Reference 9). The non-PRA method for the categorization for passive components is consistent with the ANO-2 methodology for passive components (Reference 14) approved for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and 3 moderate- and high-energy systems. The use of the ANO-2 methodology in the SSC categorization process is provided in Section 3.3.1.2 of this safety evaluation.

3.3.1.1 *Scope of the PRA*

The Browns Ferry PRAs are comprised of a full-power, Level 1, internal events PRA (IEPRA), fire PRA (FPRA), and seismic PRA (SPRA), which evaluate the CDF and LERF risk metrics. The licensee discussed in Section 3.2 of the LAR (Reference 1) that the PRA models have been assessed against RG 1.200, Revision 2 (Reference 4). Furthermore, as reviewed in subsequent sections, finding closure reviews were conducted on the identified IEPRA, FPRA and SPRA models consistent with the NRC-accepted process documented in the NEI letter to the NRC "Final Revision of Appendix X to NEI 05-04/07-12/12-16, "Close-out of Facts and Observations," dated February 21, 2017 (Reference 15). The NRC staff finds that the information provided in the LAR (Reference 1), as supplemented, supports the staff's review of PRA technical acceptability for the IEPRA (includes internal flooding), FPRA, and SPRA and therefore 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, (3) credit for diverse and flexible coping strategies (FLEX) in the PRA, and (4) assessment of PRA assumptions and sources of uncertainty.

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

In Section 3.2 of the LAR (Reference 1), the licensee stated that the IEPRA model was subjected to a full-scope peer review in August 2009 and a focused-scope in August 2015. The internal flooding PRA model was subjected to a full-scope peer review in October 2009 and a focused-scope peer review in September 2018 to address the resolutions to the finding-level F&Os from the 2009 full-scope review. The LAR, as supplemented by the licensee's response to RAI 01 in a letter dated April 28, 2021, stated that Independent Assessments for closure of the F&Os were performed in November 2018, December 2020, and January 2021, and concluded all the IEPRA (includes internal flooding) F&Os have been closed. In Section 3.1.3 of the LAR, for the IEPRA (includes internal floods), the licensee stated, in part, "there are no PRA upgrades that have not been peer reviewed." In review of the licensee's response to RAI 01 (Reference 2) discussed in a subsequent subsection in this safety evaluation, the NRC staff concluded that all F&Os were appropriately assessed by the Independent Assessment team to assure that no new methods or upgrades were inadvertently incorporated into the IEPRA without a peer review in accordance with the ASME/ANS RA-Sa-2009 PRA standard, as endorsed by the NRC.

Therefore, the NRC staff concludes that the Browns Ferry IEPRA (including internal floods) was appropriately peer reviewed, consistent with RG 1.200, Revision 2 (Reference 4), and the F&Os have been adequately dispositioned to assess the impact on the risk-informed application. Based on the above, the NRC staff finds that the Browns Ferry 10 CFR 50.69 program uses an IEPRA that is of sufficient quality to meet the requirements set forth in 10 CFR 50.69(c)(1)(i).

Internal Fire PRA Peer-Review History

As stated in LAR Section 3.2, a full-scope peer review was performed on the FPRA in May 2012 against RG 1.200, Revision 2. In June 2015, a focused-scope peer review was performed to address ASME/ANS 2009 Standard Supporting Requirements that were identified because of FPRA modeling changes made after the 2012 peer review.

In response to RAI 01.a (Reference 2), the licensee explained that in December 2020 and January 2021, the licensee conducted an Independent Assessment for closure of the finding-level F&Os and, as a result of that review, all the FPRA F&Os have been closed. Further, in response to RAI 01.d (Reference 2) the licensee committed to an implementation item to update and incorporate the most recent resolutions to the IEPRA into the FPRA as follows: "The resolutions to the internal events findings with the potential to impact the FPRA modeling will be incorporated into the FPRA."

The NRC staff has reviewed the FPRA peer review results and the licensee's resolution of the results and concludes that the Browns Ferry FPRA was appropriately peer-reviewed, consistent with RG 1.200, Revision 2 (Reference 4), and that the F&Os have been adequately dispositioned to assess the impact on the risk-informed application.

