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

April 30, 2020

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: WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 134 AND 38 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-2018-LLA-0493)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 134 to Facility Operating License No. NPF-90 and Amendment No. 38 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated November 29, 2018, as supplemented by letters dated July 15, July 29, and October 18, 2019.

The amendments revise the Watts Bar Nuclear Plant, Units 1 and 2, Facility Operating Licenses to add a new license condition 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."

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

Sincerely,

/RA/

Kimberly J. Green, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-390 and 50-391

Enclosures:
1. Amendment No. 134 to NPF-90
2. Amendment No. 38 to NPF-96
3. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. NPF-90

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

A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated November 29, 2018, as supplemented by letters dated July 15, July 29, and October 18, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to Facility Operating License No. NPF-90 as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Undine S. Shoop

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Date: 2020.04.30 13:15:53 -04'00'

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License

Date of Issuance: April 30, 2020

ATTACHMENT TO AMENDMENT NO. 134

WATTS BAR NUCLEAR PLANT, UNIT 1

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Replace page 4b of Facility Operating License No. NPF-90 with the attached revised page 4b.

(10) By May 31, 2018, TVA shall ensure that a listing organization acceptable to the NRC (as the Authority Having Jurisdiction) determines that the fire detection monitoring panel in the main control room either meets the appropriate designated standards or has been tested and found suitable for the specified purpose.

(11) The licensee shall replace the WBN, Unit 1 upper compartment cooler cooling coils with safety-related cooling coils to eliminate a potential source of containment sump dilution during design basis events prior to increasing the number of Tritium Producing Burnable Absorber Rods (TPBARs) loaded in the reactor core above 704.

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

(a) 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, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards; fire hazards by use of the fire protection program (FPP) safe shutdown equipment list (SSEL), and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009, as specified in Unit 1 License Amendment [Number].

(b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 2, Attachment 1, "List of Categorization Prerequisites," to TVA letter CNL-19-108, “Response to NRC Second Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, ‘Risk-informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors’ (WBN-TS-17-24) (EPID 2018-LLA-0493),” dated October 28, 2019.

(13) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from using the FPP SSEL approach to an internal fire probabilistic risk assessment approach).

D. The following exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Therefore, these exemptions are granted pursuant to 10 CFR 50.12.

(1) Deleted

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

WATTS BAR NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. NPF-96

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

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated November 29, 2018, as supplemented by letters dated July 15, July 29, and October 18, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to Facility Operating License No. NPF-96 as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Undine S. Shoop

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Date: 2020.04.30 13:16:50 -04'00'

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License

Date of Issuance: April 30, 2020

ATTACHMENT TO AMENDMENT NO. 38

WATTS BAR NUCLEAR PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. NPF-96

DOCKET NO. 50-391

Replace pages 4 and 5 of Facility Operating License No. NPF-96 with the attached revised pages 4 and 5.

TVA may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(9) By May 31, 2018, TVA shall report that a listing organization acceptable to the NRC (as the Authority Having Jurisdiction) has determined that the fire detection monitoring panel in the main control room either meets the appropriate designated standards or has been tested and found suitable for the specified purpose.

(10) TVA will verify for each core reload that the actions taken if $F_Q^W(Z)$ is not within limits will assure that the limits on core power peaking $F_Q(Z)$ remain below the initial total peaking factor assumed in the accident analyses.

(11) TVA will implement the compensatory measures described in Section 3.4, “Additional Compensatory Measures,” of TVA Letter CNL-18-012, dated January 17, 2018, during the timeframe the temperature indicator for RCS hot let 3 is not required to be operable for the remainder of Cycle 2. If the RCS hot leg 3 temperature indicator is returned to operable status prior to the end of Cycle 2, then these compensatory measures are no longer required.

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

(a) 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, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards; fire hazards by use of the fire protection program (FPP) safe shutdown equipment list (SSEL), and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009, as specified in Unit 2 License Amendment [Number].

(b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 2, Attachment 1, "List of Categorization Prerequisites " to TVA letter CNL-19-108, “Response to NRC Second Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, ‘Riskinformed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors’ (WBN-TS-17-24) (EPID L-2018-LLA-0493),” dated October 28, 2019.

(13) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from using the FPP SSEL approach to an internal fire probabilistic risk assessment approach).

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

E. This license is effective as of the date of issuance and shall expire at midnight on October 21, 2055.

FOR THE NUCLEAR REGULATORY COMMISSION

William M. Dean, Director
Office of Nuclear Reactor Regulation

Appendices:
1. Appendix A –
   Technical Specifications
2. Appendix B –
   Environmental Protection Plan

Date of Issuance: October 22, 2015

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 134 AND 38

TO FACILITY OPERATING LICENSE NOS. NPF-90 AND NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-390 AND 50-391

1.0 INTRODUCTION

By letter dated November 29, 2018 (Reference 1), as supplemented by letters dated July 15, 2019, (Reference 2), July 29, 2019 (Reference 3), and October 18, 2019 (Reference 4), Tennessee Valley Authority (TVA, the licensee), submitted a license amendment request (LAR) for the Watts Bar Nuclear Plant (Watts Bar or WBN), Units 1 and 2. The licensee proposed to add a new license condition to the TVA Facility Operating Licenses (FOLs) 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[1] and methods for categorizing SSCs according to their safety significance.

In email correspondence dated June 18 and September 13, 2019 (References 5 and 6, respectively) the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff requested additional information (RAI) from the licensee. By letters dated July 15, July 29, and October 28, 2019, the licensee responded to these requests (References 2, 3 and 4). The supplements provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 30, 2019 (84 FR 36969).

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[1] Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (Reference 7), describes the SSC categorization process in its entirety as an overarching approach that includes multiple approaches and methods identified for a probabilistic risk assessment (PRA) hazard and non-PRA methods.

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[2], 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

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

> 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 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 <u>Licensee's Proposed Changes</u>

In a letter dated October 28, 2019 (Reference 4), and in response to Division of Risk Assessment (DRA) RAI 01.b, the licensee proposed to amend the Watts Bar FOLs by adding the following license condition (example shown for Unit 1) that would allow for the implementation of 10 CFR 50.69:

> (12) Adoption of 10 CFR 50.69, "Risk-Informed categorization and treatment of structures, systems and components for nuclear power plants"
>
> > (a) 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, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards; fire hazards by use of the fire protection program (FPP) safe shutdown equipment list (SSEL), and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009, as specified in Unit 1 License Amendment [Number].
> >
> > (b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 2, Attachment 1, "List of Categorization Prerequisites," to TVA letter CNL-19-108, "Response to NRC Second Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems and

Components for Nuclear Power Reactors' (WBN-TS-17-24) (EPID L-2018-LLA-0493)," dated October 28, 2019.

(13) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from using the FPP SSEL approach to an internal fire 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[3]

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).

[3] Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005 (Reference 8), 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.

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 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

Nuclear Energy Institute (NEI) 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 8), describes a process for determining the safety-significance of SSCs and categorizing them into the four RISC categories defined in 10 CFR 50.69. This categorization process is an integrated decision-making process that incorporates risk and traditional engineering insights. NEI 00-04, Revision 0, provides options for licensees implementing different approaches depending on the scope of their PRA models. It also allows the use of non-PRA approaches when PRAs have not been developed. NEI 00-04 identifies non-PRA methods to be used as an approach, such as fire-induced vulnerability evaluation (FIVE) to address fire risk, seismic margin analysis (SMA) to address seismic risk, and guidance in Nuclear Management and Resource Council (NUMARC) 91-06, "Guidelines for Industry Actions to Assess Shutdown Management" (Reference 9), to address shutdown operations. As stated in Regulatory Guide (RG) 1.201, Revision 1, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," May 2006 (Reference 7), such non-PRA-type evaluations (e.g., fire-induced vulnerability evaluation and seismic margins analysis) will result in more conservative categorization, in that, special treatment requirements will not be allowed to be relaxed for SSCs that are relied upon in such evaluations. The degree of relief that the NRC will accept under 10 CFR 50.69 (i.e., SSCs subject to relaxation of special treatment requirements) will be commensurate with the assurance provided by the evaluations performed to assess and characterize the SSC's risk.

Sections 2 through 10 of NEI 00-04 describe steps/elements of the SSC categorization process to be performed for meeting the requirements of 10 CFR 50.69(c), as follows:

- 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(e), and Section 12 of NEI 00-04 provides guidance on periodic review related to the requirements in 10 CFR 50.69(f). 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).

RG 1.201, Revision 1 (Reference 7), endorses the categorization method described in NEI 00-04, Revision 0 (Reference 8), with clarifications, limitations, and conditions. RG 1.201, Revision 1, states that the applicant is expected to document, at a minimum, the technical adequacy of the internal initiating events PRA. Licensees may use either PRAs or alternative approaches for hazards other than internal initiating events. RG 1.201, Revision 1, clarifies that the NRC staff expects that licensees proposing to use non-PRA approaches in their categorization should provide a basis in the submittal for why the approach, and the accompanying method employed to assigned safety significance to SSCs, are technically acceptable. The guidance further states that as part of the NRC's review and approval of a licensee's or applicant's application requesting to adopt 10 CFR 50.69, the NRC staff intends to impose a license condition that will explicitly address the scope of the PRA and non-PRA methods used in the licensee's categorization approach. If a licensee or applicant wishes to change its categorization approach and the change is outside the bounds of the NRC's license condition (e.g., switch from a seismic margins analysis to a seismic PRA), the licensee or applicant will need to seek NRC approval, via a license amendment, for the implementation of the new approach in its categorization process. In addition, RG 1.201, Revision 1, states that all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv).

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Reference 10) describes an acceptable approach for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors. RG 1.200 endorses, with clarifications, the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009 (Reference 11). Revision 2 of RG 1.200 provides guidance for determining the acceptability of a PRA by comparing the PRA to the relevant parts of the ASME/ANS RA-Sa-2009 using a peer review process. The guidance discusses the need to perform peer reviews for PRA upgrades. A PRA upgrade is defined in ASME/ANS RA-Sa-2009 as "the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences."

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 12), provides guidance on the use of PRA findings and risk insights in support of changes to a plant's licensing basis. This RG provides risk acceptance guidelines for evaluating the results of such evaluations.

NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-Making" (Reference 13), provides guidance on how to treat uncertainties associated with PRA in risk-informed decision-making. The guidance fosters an understanding of the uncertainties associated with PRA and their impact on the results of the PRA and provides a pragmatic approach to addressing these uncertainties in the context of the decision-making.

3.0 TECHNICAL EVALUATION

3.1 Method of NRC Staff Review

The NRC staff reviewed the licensee's 10 CFR 50.69 categorization process against the categorization process described in NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7), and the acceptability of the licensee's PRA for use in the application of the 10 CFR 50.69 categorization process.

3.2 Overview of the Categorization Process (NEI 00-04, Revision 0, Section 2)

The guidance in RG 1.201 provides that the categorization process described in NEI 00-04, with any noted exceptions or clarifications, is acceptable for implementation of 10 CFR 50.69. Section 2 of NEI 00-04, Revision 0, in part, states that the categorization process includes eight primary steps:

1. Assembly of Plant-Specific Inputs (Section 3 of NEI 00-04, Revision 0)
2. System Engineering Assessment (Section 4 of NEI 00-04, Revision 0)
3. Component Safety Significance Assessment (Section 5 of NEI 00-04, Revision 0)
4. Defense-In-Depth Assessment (Section 6 of NEI 00-04, Revision 0)
5. Preliminary Engineering Categorization of Functions (Section 7 of NEI 00-04, Revision 0)
6. Risk Sensitivity Study (Section 8 of NEI 00-04, Revision 0)
7. IDP Review and Approval (Section 9 of NEI 00-04, Revision 0)
8. SSC Categorization (Section 10 of NEI 00-04, Revision 0)

The licensee stated in Section 3.1.1 of the LAR (Reference 1) that it will implement the risk-informed categorization process in accordance with NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7).

The licensee provided further discussion of specific elements within the 10 CFR 50.69 categorization process. Elements of the categorization process for which the licensee provided clarity in the LAR are bulleted below. A more detailed review of those specific elements in the categorization process is discussed in the applicable sections of this SE.

- Passive Characterization:

  Passive components are not modeled in the PRA. Therefore, a different method to perform the assessment is used to assess the safety significance of these components, as described in Section 3.5.3.4 of this SE. The process used addresses those components that have only a pressure retaining function and the passive function of active components, such as the pressure/liquid retention of the body of a motor-operated valve.

- Qualitative Characterization:

  System functions are qualitatively categorized as HSS or LSS based on the seven questions in Section 9.2 of NEI 00-04, Revision 0 (Refer to Section 3.5.6 of this SE).

- Cumulative Risk Sensitivity Study:

  For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of LSS components, identified in all PRA models, results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174, Revision 3 (Reference 12), (Refer to Section 3.5.5 of this SE).

- Review by the IDP:

  The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components (Refer to Section 3.5.6 of this SE).

- Use of Fire Safe Shutdown Equipment List (FSSEL):

  Watts Bar has proposed the use of the FSSEL to assess the fire risk. The use of this method is a deviation from the approaches and methodologies described in NEI 00-04, Revision 0 (Reference 8), (Refer to Section 3.5.3.1 of this SE).

Attachment 1 of Enclosure 1 to the licensee's letter dated July 15, 2019 (Reference 2), provides a list of steps/elements that the licensee will include in the 10 CFR 50.69 programmatic procedures prior to the use of the categorization process. A more detailed review of the steps/elements for the programmatic procedures is discussed in Section 3.5 of this SE.

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

Section 50.69(c)(1)(ii) of 10 CFR requires licensees to determine SSC functional importance using an integrated, systematic process for addressing initiating events (i.e., 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-basis functions and functions credited for mitigation and prevention of severe accidents.

Section 3 of NEI 00-04, Revision 0 (Reference 8), states that the assembly of plant-specific inputs involves the collection and assessment of the key inputs to the risk-informed categorization process. This includes design and licensing information, PRA analyses, and other relevant plant data sources. This step also includes the critical evaluation of plant-specific risk information to ensure that the PRA is modeled adequately to support the risk characterization of the SSCs for this application. In addition, Section 4 of the NEI 00-04 guidance states, in part, "that the next step is the identification of system functions, including design basis and beyond-design-basis functions identified in the PRA, and that system functions should be consistent with the functions defined in design basis documentation and maintenance rule functions."

Furthermore, the guidance in Section 3 of NEI 00-04, Revision 0, summarizes the use of PRA, if such PRA models exist, or, in the absence of a quantifiable PRA, the use of other methods to evaluate risk including the FIVE methodology, SMA, IPEEE Screening, and Shutdown Safety Plan. The NRC staff acknowledges that elements of the categorization process are not always performed in chronological order and may be performed in parallel, such that, the systematic process for evaluating the plant-specific PRA may include other aspects of the categorization process (e.g., system selection, system boundary definition, identification of system functions,

and mapping of components to functions). The licensee's risk categorization process uses PRAs to assess risks from the internal events PRA (IEPRA) (includes internal floods) and for seismic events.

For the other applicable risk hazard groups, the licensee's process uses non-PRA methods for risk characterization, such as the Watts Bar FSSEL to assess fire risk, and guidance provided in NEI 00-04 to assess the risk from external hazards (e.g., high winds, external floods) and other hazards.[4] To assess risk from shutdown operations, the licensee's categorization process uses the Shutdown Safety Management Plan. The NRC staff review of the quality and level of detail for the acceptability of the IEPRA at the time of the submittal and non-PRA methods is provided in Sections 3.5.1 through 3.5.3 of this SE.