Seismic PRA Peer-Review History

In Section 3.1.6 of the enclosure to the LAR (Reference 1), the licensee stated that the proposed categorization process will use a peer-reviewed SPRA model. The NRC staff's review of the technical acceptability of the licensee's SPRA was based on the results of the peer review and the associated Independent Assessment for closure of F&Os described in LAR Section 3.2. To support its review of this LAR, the NRC staff utilized information from the licensee's submittal in response to the 10 CFR 50.54(f) information request arising from Near Term Task Force Recommendation 2.1 (Reference 16), as supplemented (Reference 17), and the corresponding staff response letter dated September 24, 2020 (Reference 18). The last full-scope peer review of the SPRA was performed in May 2019 against the SPRA requirements in ASME/ANS RA-S Case 1 (Reference 19), also known as Code Case for Standard ASME/ANS 2013 PRA standard. The NRC accepted the Code Case for use in licensing actions by letter dated March 12, 2018 (Reference 20), and subsequently, endorsed it in RG 1.200, Revision 3, dated December 2020 (Reference 5). In Section 3.2 of the LAR, the licensee stated that in November 2019, it conducted an Independent Assessment for closure of the finding-level F&Os, which concluded that all of the SPRA F&Os had been closed.

The licensee's full-scope, internal events, at-power, PRA was used as the basis for its SPRA. Therefore, the technical acceptability of the IEPRA model is an important consideration. In RAI 08.a (Reference 11), NRC staff requested clarification that resolutions to the IEPRA findings with the potential to impact the SPRA will be incorporated into SPRA. In response to RAI 08.a (Reference 2), the licensee explained that it included an item in the list of categorization prerequisites controlled by the proposed license condition. These prerequisites would need to be completed prior to the implementation of the program. The revised LAR Attachment 1 list was provided in Enclosure 1 of the licensee's letter dated April 28, 2021 (Reference 2), and states that "The resolutions to the internal events findings with the potential to impact the SPRA modeling will be incorporated into the SPRA." The NRC staff's review finds that the IEPRA is technically acceptable to use as a foundation for the SPRA because (1) the IEPRA was found to be technically acceptable for this application, as discussed above in this safety evaluation; (2) changes to the IEPRA will be propagated to the SPRA following the implementation item controlled via the proposed license condition; and (3) the staff did not identify any other issues in the IEPRA that would impact the SPRA.

Seismic PRA Evaluation

In RAI 08.b (Reference 11), the NRC staff requested a description of the approach that will be used to propagate changes made to IEPRA resulting from commitments made for this application or future routine maintenance and updates to the SPRA, given the IEPRA model is the foundation for the SPRA model.

In response to RAI 08.b (Reference 2), the licensee indicated that it maintains a "living model" that assesses the change in risk due to changes in the as-built, as-operated plant to determine when those changes should be addressed by the 10 CFR 50.69 program. The licensee stated that a commitment through the proposed license condition was added to LAR Attachment 1, List of Categorization Prerequisites, as follows, in order to establish an approach that will appropriately propagate changes in the IEPRA to the SPRA:

> TVA will assess the impact on the internal events with internal flooding “living model” with respect to the risk importance measures used to assign the safety classification (high or low) from pending model changes to be compared to previously categorized system SSCs to confirm that the criteria for LSS and HSS is still applicable, and reclassify, in accordance with NEI 00-04, (i.e., PRA model update, and at least once per two fuel cycles in a unit).

The NRC staff finds that the above-mentioned addition will appropriately identify changes to the IEPRA that impact categorization and propagate them to the SPRA.