Section 50.69(c)(1)(v) of 10 CFR requires that SSC categorization be performed for entire systems and structures, not for selected components within a system or structure. The NRC staff finds the process described in the LAR, as supplemented by letters dated July 15, 2019 (Reference 2), July 29, 2019 (Reference 3), and October 18, 2019 (Reference 4), is consistent with NEI 00-04, as endorsed by the staff in RG 1.201, Revision 1, and is capable of collecting and organizing information at the system level for defining boundaries, functions, and components.

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

Section 50.69(c)(1)(ii) of 10 CFR requires, in part, the functions to be identified and considered in the categorization process include design basis functions and functions credited for mitigation and prevention of severe accidents. Revision 0 of NEI 00-04 (Reference 8), includes guidance to identify all functions performed by each system and states that the IDP will categorize all system functions. All system functions include all functions involved in the prevention and mitigation of accidents and may include additional functions not credited as hazard mitigating functions depending on the system.

The guidance in NEI 00-04 also includes consideration of interfacing functions. Section 4 of NEI 00-04 provides guidance for circumstances when the categorization of a candidate low safety-significant SSC within the scope of the system being considered cannot be completed because it also supports an interfacing system. The guidance states, in part, "[i]n this case, the SSC will remain uncategorized until the interfacing system is considered […]. Therefore, the SSC will remain uncategorized and continue to receive its current level of treatment requirements." Furthermore, Section 7.1 of the NEI 00-04 guidance states, in part, "[d]ue to the overlap of functions and components, a significant number of components support multiple functions. In this case, the SSC or part thereof should be assigned the highest risk significance for any function that the SSC or part thereof supports."

In Section 2.2 of the LAR (Reference 1), the licensee stated "[t]he 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." Section 3.1.1 of the LAR summarizes the different hazards and plant states for which functional and risk significant information will be collected. Furthermore, in Section 3.1.1 of the

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[4] Other hazards include any internal or external hazards that are not considered as part of the development of an internal events, internal flood, internal fire, seismic, high wind, or external flood PRA using the applicable parts of the ASME/ANS PRA standard, as endorsed by the NRC.

LAR, the licensee confirmed that the SSC categorization process documentation will include, among other items, system functions, identified and categorized with the associated bases, and mapping of components to support function(s).

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

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

This step in the licensee's categorization process assesses the safety significance of components using quantitative or qualitative risk information from a modeled PRA hazard, other hazards that can be screened, and non-PRA method(s). In the NEI 00-04 guidance (Reference 8), component risk significance is assessed separately for the following hazard groups:

- Internal Events (includes internal floods)
- Internal Fire Events[5]
- Seismic Events
- External Hazards (e.g., high winds, external floods)
- Other Hazards
- Shutdown Events
- Passive Component Categorization[6]

Section 50.69(c)(1)(i) of 10 CFR requires, in part, the use of PRA to assess risk from internal events as a minimum. This section of the rule further specifies that the PRA used in the categorization process must be of sufficient quality and level of detail and subject to an acceptable peer review process. Section 50.69(b)(2) of 10 CFR allows, and the guidance in NEI 00-04, Revision 0 (Reference 8) summarizes, the use of PRA, if such PRA models exist, or, in the absence of quantifiable PRA, the use of other methods (e.g., FIVE, SMA, IPEEE screening, and shutdown safety management plan).

In Sections 3.1.1 and 3.2.1 of the LAR (Reference 1), the licensee stated that the Watts Bar categorization process uses PRA modeled hazards to assess risks for the internal events (includes internal flood) and a seismic PRA (SPRA). For the other risk contributors, the licensee's process uses the following non-PRA methods to characterize the risk:

- Internal Fire Events: Living FSSEL performed for the Watts Bar Fire Protection program.

- External Hazards: Screening analysis performed for IPEEE (Reference 14), updated using criteria from Part 6 of the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by the NRC.

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[5] Deviation from RG 1.201: The methodology proposed for use of the FSSEL to assess the risk for internal fire events was not endorsed by the NRC in RG 1.201, Revision 1 (Reference 7).

[6] Deviation from RG 1.201: The methodology proposed for the categorization of passive components was not cited in NEI 00-04, Revision 0 (Reference 8), or RG 1.201, Revision 1 (Reference 7), but as approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) for issuance of a different license amendment (Reference 15).

- Other Hazards: Screening analysis performed for the IPEEE (Reference 14), updated using criteria from Part 6 of the ASME/ANS RA-Sa-2009 PRA Standard, as endorsed by the NRC.

- Shutdown Events: Safe Shutdown Risk Management program consistent with NUMARC 91-06 (Reference 9).

- Passive Components: ANO-2 passive categorization methodology (Reference 15).

The approaches and methods proposed by the licensee to address internal events, 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 8), as endorsed in RG 1.201, Revision 1 (Reference 7). The non-PRA method for the categorization of passive components is consistent with the ANO-2 methodology for passive components (Reference 15), approved for risk-informed safety classification and treatment for repair/replacement activities in Class 2 and 3 moderate and high energy systems. A detailed staff review of the use of the ANO-2 method in the SSC categorization process is provided in Section 3.5.3.5 of this SE. To address internal fire events, the licensee proposed to use an alternative method not specified in the NEI 00-04 guidance as endorsed by the NRC (Reference 8). A detailed staff review of the licensee's proposed approach for the use of the FSSEL is provided in Section 3.5.3.1 of this SE.

3.5.1 Evaluation of PRA Acceptability to Support the SSC Categorization Process

Consistent with Section 5 of NEI 00-04, Revision 0 (Reference 8), the component safety significance assessment must include evaluations for each of the hazards: (1) internal events hazard, (2) internal fire hazard, (3) seismic hazard, (4) other hazards (e.g., high wind, external floods, etc.), and (5) shutdown events.

Section 50.69(c)(i) of 10 CFR requires, in part, that a licensee's 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. RG 1.200 provides guidance for determining the technical adequacy of a PRA by comparing the PRA to the relevant parts of the ASME/ANS 2009 Standard using a peer review process. 10 CFR 50.69(b)(2)(iii) further requires that the results of the peer review process conducted to meet 10 CFR 50.69(c)(1)(i) criteria be submitted as part of the application.

3.5.1.1 Scope of PRA

The Watts Bar PRA is comprised of a full-power, Level 1, IEPRA, including internal floods, which evaluates the CDF and LERF risk metrics and a SPRA.

The licensee discussed in Section 3.3 of the LAR (Reference 1) that the IEPRA (which includes internal floods) and SPRA models have been assessed against RG 1.200, Revision 2 (Reference 10). Furthermore, LAR Section 3.3 states that a finding closure review was conducted on the identified PRA models in June 2017 (internal events with internal flooding) and April 2017 (seismic). Closed findings were reviewed and closed using the 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 16).

The NRC staff finds that the information provided in the LAR and the supplements (References 2, 3, and 4) were sufficient to support the staff's review of the IEPRA (which includes internal floods) and SPRA for technical acceptability and therefore, meets the requirements set forth in 10 CFR 50.69(b)(2)(iii).

Aspects considered by the staff to evaluate the scope of the PRA include: (1) peer review history, (2) the Appendix X, Independent Assessment process, (3) credit for FLEX in the PRA, and (4) assessment of assumptions and sources of uncertainty. In e-mail correspondences to the licensee on June 18 and September 13, 2019 (References 5 and 6, respectively), the NRC staff issued RAIs to further assess the acceptability of Watts Bar's IEPRA (which includes internal floods) and SPRA for consistency with RG 1.200, Revision 2 (Reference 10), and NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7). The staff's review of these aspects of the PRA and the licensee's supplemental responses to the applicable RAIs are provided in Subsections 3.5.1.2 through 3.5.2.2 and Subsections 3.5.3.1 and 3.5.3.2 of this SE.