Concerning integrated importance measures, Section 5.6 of NEI 00-04, Revision 0 (Reference 9), titled, “Integral Assessment,” discusses the need for an integrated computation using the available importance measures. The guidance further states, in part, that the “integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, and seismic models) by the fraction of the total core damage frequency [or large early release frequency] contributed by that contributor.” The guidance also provides formulas to compute the integrated Fussell-Vesely and integrated Risk Achievement Worth. Section 5.6 of the LAR states that the weighted average importance method presented in NEI 00-04, Section 1.5 will be used to integrate seismic PRA results into the overall importance measures “similar” to the approach described in the Watts Bar Nuclear Plant, Units 1 and 2 (Watts Bar) response to 10 CFR 50.69 RAI-07 for the integration of risk importance measures across all hazards. In RAI 06 (Reference 11), the NRC staff noted that SPRA basic events, such as structural failures, may often not align with basic events in other PRA models. The licensee did not state in the LAR whether the same approach will be used at Browns Ferry to integrate importance measures as is being used for the Watts Bar 10 CFR 50.69 program, particularly as it pertains to SPRA basic events. In response to RAI 06 (Reference 2), the licensee confirmed that the same approach that is being used for the Watts Bar 10 CFR 50.69 program “is applicable” to the Browns Ferry 10 CFR 50.69 to integrate importance measures.

The NRC staff found in its safety evaluation of the Watts Bar 10 CFR 50.69 LAR that Watts Bar’s approach for determining IEPRA and SPRA important measures and calculating integrated importance measures is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1. In addition, the NRC staff’s safety evaluation stated that Watts Bar will use safety significant threshold values for the SPRA that are 10 percent lower than the threshold values presented in NEI 00-04. The NRC staff accepted this approach because it is more conservative compared to using the threshold values presented in NEI 00-04 for SPRA, but did not explicitly endorse the approach. The Browns Ferry approach for integrating importance measures is the same approach that the NRC staff evaluated in the Watts Bar 10 CFR 50.69 LAR. Therefore, the NRC staff finds that the Browns Ferry approach for integrating importance measures is acceptable for the same reasons the Watts Bar approach was found to be acceptable by the NRC staff. The scope of modeled hazards for Browns Ferry includes the IEPRA (includes internal floods), FPRA, and SPRA. The NRC staff finds that the licensee’s use and treatment of importance measures is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (References 9 and 3).

Credit for FLEX Equipment

The NRC memorandum dated May 30, 2017, “Assessment of the Nuclear Energy Institute 16-06, ‘Crediting Mitigating Strategies in Risk-Informed Decision Making,’ Guidance for Risk-Informed Changes to Plants Licensing Basis” (Reference 21), provides the NRC staff’s assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200, Revision 2 (Reference 4).

In LAR Section 3.1.13, the licensee stated that the Browns Ferry PRA does “not credit portable FLEX equipment for core damage or release mitigation,” with one exception. A nitrogen bottle held by a permanently installed cart is credited in the SPRA to open an air-operated drywell vent valve given the loss of supply air. The NRC staff notes that the modelling required to credit a backup compressed gas source for an air-operated valve does not require a new method or data. Therefore, the NRC staff finds that the TVA IEPRA (includes internal floods), FPRA, and SPRA does not credit FLEX equipment for the SSC categorization process, with one exception, and that exception is appropriately modeled in the SPRA without the use of new methods or upgrades.

Identification of Key Assumptions and Sources of Uncertainty

In the LAR, the licensee confirmed that guidance in NUREG-1855, Revision 1 (Reference 7) 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.

Further, in response to RAI 02 (Reference 2), the licensee stated that it performed a comprehensive reevaluation of all assumptions and sources of PRA model uncertainty specifically for the 10 CFR 50.69 application rather than relying on the uncertainty analysis performed for the Browns Ferry Technical Specifications Task Force (TSTF)-425 LAR (Reference 22). The licensee stated that a list of assumptions and sources of uncertainty was compiled that included uncertainty associated with plant-specific features, modeling choices, and generic industry concerns. The licensee stated that for the IEPRA, FPRA and SPRA, plant-specific Browns Ferry PRA assumptions and sources of uncertainty from PRA documentation were reviewed along with generic industry uncertainty issues identified in Electric Power Research Institute (EPRI) Topical Reports (TRs) on uncertainty analysis cited by NUREG-1855 (i.e., EPRI TRs 1016737 and 1026511 (Reference 23 and 24). A disposition was developed for each identified assumption and source of uncertainty, addressing the impact of the issue on the 10 CFR 50.69 application. This included discussions for assumptions and sources of uncertainty judged not to be “key” to the application explaining why the issue does not need to be further addressed. The licensee explained that the following criteria were used to screen out PRA model assumptions or sources of uncertainty from being key to the application:

- The assumption or source of uncertainty is addressed by implementing a “consensus model” as defined in NUREG-1855 Revision 1 and EPRI TR 1013491. Assumptions and sources of uncertainty for which there is extensive historical precedent and produce results that are reasonable and realistic are screened.

- The assumption or source of uncertainty has no impact or insignificant impact on the PRA results and therefore no impact or insignificant impact on the categorization process.

- The assumption or source of uncertainty introduces a realistic conservative bias in the PRA model results based on the discussion of "realistic conservatism" in EPRI TR 1013491. Assumptions that introduce realistic (slight) conservatism are screened.

- There is no reasonable alternative assumption or reasonable modeling refinement that would change the risk profile of the plant based on discussion of "reasonable alternative assumptions" in RG 1.200, Revision 2.

- The potential conservatism would result in assigning a component HSS, when in it could be assigned LSS.

The licensee explained that a qualitative discussion was sufficient to demonstrate that each identified assumption or source of uncertainty met one or more of the above screening criteria and therefore, was not key. However, in some cases, the qualitative evaluation was complemented by insights from PRA results to help screen the assumption or source of uncertainty, based on insignificant impact on CDF and LERF, or importance measures. The licensee stated that based on its review it was determined there are no assumptions or sources of uncertainty that qualify as "key" to this application. Accordingly, no additional sensitivity cases are required. The licensee also clarified that the content of LAR Attachment 6 is superseded by the response to RAI 02 (Reference 2).

The NRC staff also reviewed whether the state of knowledge correlation (SOKC) was addressed by the licensee. RG 1.174, Revision 3 (Reference 6) and Section 6.4 of NUREG-1855, Revision 1, for a Capability Category II risk evaluation, indicate that the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the PRA input parameters and model uncertainties in quantification of the PRA models. In general, the point estimate CDF and LERF obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF/LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the SOKC is unimportant (i.e., the risk results are well below the acceptance guidelines). In RAI 04 (Reference 2), the NRC staff noted the LAR does not stipulate whether the total CDF and LERF values presented in LAR Attachment 2 are mean values and noted the small margin between the LERF for Units 1 and 2 of 9.3E-06 and 9.4E-06 per year, respectively, and the RG 1.174, Revision 3 LERF threshold of 1E-05 per year. Accordingly, the risk increase due to consideration of the SOKC could impact the conclusions of the NEI 00-04 Section 8 overall sensitivity study results by increasing the Browns Ferry LERF values above 1E-05 per year. In response to RAI 04.a (Reference 2), the licensee stated that the IEPRA and FPRA include SOKC. In response to RAI 04.b (Reference 2), concerning the FPRA, the licensee explained that besides the component failure types, uncertainty propagation is performed for fire ignition frequencies, non-suppression probabilities, severity factors, hot short probabilities, and human failures. The licensee summarized how the parametric uncertainty analysis was performed for each parameter so that the impact of the SOKC could be considered for correlated inputs.

The licensee further explained that the existing FPRA is based on the IEPRA prior to F&O closure and it will be updated to account for changes to the IEPRA during the F&O closure process. Therefore, this licensee added a commitment through the proposed licensee condition to evaluate the total CDF and LERF against the RG 1.174 risk acceptance guidelines after the FPRA and SPRA are updated to incorporate IEPRA F&O resolutions using mean values and consideration of the SOKC. The revised list of prerequisites states:

> The resolutions to the internal events findings with the potential to impact the FPRA and SPRA modeling will be incorporated into the FPRA and SPRA. Following the model updates, the values of total CDF and total LERF for each unit will be evaluated for conformance to the Regulatory Guide 1.174 risk acceptance guidance. The SOKC will be evaluated for importance by assessing the mean risk results relative to acceptance guidelines.