3.5.1.2 PRA Peer Review History

Internal Events (including internal flooding)

In Section 3.3 of the LAR, the licensee stated that "[t]he internal events PRA model with internal flooding was subjected to a self-assessment and a full-scope peer review conducted in November 2009." The peer review was assessed against the guidance of RG 1.200, Revision 2 (Reference 10), which endorses the ASME/ANS Ra-Sa-2009 PRA Standard (Reference 11). Subsequently, in June 2017, TVA performed an Independent Assessment for closure of the finding-level Facts and Observations (F&Os) and concluded all but seven of the IEPRA (which includes internal floods) F&Os has been closed. A detailed staff review of this June 2017, Independent Assessment is included below, in Section 3.5.1.3 of this SE.

In Section 3.2 of the LAR, for the IEPRA (includes internal floods) and seismic PRA, TVA stated, in part, "there are no PRA upgrades that have not been peer reviewed." Section X.1.3 of Appendix X to NEI 05-04, 07-12, and 12-13,[7] as accepted by the NRC staff in a memorandum dated May 3, 2017,[8] provides guidance for including a written assessment and justification of whether the resolution of each F&O, within the scope of the Independent Assessment, constitutes a PRA upgrade or maintenance update as defined in the ASME/ANS Ra-Sa-2009 PRA Standard (Reference 11). The NRC staff's review of the internal events and internal flooding PRAs was based on the results of the peer review of the internal events and internal flooding PRAs; the associated F&O closure review described in LAR Sections 3.2.1 and 3.3 and presented in LAR Attachment 3; and previously docketed information on PRA quality submitted to the NRC for relocation of surveillance frequencies to licensee control (Technical Specification Task Force (TSTF)-425) dated October 12, 2018 (Reference 17).

[7] *Errata;* Anderson, V. K., Nuclear Energy Institute, letter to Stacey Rosenberg, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-Out of Facts and Observations,'" dated February 21, 2017 (Reference 16).

[8] Giitter, J., and Ross-Lee, M. J., U.S. Nuclear Regulatory Commission, letter to Mr. Greg Krueger, Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute, Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," dated May 3, 2017 (Reference 19).

Consequently, in LAR Attachment 3 the licensee submitted all the remaining open F&Os from the peer reviews. For each F&O, the licensee provided a disposition for this application. The NRC staff's RAI to clarify the licensee's disposition for some of the findings is described in the following paragraphs.

Regarding resolution of F&Os 2-28 and 7-10 related to updating human failure events (HFEs) timing and limiting joint human error probabilities (JHEPs) to a floor value of 1E-05, it was unclear to the NRC staff if these resolutions would be incorporated into the PRA model of record (MORs) used for categorization and the justification used to demonstrate that the accident sequences or contributors significant to the application decision were not adversely impacted. Therefore, the NRC staff requested clarification in DRA RAI 08.a (Reference 5). In response to DRA RAI 08.a (Reference 3), the licensee stated that the JHEP floor values will be included in the next model update prior to system categorization as documented in Attachment 1 of Reference 2. The NRC staff finds that upon completion of the implementation items provided in Attachment 1 of Reference 2, the JHEP values will be updated in the MOR to a floor value of 1.0E-05, consistent with industry practices described in NUREG-1792 (Reference 18) and, therefore, are acceptable for use in the SSC categorization process.

To specifically address the justification provided for F&O 7-10, the licensee conducted a review of the flooding impacts on the identified operator actions that require less than an hour to perform and conducted a bounding sensitivity analysis to demonstrate that the adjustment of the HRAs with a one hour or less timeframe did not result in a significant change to the model results. In further review of the response to RAI Licensing Branch A (APLA)-04 in TVA's letter dated May 7, 2019, to support the adoption of TSTF-425 (Reference 20), the licensee evaluated the collective impact on CDF and LERF of the human error probabilities (HEPs) using updated bounding values of 0.1 each and determined the impact was less than two percent. Based on the staff's review of the information provided in the licensee's responses to DRA RAI 08.a and RAI APLA-04 (References 3 and 20, respectively), the NRC staff finds that the licensee's conservative sensitivity study is adequate to account for flood-related HEPs for actions requiring a less than one-hour response, and therefore, is acceptable for use in the SSC categorization process.

Open F&Os 5-8 and 7-22 regarded exclusions of operator action HEPs and secondary side isolation that resulted in over-conservative results. The NRC staff noted that PRA model conservatisms can impact component risk importance rankings and therefore requested, in DRA RAI 08.b.i (Reference 5), clarification that these conservatisms would not impact any categorization determination. The licensee stated in its response (Reference 4) that it performed a sensitivity study applying a 0.1 recovery factor and determined that these exclusions did not impact the LERF cutset results and therefore have no impact on SSC categorization. The licensee concluded that this would not be a key source of uncertainty. Based on the information provided in the licensee's response, the NRC staff finds the final dispositions for the F&Os are acceptable for this application.

The NRC staff requested in DRA RAI 08.c (Reference 5) clarification regarding the disposition of open F&O 7-21 that identifies an issue with pipe break failure error factors which were not propagated throughout the model for all individual break sizes. It was not clear if appropriate error factors were used for internal flooding initiating events. In response, the licensee performed a sensitivity study which consisted of applying the appropriate error factors to the initiators and concluded that modifying the error factors of the individual initiators with appropriate values did not result in a significant change to the PRA model results and no change to the list of key assumption/sources of uncertainty identified in Attachment 6 of the LAR

(Reference 1). The NRC staff determined that the response was adequate and did not adversely impact the SSC categorization.

For F&O 5-8, associated with supporting requirements (SRs) LE-C2, LE-C7, LE-C9, and LE-E1, for the disposition provided in Attachment 3 of the LAR, the Independent Assessment Team (IAT) concluded that SR LE-C9 may be considered met at capability category (CC)-II - III and SR LE-C2 remains met at CC-I because there are operator actions following the onset of core damage that were treated conservatively and not updated to address the F&O. The disposition does not address whether the IAT determined if the other SRs, LE-C7 and LE-E1, were met or not met at CC-II. In response to DRA RAI 08.d.i (Reference 3), the licensee confirmed that during the 2009 peer review, the two SRs, LE-C7 and LE-E1 were previously met at all three categories (CC-I through CC-III). The IAT did not document any additional issues regarding these SRs. The NRC staff finds that the LE-C7 and LE-E1 are met at CC-III; therefore, there is no impact on the risk-informed application as it pertains to these F&Os.

Another aspect of F&O 5-8 was the conservatisms of not modeling specific operator actions to mitigate LERF. In response to RAI 08.d.i (Reference 3), the licensee performed a sensitivity study that included the operator actions in question and determined that two additional SSCs would be classified as HSS with regards to LERF. The study also determined that these two SSCs were already classified as HSS with regard to CDF. The NRC staff finds that the exclusion of these operator actions does not impact the application because the SSCs are classified as HSS; therefore, the SSCs will not receive relaxation of treatment requirements in accordance with the 10 CFR 50.69 categorization process.

In response to F&O 3-6, regarding credit of State of Knowledge Correlation (SOKC) for interfacing system loss of coolant accidents (ISLOCA), the licensee proposed implementation item No. 14 that is encompassed in the license condition provided in Attachment 1 to Reference 2 (and subsequently Reference 4) to properly incorporate SOKC in the ISLOCA model prior to risk-informed categorization. The staff finds that upon the completion of the implementation item to incorporate the SOKC for ISLOCA the resolution of F&O 3-6 is acceptable for this risk-informed application.

For F&O 1-6 pertaining to uncertainty data, in response to DRA RAI 08.e.i, the licensee performed a sensitivity study (Reference 17) and concluded that risk categorization point estimate results would not be affected by the missing uncertainty data and does not affect risk categorization. The NRC staff finds the licensee's assessment is acceptable.

In conclusion, the Watts Bar IEPRA (includes internal floods) was peer reviewed consistent with RG 1.200, Revision 2. The NRC staff finds that upon completion of the implementation items provided in Attachment 1 of Enclosure 2 to the letter dated October 28, 2019, the open F&Os have been adequately dispositioned for impact on SSC categorization and therefore the IEPRA meets the requirements set forth in 10 CFR 50.69(c)(1)(i).

Seismic

The NRC staff's review of the licensee's seismic PRA was based on the results of the peer review of the seismic PRA and the associated F&O closure review described in LAR Sections 3.2.3 and 3.3. In the course of its review for this LAR, the staff utilized information from TVA's submittal in response to the 10 CFR 50.54(f) information request arising from Near Term Task Force (NTTF) recommendation 2.1 (Reference 21), the supplement to that submittal dated April 10, 2018 (Reference 22), and the corresponding staff response letter dated July 10,

2018 (Reference 23). The last full-scope peer review of the licensee's seismic PRA was performed in March 2016 against the SPRA requirements in ASME/ANS RA-Sb-2013, also known as Addendum B of the Standard ASME/ANS 2013 PRA standard. RG 1.200, Revision 2, endorses ASME/ANS RA-Sa-2009 which is also known as Addendum A of the PRA Standard and does not endorse Addendum B.

In response to DRA RAI 11 (Reference 2) to demonstrate that the SRs in Part 5 of Addendum B are consistent with those in Addendum A, the licensee stated that its response to RAI APLB 02 for relocation of surveillance frequencies (Reference 17) also applied to this application. In that response, the licensee performed an assessment consistent with the information in Southern Nuclear Operating Company, Inc.'s (SNC) letter to the NRC, NL-17-1201, (Reference 24). TVA stated that Tables 1, 2, and 3 of the assessment in SNC's letter to the NRC, NL-17-1201 were incorporated by reference by TVA for this application and supplemented by licensee-specific information for SRs SHA-B3, SHA-C3, SFRC3, SFR-C6, SFR-G3, and SPR-B1. The staff's review of TVA's basis finds that TVA has appropriately assessed its SPRA against the cited SRs for this application. The NRC staff has previously accepted the assessment in Tables 1, 2, and 3 of the assessment in SNC's letter to the NRC, NL-17-1201 as documented in a letter dated August 10, 2018 (Reference 25). Because the licensee incorporated by reference the assessment in SNC's letter to the NRC, NL-17-1201, dated July 11, 2017, for its SPRA, the NRC staff's acceptance of that assessment is valid for the licensee's SPRA for this application. Based on its review of the licensee's comparison of SRs of Part 5 of Addendum B of the PRA Standard, to those in Addendum A, the staff finds the TVA's use of Addendum B to be an acceptable alternative to the NRC endorsed approach for this application because it adequately addresses the technical elements for the development of a SPRA.

The SPRA peer review and F&O Closure review used NEI 12-13 and did not address the NRC letter dated March 7, 2018 regarding external hazard peer reviews (Reference 26). In response to RAI 02 (Reference 2), the licensee stated that its response to RAI APLB 01 for relocation of surveillance frequencies (Reference 17) was applicable to this application. In that response, the licensee provided descriptions on how it met the NRC staff's clarifications and qualifications on NEI 12-13 in the March 7, 2018 letter. The licensee's response to the 10 CFR 50.54(f) letter, dated June 30, 2017, included, in Section A.2.2 of Appendix A of that submittal, information regarding the qualifications of each WBN SPRA peer reviewer (Reference 21). The information in Section A.2.2 also stated that the peer reviewers met the independence criteria in NEI 12-13. Based on the information provided by the licensee as well as the information available in Section A.3 of Appendix A of the licensee's response to the 10 CFR 50.54(f) letter, the NRC staff finds the licensee's response acceptable for this application because it adequately addresses the qualifications and independence of the SPRA peer review team in the context of WBN SPRA.

In LAR Attachment 3 the licensee submitted all the remaining open finding-level F&Os from the peer reviews for the SPRA. For each finding-level F&O, the licensee provided a disposition for this application. The NRC staff reviewed the licensee's resolution of all the peer review findings and assessed the potential impact of the findings on the categorization of SSCs. The NRC staff determined that each finding for the SPRA was appropriately addressed for this application.

3.5.1.3 Appendix X, Independent Assessment Process for F&O Closure

Section X.1.3 of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 (Reference 16), provides guidance to perform an Independent Assessment for the closure of F&O(s) identified from a full-scope or focused-scope peer review. Appendix X includes guidance for the Independent

Assessment process regarding: (i) the qualifications of the Independent Assessment Team (IAT) members, (ii) pre-review activities, (iii) on-site review activities, and (iv) post-review activities, thus assuring that closure of the F&Os are met at CC-II for the applicable supporting requirements (SR) in the ASME/ANS Ra-SA-2009 PRA Standard (Reference 11), as endorsed by RG 1.200 Revision 2 (Reference 10).

LAR Section 3.3 (Reference 1) states, in part, "[a] finding closure review was conducted on the identified PRA models on June 2017 and April 2017 (seismic). Closed findings were reviewed and closed using the process documented in the NEI letter to the NRC 'Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&O),' […], as accepted by the NRC…"

In its review of the licensee's response to DRA RAI 01.a (Reference 2), the NRC staff concluded that no written assessments for all closed F&Os were provided to the IAT to assure that no new methods and/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, in RAI 01-01 (Reference 6), the NRC staff requested that the licensee clarify how the Watts Bar F&O Closure Review for the IEPRA was performed in accordance with guidance as accepted by the NRC (Reference 19). In response to the RAI, (Reference 4), Watts Bar reconvened the closure review team and provided written justification to conclude that no PRA upgrades were incorporated for closure of the F&Os. The reconvened closure team assessed the written justification provided by the licensee and concurred that the actions performed for closure of the F&Os did not constitute a PRA upgrade. Accordingly, the NRC staff finds that upon the completion of the subsequent review performed by the IAT in determining whether closure of the F&O constituted an upgrade, or a maintenance update, the licensee's assessment has been conducted consistent with the staff acceptance, with conditions, of Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13 (Reference 16).

During the NRC staff's review, it was unclear if the resolutions used to close the F&Os for the June 2017 IAT were incorporated into the PRA MORs to be used for SSC risk categorization. In response to DRA RAI 01.b.iii (Reference 2), the licensee provided an updated license condition that encompasses implementation item No.10 provided in Attachment 1 of Enclosure 2 to the letter dated October 28, 2019 (Reference 4), to include these resolutions into the MORs prior to implementing the 10 CFR 50.69 SSC risk categorization process. The NRC staff finds that upon the completion of the implementation items delineated by the license condition provided in Attachment 1 of Enclosure 2 to the letter dated October 28, 2019, the PRA MOR will reflect the as-built-as-operated plant and the living PRA model reviewed by the IAT and will therefore be consistent with NEI 00-04 guidance as endorsed by the NRC.

The licensee confirmed in response to DRA RAI 01.c (Reference 2) that the IAT members performed the closeout sessions on site and that subsequent reviews were performed either by WebEx or teleconference. The NRC staff finds that the licensee performed the remote consensus meetings consistent with Appendix X guidance, as accepted by the staff (Reference 19).

In DRA RAI 01.d (Reference 5), the NRC further requested the licensee confirm how the IAT assured that the aspects of the underlying SR that were previously not met or met at CC-1, are now met at CC-II. In response to the RAI (Reference 2), the licensee confirmed that the scope of the subsequent review also included documenting for each closed F&O, that the resolution met CC-II requirements for the SRs that were identified as applicable to the F&O. Accordingly, the NRC staff finds that upon the subsequent review performed the licensee added additional

documentation to clarify that the F&Os resolutions were reviewed and determined to meet CC-II for the applicable SR(s).

Appendix X guidance states, in part:

> The relevant PRA documentation should be complete and have been incorporated into the PRA model and supporting documentation prior to closing the finding." [For closure of F&O(s) after the on-site review, Appendix X guidance explicitly states,] "[t]he host utility may, in the time between the on-site review and the finalization of the Independent Assessment team report, demonstrate that the issue has been addressed, that a closed finding has been achieved, and that the documentation has been formally incorporated in the PRA Model of Record [MOR].