The NRC staff finds that the assessment performed to identify the key assumptions and sources of uncertainty, and to address SOKC, is consistent with the guidance provided in NUREG-1855, Revision 1 and is acceptable for this application.

<u>PRA Acceptability Conclusions</u>

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, FPRA, and SPRA 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 (Reference 4) provides guidance for determining the acceptability of the PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer-review process.

The licensee has subjected the IEPRA, FPRA, and SPRA to the peer-review processes and submitted the results of the peer review. The NRC staff reviewed the peer-review history, the results of the peer review, 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, FPRA, and SPRA, with the commitments made in the license condition to update the FPRA and SPRA prior to implementation, will be 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 reviewed consistent with the guidance in RG 1.200, Revision 2 and NUREG-1855 (References 3, 4, and 7, respectively), as applicable, and addressed appropriately for this application.

The NRC staff finds the licensee provided the required information, and the IEPRA (includes internal floods) FPRA, and SPRA with commitments made in the license condition to update the FPRA and SPRA before implementation of 10 CFR 50.69 are acceptable and therefore meets the requirements set forth in paragraphs 50.69(c)(1)(i) and (ii) of 10 CFR.

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

<u>Non-Seismic External Hazards and Other Hazards</u>

This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation, and nearby facility accidents, and other hazards.

In the safety evaluation report for the Browns Ferry IPEEE for Units 1, 2 and 3, the NRC staff states, in part, "[t]he high winds, floods, transportation and other external events (HFO) areas were eliminated based on either compliance with 1975 NRC Standard Review Plan (SRP) criteria or on the basis of a bounding probabilistic assessment resulting in a CDF estimate less than 1E-6 per reactor year, i.e., below the NUREG-1407 [(Reference 25)] screening criterion." In Section 3.1.7 of the LAR (Reference 1), the licensee stated, in part, that all other external hazards (i.e., not seismic or fire hazards) were screened from applicability to Browns Ferry per a plant-specific evaluation in accordance with GL 88-20 and updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009. In Attachment 4 of the LAR (Reference 1), the licensee provided the results of the plant-specific evaluation that assessed the IPEEE results to using endorsed criteria in the ASME/ANS RA-Sa-2009 PRA Standard and current plant hazard information.

In LAR Section 3.1.7, the licensee stated that no SSCs were explicitly credited to allow a scenario to screen, and therefore, "[s]creened hazards are considered insignificant for every SSC and, therefore, will not be considered during the categorization process." In RAI 07.a (Reference 11), NRC staff stated that it appeared, based on the cited statement from the LAR, that at the time an SSC is categorized, it will not be evaluated using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard because that evaluation has already been made. The NRC staff noted that plant changes, plant or industry operational experience, and identified errors or limitations in the PRA models could potentially impact the conclusion that the SSC is not needed to screen an external hazard. Therefore, the NRC staff requested clarification of whether or not an SSC will be evaluated during categorization of the SSC using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard and, if not, to provide justification for this approach. In response to RAI 07.a (Reference 2), the licensee confirmed that SSCs will be evaluated during categorization using the guidance in NEI 00-04, Figure 5-6 to ensure that the SSCs are not credited in screening an external hazard.