Based on its review of the licensee's responses to DRA RAI 01.a through d (Reference 2), the NRC staff concludes that the closure of F&Os for the IEPRA (including internal floods) was performed consistent with Appendix X to NEI 05-04, 07-12, and 12-13 (Reference 16) as accepted by the NRC staff, with conditions by the NRC staff and NEI 00-04 (Reference 8), as endorsed by the NRC for SSC categorization.

Regarding the SPRA model, the licensee stated in the LAR (Reference 1) that in April 2017, an F&Os closure review was performed by an independent team on all SPRA finding-level F&Os. The NRC staff notes that in its supplement to the 10 CFR 50.54(f) submittal dated April 10, 2018 (Reference 22), the licensee confirmed that the independent F&O closure review adhered to the guidance in Appendix X of NEI 12-13 and the two conditions spelled-out in the NRC acceptance letter dated May 3, 2017 (Reference 19). Therefore, the staff finds that the closure of the SPRA finding-level F&Os is acceptable for this application.

3.5.1.4 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 27), 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 in RG 1.200, Revision 2 (Reference 10). In the response to DRA RAI 09 (Reference 2), the licensee stated that "WBN includes the permanently installed FLEX Diesel Generators (DGs) within the PRA model and supporting components including fuel tank, alignment of breakers, buses and operator actions to align the FLEX DGs." The licensee also stated that portable FLEX equipment was not included in the PRA models and confirmed that the FLEX diesels and supporting equipment are included in both the IEPRA (includes internal floods) and SPRA models. In response to DRA RAI 09.b.ii, the licensee performed a sensitivity study increasing the failure probabilities of FLEX equipment threefold. The internal events impact, based on the sensitivity, resulted in no change to the importance measures when compared to the base model importance measures. The licensee also performed a sensitivity case for the SPRA model and concluded that no unique events compared to the internal events model were introduced. In response to DRA RAI 09b.iii regarding the Human Reliability Analysis (HRA) for FLEX diesel generators, the licensee used the HRA Calculator to develop the performance shaping factors. Since each type of FLEX diesel generator has different starting and loading procedures, separate HEPs were developed by the licensee. In response to DRA RAI 09.c.ii, the licensee's inclusion of the permanently installed FLEX diesel generators and supporting

equipment does not constitute a PRA upgrade as defined in the ASME/ANS RA-Sa 2009 Standard (Reference 11) for both IEPRA (includes internal floods) and the SPRA. The staff finds that the licensee's inclusion of FLEX equipment and actions in its IEPRA (including internal floods) and SPRA is acceptable for this application because the inclusion is in accordance with the endorsed PRA standard, the licensee performed a sensitivity analysis to determine the impact on this application, and the FLEX equipment will be part of the sensitivity analysis according to the guidance in NEI 00-04, Revision 0, during the categorization process.

3.5.1.5 Assessment of Assumptions and Approximations

Sections 50.69(c)(1)(i) and (ii) of 10 CFR require that a licensee's PRA be of sufficient quality and level of detail to support the 10 CFR 50.69 categorization process, and that 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.

Table A-1 of RG 1.200, Revision 2, entitled, "Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA" (Reference 10), includes the staff clarification for Section 1-6.1 of ASME/ANS RA-Sa-2009 (Reference 11). The resolution for this clarification states, in part, "[t]herefore, the peer review shall also assess the appropriateness of the assumptions." Regulatory Guide 1.174, Revision 3 (Reference 12), cites NUREG-1855, Revision 1 (Reference 28), as related guidance that includes changes associated with expanding the discussion of sources of uncertainties. NUREG-1855, Revision 1, states, in part, "RG 1.200 [NRC 2009] and the PRA consensus standard published by ASME and ANS (ASME/ANS, 2009) each recognize the importance of identifying and understanding uncertainties as part of the process of achieving acceptability in a PRA, and these references provide guidance on this subject."

Section C.3.3.2 of RG 1.200, Revision 2, clarifies that "[f]or each application that calls upon this regulatory guide, the applicant identifies the key assumptions and approximations relevant to that application. This will be used to identify sensitivity studies as input to the decision-making associated with the application." Revision 2 of RG 1.200 defines the terms "key assumption" and "key source of uncertainty" in Section C.3.3.2, "Assessment of Assumptions and Approximations."

Identification and Characterization of Key Assumptions and Key Sources of Uncertainty

Section 3.2.7 of the LAR states that guidance in NUREG-1855, Revision 0, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2009 (Reference 13), and Electric Power Research Institute (EPRI) TR-1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," December 2008 (Reference 29), were used to identify, characterize, and screen model uncertainties. The NRC staff notes that EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Application with a Focus on the Treatment of Uncertainty," December 2012 (Reference 30), is not included for consideration in NUREG-1855, Revision 0.

In DRA RAI 04.a (Reference 5), the NRC staff requested the licensee to provide a description and justification of the process (e.g., NUREG-1855, Revision 0 or NUREG-1855, Revision 1 and RG 1.200) used to identify the key assumptions and sources of uncertainty provided in the LAR. In response to DRA RAI 04.a (Reference 3), the licensee cited its response to RAI 04.a for the

Sequoyah application for use of 10 CFR 50.69 (Reference 31). In response, the licensee stated that this process was performed by reviewing PRA documentation for generic issues identified in Table A-1 of EPRI 1016737 (Reference 29), as well as identifying plant-specific assumptions and uncertainties, and is therefore consistent with sub step E-1.1 of NUREG-1855, Revision 1. The licensee also stated in the DRA RAI 04.a.i response that the SPRA identified assumptions and sources of uncertainty related to the seismic hazard development, fragility analyses, and plant response model and those assumptions and sources of uncertainty were not characterized as "key." Further, the licensee stated that additional key sources of uncertainty for the SPRA were not identified since the SPRA was built on the IEPRA model.

The NRC staff finds that the licensee appropriately searched for, identified, and evaluated key assumptions and sources of uncertainties for the base IEPRA (includes internal floods) and SPRA for this application consistent with the guidance provided in RG 1.200, and NUREG-1855, Revision 1 (Reference 28). The NRC staff concludes that the licensee provided sufficient information to appropriately disposition the identified assumptions and sources of uncertainty for this application.

<u>Treatment of the Key Assumptions and Sources of Uncertainty</u>

Section 5 of the NEI 00-04 guidance, as endorsed by the NRC, states, in part:

> An analysis of the impacts of parametric uncertainties on the importance measures used in this categorization process was performed and documented in EPRI TR-1008905, Parametric Uncertainty Impacts on Option 2 Safety Significance Categorization […]. The conclusion of this analysis was that the importance measures used in combination with identified set of minimum sensitivity studies adequately address parametric uncertainties.

Furthermore, the guidance in NEI 00-04, Revision 0 (Reference 8), specifies sensitivity studies should be conducted for each PRA model to ensure that PRA assumptions and sources of uncertainty (e.g., human error, common cause failures, and maintenance probabilities) do not mask the SSC(s) importance. Table 5-2 of NEI 00-04 provides several recommended sensitivity studies for the internal events PRA. The licensee provided a list of identified key assumptions and sources of uncertainty as well as the corresponding dispositions of the key assumptions and sources of uncertainty for this application. In DRA RAI 05 (Reference 5), the NRC staff requested the licensee to describe how new key assumptions and sources of uncertainty that could impact component categorization will be determined as part of its 10 CFR 50.69 program. In response, the licensee stated that as part of the proposed license condition in a categorization prerequisite, included in Attachment 1 of Enclosure 2 to the October 28, 2019 letter (Reference 4), it will ensure procedures are updated to perform recategorization upon any impact of a key source and to evaluate for new key sources of uncertainty. The NRC staff finds that the licensee's approach is acceptable for this application because it will identify and evaluate new key assumptions and sources of uncertainty as part of the categorization process.