In RAI 07.b (Reference 11), the NRC staff noted that in a staff assessment of the Browns Ferry flood hazard mitigating strategy assessment (MSA) dated September 5, 2017 (Reference 26), contrary to the statement in LAR Section 3.1.7, SSCs would be relied upon to mitigate the impact of an extreme flooding event such as Local Intense Precipitation (LIP). The report includes discussion of credit for passive features and active equipment such as FLEX equipment and sump pumps to protect against LIP. The NRC staff noted that these SSCs appear to be credited in screening the external flooding hazard. Therefore, the NRC staff requested identification of any active or passive SSCs credited for screening external flooding and discussion of how they would be treated in the proposed risk-informed categorization process. In response to RAI 07.b (Reference 2), the licensee explained it has evaluated the external flooding hazard for Browns Ferry and concluded that the only scenario with the potential to impact plant equipment is a LIP event. The licensee evaluated each plant location where the maximum water level associated with the LIP could impact plant equipment and identified the following SSCs credited to mitigate LIP:

- Reactor Building Airlock Doors
- Diesel Generator Building Watertight Doors
- Intake Pumping Station Watertight Doors
- Intake Pumping Station Rain Water Sump Pumps
- Radwaste Building Watertight Doors

The licensee stated that these SSCs are "assigned a HSS classification, hence, RISC-1 or RISC-2, if classified." Additionally, the licensee stated that FLEX equipment is outside the scope of SSCs eligible to be categorized under 10 CFR 50.69 at Browns Ferry.

In RAI 07.c (Reference 11), the NRC staff requested identification of any active or passive SSCs credited for screening high winds and the tornado hazard, including tornado-generated missiles, and discussion of how they would be treated in the proposed risk-informed categorization process. In response to RAI 07.c (Reference 2), the licensee indicated that there are no active or passive SSCs identified [at this time] for screening high winds for tornado hazards, but that the TVA 10 CFR 50.69 categorization process requires the assessment of a full scope of hazards, including other external risks (e.g., tornadoes, external flood, etc.)

The NRC staff finds that TVA's categorization process is consistent with Figure 5-6 of NEI 00-04, as endorsed in RG 1.201, Revision 1, for consideration of non-seismic external hazards and other hazards.

In summary, the use of the Browns Ferry IPEEE results described by the licensee in the LAR (Reference 1), supplemental information provided in response to RAI 07 (Reference 2), and the licensee's assessment of the other external hazards (i.e., high winds, tornadoes, and external flood) is consistent with Section 5 of NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (References 8 and 3, respectively). The NRC staff concludes that the licensee's treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

Component Safety Significance Assessment for Passive Components

In Section 3.1.2 of the LAR (Reference 1), 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 (References 8 and 3, respectively), for passive component categorization. This method was approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 14). 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" (Reference 27). The ANO-2 methodology relies on the conditional core damage and large early release probabilities associated with pipe ruptures. Safety significance is generally measured by the frequency and the consequence of, in this case, pipe ruptures. Treatment requirements (including repair/replacement) only affect the frequency of passive component failure. Categorizing solely based on consequences, which measures the safety significance of the pipe given that it ruptures, is conservative compared to including the rupture frequency in the categorization. The categorization will not be affected by changes in frequency arising from changes to the treatment. Therefore, the NRC staff finds that the use of the 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 (Reference 1), the licensee stated, "[t]he passive categorization process is intended to apply the same risk-informed process accepted in the ANO 2-R&R-004 for the passive categorization of Class 2, 3, and non-class components." The licensee stated that "[a]ll ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP." 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*

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

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

NEI 00-04, Revision 0 provides guidance that includes programmatic configuration control and a periodic review to ensure that all aspects of the 10 CFR 50.69 program (i.e., includes traditional engineering analyses) and PRA models used to perform the risk assessment continue to reflect the as-built-as-operated plant and that plant modifications and updates to the PRA over time are continually incorporated. Sections 11 and 12 of NEI 00-04, Revision 0, include discussion on periodic review and program documentation and change control. Maintaining change control and periodic review will also maintain confidence that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs continually reflect the Browns Ferry as-built, as-operated plant.

The licensee stated that the decision to update the PRA models for the 10 CFR 50.69 program is covered by procedures and presented to the IDP for concurrence.

The NRC staff finds the risk management process described by the licensee in the LAR (Reference 1) as supplemented in a letter dated April 28, 2021 (Reference 2) to assess the impact of model changes on the application is consistent with Section 12 of NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1, and consistent with the requirements in 10 CFR 50.69(e). Based on the above, the NRC staff has determined that the proposed change satisfies the fifth key principle for risk informed decision making prescribed in RG 1.174, Revision 3.