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

3.5.1.6 Summary of IEPRA and SPRA Acceptability

The NRC staff finds that the licensee has followed the guidance in RG 1.200 and submitted the results of the peer review, and therefore, meets the requirement in 10 CFR 50.69(b)(2)(iii). The NRC staff has reviewed the peer review results, along with information provided in the LAR as supplemented, and finds that the quality and level of detail of the IEPRA and seismic PRA, upon completion of the implementation items provided in Attachment 1 of the letter dated October 28, 2019 (Reference 4), as part of the proposed license condition, prior to SSC categorization, is sufficient to support the categorization of SSCs as required by 10 CFR 50.69 (b)(2)(ii) and use the process endorsed by the NRC staff in RG 1.201. Therefore, the NRC staff concludes that the quality of the IEPRA and seismic PRA meets the requirement in 10 CFR 50.69(c)(1)(i) and (ii).

3.5.2 Importance Measures and Integrated Importance Measures

Section 50.69(c)(1)(i) of 10 CFR states, in part, "[c]onsider results and insights from the plant-specific PRA. These requirements are met, in part, by using importance measures and sensitivity studies consistent with the ASME/ANS RA-Sa-2009 PRA standard (Reference 11), as endorsed in RG 1.200, Revision 2 (Reference 10). RG 1.200, Revision 2, states, in part:

> Methods such as importance measure calculations (e.g., Fussell-Vesely Importance [F-V], risk achievement worth [RAW], risk reduction worth [RRW], and Birnbaum Importance) are used to identify the contributions of various events to the estimation of CDF for both individual sequences and the total CDF [i.e., both contributors to the total CDF, including the contribution from the different hazard groups and different operating modes (i.e., full-and low-power and shutdown) and contributors to each contributing sequence are identified].
>
> The results of the Level 2 PRA are examined to identify the contributors (e.g., containment failure mode, physical phenomena) to the model estimation of LERF or LRF [Large Release Frequency] for both individual sequences and the model as a whole […].

NEI 00-04, Revision 0 (Reference 8), provides guidance where the F-V and RAW importance measures are obtained for each component and each PRA modeled hazard (i.e., separately for the IEPRA (including internal flood) and fire PRA) and the values are then compared to specified criteria as follows:

> Components which have importance measures values that exceed the risk criteria (i.e., F-V greater than 0.005, RAW greater than 2, CCF [common cause failure] RAW greater than 20) are assigned candidate[9] safety-significant.

Section 5.1 of NEI 00-04, Revision 0, recommends that a truncation level of five orders of magnitude below the baseline CDF (or LERF) value should be used for calculating the F-V risk importance measures. The guidance also recommends that the truncation level used should be sufficient to identify all functions with a RAW value greater than 2.

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[9] The term *preliminary* is used synonymous with the term *candidate* in NEI 00-04, Revision 0, guidance. The *candidate* safety significance is not the assigned RISC categorization for the SSC until the IDP has completed its review and approval, consistent with NEI 00-04, Revision 0, guidance, as endorsed by RG 1.201.

3.5.2.1 Importance Measures

In Section 3.1.1 of the LAR (Reference 1), the licensee stated, in part, "[t]he IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address...the interpretation of risk importance measures..." In Section 3.2.3 of the LAR, the licensee stated that the WBN categorization process for seismic hazards will use a peer reviewed plant-specific seismic PRA (SPRA) model. A more detailed review of the characterization for seismic risk using a SPRA is discussed in Section 3.5.1.2 of this SE. Similarly, the assessment for the internal fires does not include the generation of importance measures. A more detailed NRC staff review of the use of the FSSEL is provided in Section 3.5.3.1.

The NRC staff finds that the licensee's use and treatment of importance measures are consistent with the guidance in NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7).

3.5.2.2 Integrated Importance Measures

Section 5.6 of NEI 00-04, Revision 0 (Reference 8), titled, "Integral Assessment," discusses the need for an integrated computation using the available importance measures. It further states, in part, that the "integrated importance measure essentially weights the importance from each risk contributor (e.g., internal events, fire, and seismic PRA 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 F-V, and integrated RAW. The scope of modeled hazards for TVA includes the IEPRA (includes internal floods) and SPRA.

During the NRC staff's review of the LAR (Reference 1), the staff was unclear of the licensee's method to determine integrated importance measures. Therefore, the NRC staff requested in DRA RAI 07 (Reference 5), how the licensee will address the integration of importance measures across all hazards. In response to DRA RAI 07 (Reference 3), the licensee confirmed that WBN will not employ a single top model and will use the approach stated in Section 1.5 and 5.6 of NEI 00-04. In response to DRA RAI 07-01 (Reference 4), the licensee provided details on how the importance measures are calculated for SSCs from each 'bin' in the SPRA. The licensee explained that all failure modes are mapped to the SSC's unique identifier and that the mapping includes all random failures and all seismic failures, thereby accounting for every failure mode in the SPRA. In addition, the importance measure calculation also includes a weighting based on the frequency of the discretized bins. The licensee stated that the risk measures for non-seismic (random) common cause failures events are calculated separately since these failure modes have a different threshold for High Safety Significance (HSS).

The NRC staff requested in DRA RAI 07-01b (Reference 6) that the licensee provide details and justification to support how the integrated importance measures will be calculated for the SPRA modeled basic events which are not aligned with basic events in other PRAs. In its response, the licensee stated that the importance evaluations performed in accordance with the process in NEI 00-04 were determined on a component basis. The licensee stated that subcomponents that were not directly modeled in other PRAs could be treated as another failure mode for the component to which it was associated. The licensee explained that the importance of such a subcomponent would be accounted for in the importance calculation for the corresponding component using the NEI 00-04 formulae for the integral assessment. For the case of SSCs that are unique to the SPRA and for which the seismic basic events were not explicitly modeled

in other PRAs, the licensee stated that if such SSCs are categorized HSS based on the SPRA, then an integral assessment computation was unnecessary, and the safety significance would be presented to the IDP for their consideration in the decision-making process. The licensee provided examples to support its response. Based on its review, the NRC staff finds the licensee's approach for determining the SPRA-specific importance measures for basic events and calculating the corresponding integrated importance measures is acceptable for this application and follows the guidance of NEI 00-04. The NRC staff also finds that the licensee's use and treatment of importance measures for the IEPRA is consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201, Revision 1 (Reference 7).

The licensee stated that the importance measures used for the SPRA will be lower than those in NEI 00-04 by ten percent. The use of an importance measure threshold lower than that in NEI 00-04 by ten percent will result in additional SSCs being candidate HSS from the SPRA. In addition, Section 3.3.2 of the Vogtle 10 CFR 50.69 Safety Evaluation (Reference 32), applied a ten percent margin to the NEI 00-04 thresholds. The NRC staff finds that the use of the ten percent penalty for the importance measure thresholds for categorization using the SPRA is acceptable because it is conservative for this application compared to NEI 00-04. The licensee cited NEI 16-09, "Risk-Informed Engineering Programs (10 CFR 50.69) Implementation Guidance" for this ten percent lower importance measures. The staff has not endorsed NEI 16-09 and the staff's acceptance of the ten percent penalty for this application does not constitute an endorsement of NEI 16-09, in part or in full.

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

As required by 10 CFR 50.69(c)(1)(ii), SSC functional importance must be determined using an integrated, systematic process for addressing initiating events, 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 accidents.

As described in Sections 3.2.2, 3.2.4, and 3.2.5 of the LAR (Reference 1), the licensee's categorization process includes the following non-PRA methods, respectively:

- FSSEL performed for the Fire Protection Program
- Screening analysis performed for the IPEEE for other external hazards
- Screening analysis performed for the IPEEE for other hazards
- Shutdown safety management plan consistent with NUMARC 91-06 (Reference 9)

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

3.5.3.1 Internal Fire Hazard

In the absence of a FIVE analysis or fire PRA (FPRA) specified in the NEI 00-04 (Reference 8) guidance as approaches to address risk from fire, TVA described in Section 3.2.2 of the LAR (Reference 1) an alternate approach. The alternate approach considers the use of the Watts Bar FSSEL for the evaluation of an SSC's safety significance as it pertains to internal fire events during its 10 CFR 50.69 categorization process. The licensee proposed that all the SSCs on the FSSEL will be categorized as HSS and will not be allowed to be re-categorized by the IDP, consistent with the guidance in NEI 00-04 for non-PRA methods. The licensee's SSEL is the result of TVA's safe shutdown analysis methodology used to identify, select, and analyze

systems, components, and cables needed to demonstrate compliance with 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979."