4.0 CHANGES TO THE RENEWED FACILITY OPERATING LICENSES

Based on the NRC staff's review of the LAR and the licensee's responses to the staff's RAIs, the staff identified specific actions, as described below, that are necessary to 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 (Reference 3), and NEI 00-04, Revision 0 (Reference 9).

The NRC staff's finding on the acceptability of the PRA evaluation in the licensee's proposed 10 CFR 50.69 process is conditioned upon the License Condition provided below that delineates completion of the implementation items and list of prerequisites to address changes to the PRA model or documentation. The NRC staff finds that the clarifications to the NEI 00-04, Revision 0 guidance (Reference 9) and other changes that were described by the licensee are routine and will be systematically addressed through the configuration management and control and periodic update processes as described in Section 3.3.2 of this safety evaluation. In response to RAI 01 (Reference 2), the licensee proposed the following amendment to the RFOLs for Browns Ferry. The proposed license condition states (example for Unit 1):

> (25) Adoption of 10 CFR 50.69, "Risk-Informed Categorization and treatment of structures, systems and components for nuclear power plants"

(1) TVA 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, internal fire, and seismic risk; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; and the results of non-PRA evaluations that are based on the Individual Plant Examinations of External Events (IPEEE) Screening Assessment for External Hazards, and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009; Internal fires and seismic hazards are evaluated with BFN specific PRA models; as specified in License Amendment [XXX].

(2) TVA shall complete the numbered items listed in Enclosure 2, List of Categorization Prerequisites, of TVA letter ML21118B079, dated April 28, 2021, prior to implementation. All issues identified in the enclosure will be addressed and any associated changes will be made, focused scope peer reviews will be performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and any findings will be resolved and reflected in the PRA of record prior to implementation of the 10 CFR 50.69 categorization process.

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

The NRC staff finds that the proposed license condition and referenced implementation items are acceptable because they adequately implement 10 CFR 50.69 using models, methods, and approaches consistent with the applicable guidance that has previously been endorsed by the NRC. The NRC staff, through an onsite audit or during future inspections, may choose to examine the closure of the implementation items with the expectation that any variations discovered during this review, or concerns with regard to adequate completion of the implementation item, will be tracked and dispositioned appropriately in accordance with the requirements of 10 CFR 50.69(f) and 10 CFR Part 50, Appendix B, Criterion VI, and could be subject to NRC enforcement action(s).

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendments on June 16, 2021. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The NRC has previously issued a proposed finding that the amendments involve no significant hazards consideration in the *Federal Register* on September 8, 2020 (85 FR 55507), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

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Date: July 27, 2021

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 – ISSUANCE OF AMENDMENT NOS. 317, 340, AND 300 REGARDING ADOPTION OF TITLE 10 OF THE CODE OF FEDERAL REGULATIONS SECTION 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR NUCLEAR POWER PLANTS" (EPID L-2020-LLA-0162) DATED JULY 27, 2021.

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OFFICE	NRR/DRA/APLC/BC	NRR/DNRL/NPHP/BC*	NRR/DNRL/NVIB/BC*
NAME	SRosenberg	MMitchell	ABuford
DATE	6/21/2021	6/30/2021	6/23/2021
OFFICE	NRR/DSS/SNSB/BC*	NRR/DSS/SCPB/BC*	NRR/DEX/EICB/BC*
NAME	SKrepel	BWittick (HWagage for)	MWaters
DATE	6/25/2021	6/25/2021	6/25/2021
OFFICE	NRR/DEX/EEEB/BC(A)*	NRR/DEX/EMIB/BC(A)*	OGC – NLO*
NAME	SRay	ITseng	KGamin
DATE	6/29/2021	6/30/2021	07/21/2021
OFFICE	NRR/DORL/LPLII-2/BC	NRR/DORL/LPLII-2/PM	
NAME	DWrona	MWentzel	
DATE	7/27/2021	7/27/2021	

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