The safe shutdown functions necessary to satisfy the performance goals and safe shutdown functions of Appendix R are the reactivity control function, the reactor coolant makeup function, the reactor coolant pressure control function, the decay heat removal function, the process monitoring function, and the support function. TVA used various analytical approaches to ensure that sufficient plant systems are available to perform fire safe shutdown functions. Numerous plant systems are available, alone and in combination with other systems, to provide the required functions and TVA identified a minimum set of plant systems and components to demonstrate that the plant can achieve and maintain safe shutdown. The safe shutdown analysis methodology ensures that the safe shutdown systems selected are capable of: achieving and maintaining subcritical conditions in the reactor; maintaining reactor coolant inventory; achieving and maintaining hot shutdown conditions for an extended period of time; performing cold shutdown repairs needed to achieve and maintain cold shutdown (or, for control building fires that require shutdown from outside of the main control room, achieving cold shutdown conditions within 72 hours); and maintaining cold shutdown conditions thereafter.

Appendix R, Section III.G.1 of 10 CFR Part 50 requires that fire protection features be provided for those SSCs important to safe shutdown and that these features must be capable of limiting fire damage so that one train of systems necessary to achieve and maintain hot shutdown conditions from either the Control Room or the Emergency Control Station(s) is free of fire damage, and that systems necessary to achieve and maintain cold shutdown from either the Control Room or the Emergency Control Station(s) can be repaired within 72 hours.

Appendix R, Section III.G.2 of 10 CFR Part 50 requires that, except as provided in Section III.G.3, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions be located within the same fire area outside of primary containment, a means of ensuring that one of the redundant trains is free of fire damage shall be provided.

Appendix R, Section III.G.3 of 10 CFR Part 50 describes where alternative or dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room, zone under consideration should be provided.

In DRA RAI 06.a (Reference 5), the NRC staff requested TVA provide justification that the Fire SSEL method is technically adequate relative to the acceptable methods in NEI 00-04. In response to RAI 6 (Reference 2), the licensee referred to RAI 07 from the Sequoyah RAI response (Reference 31). The response cited the results of an NEI and industry study of several plants that compared the number of HSS SSCs identified by the approaches and methods used to assess risk for fire: (1) FPRA, (2) FIVE, and (3) FSSEL. TVA stated, "[t]he study concludes that the proposed FSSEL approach is conservative by introducing significantly more SSCs assigned a HSS classification than use of a FPRA or FIVE, […in addition,] the SSEL approach included all the SSCs identified by the FPRA and the FIVE." While the NRC staff did not review the industry cited study, the NRC staff agrees that the Watts Bar Fire Protection Program identifies a comprehensive list of SSCs (i.e., SSEL) credited to achieve and maintain safe shutdown consistent with the Sequoyah RFOL for applicable requirements to 10 CFR 50.48. The NRC staff finds that the identified SSCs included on the FSSEL and assigned HSS are acceptable for use in the SSC categorization process because the licensee

used the deterministic criteria from Appendix R to identify the functions necessary to achieve and maintain safe shutdown and assigns a HSS categorization to all those SSCs that support the Appendix R functions. SSCs that are categorized HSS do not receive relaxation or special treatment under the 10 CFR 50.69 categorization process; therefore, the licensee's approach to use the FSSEL in the 10 CFR 50.69 program is adequate for the categorization of SSCs. The NRC staff finds that TVA's process used to identify, select, and analyze systems, components, and cables to demonstrate compliance with 10 CFR Part 50, Appendix R, will continue to be managed consistent with the applicable regulatory Appendix R requirements.

In DRA RAI 06.d (Reference 5), the NRC staff requested the licensee to discuss how the credit for operator actions is considered in the analysis for determining SSCs identified based on the FSSEL. In response to DRA RAI 06.d (Reference 2), the licensee stated, "the SSEL Fire Protection Program is based on credible methods (including operator actions) for safely shutting down the plant and maintaining in safe-shutdown for a period of time." In further response to DRA RAI 06.d (Reference 2), the licensee stated "[o]perator actions are not explicitly considered in the safety classification; therefore, there is no assignment of operator action failure probabilities." In Section 3.2.2 of the LAR (Reference 1), TVA stated, "using the Fire SSEL would identify all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge." The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee's use of the Fire SSEL identifies all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge.

LAR Figure 3.1 (Reference 1) illustrates a process flowchart that will be used to assess the fire risk during the 10 CFR 50.69 categorization process. The flowchart assesses (1) whether the SSC is on the FSSEL and (2) for SSCs not on the SSEL, whether the SSC is relied upon to maintain safe shutdown for fire. Answers of yes for either of these two questions result in HSS categorization of the SSC. In Section 3.2.2 of the LAR, TVA stated that the identified SSCs pertaining to the regulatory deviations, multiple spurious operations (MSOs), and additional equipment that is determined to be relied upon to establish and maintain safe shutdown will be retained as HSS for the 10 CFR 50.69 program at Watts Bar. In DRA RAI 06.b(i) (Reference 5), the NRC staff requested the licensee to provide clarification regarding the additional SSCs that are not on the FSSEL but will be identified as HSS. In response to DRA RAI 06.b(i) (Reference 2), TVA stated, in part, "those SSCs relied upon to maintain safe shutdown for fire, the reliance referred to in this diamond include SSCs credited by the Fire Protection Program to mitigate multiple spurious operations (MSOs), SSCs credited for exemptions or deviations taken by the Fire Protection Program, and fire protection equipment SSCs (including fire dampers)." In response to DRA RAI 06.c (Reference 2), the licensee further confirmed that the fire protection system SSCs, including fire detection equipment, suppression equipment, and fire dampers are not specifically included in the FSSEL, but that for the purposes of the 10 CFR 50.69 categorization, the fire protection system SSCs are included within the scope of components assigned HSS classification for internal fire hazards.

Additionally, in response to DRA RAI 06.b.(ii) (Reference 2), TVA confirmed, "SSCs assigned as candidate HSS for non-modeled hazards (including) fire are not allowed to be re-categorized to LSS by the IDP; therefore, they remain HSS." In Section 3.2.2 of the LAR (Reference 1), TVA considered the regulatory exemptions related to Fire Safe Shutdown, previously identified fire-induced MSOs, and additional equipment that is relied upon to establish and maintain safe shutdown. Based on the RAI supplement, the NRC staff finds that retaining the SSCs identified based on the FSSEL as HSS is conservative and acceptable because TVA's approach to using the FSSEL considers additional equipment that includes regulatory exemptions related to the

fire safe shutdown program, previously identified fire-induced MSOs, and additional equipment that is relied upon to establish and maintain safe shutdown that will be retained as HSS.

Section 9.2.3 of NEI 00-04 (Reference 8) guidance for non-safety-related SSCs identified as candidate LSS states, in part, for SSCs, which are important-to-safety, the IDP must consider if the risk information used in the categorization process provides an adequate basis for categorizing the SSC as RISC-4. The FSSEL is a screening approach; therefore, there are no importance measures used in determining safety significance related to the fire hazard and assessment for LSS SSCs can be limited. Regulatory Guide 1.201 (Reference 7) states, in part, "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to provide reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." For SSCs identified as safety-related candidate LSS (i.e., RISC-3), Section 9.2.2 of the NEI 00-04 guidance stipulates consideration by the IDP for DID and safety margin implications for confirming LSS. The NRC staff finds that, in the absence of importance measures to assess the safety significance related to the fire hazard, the other assessments (e.g., DID, safety margin, etc.) for determining the risk categorization of SSCs not included on the Fire SSEL will be performed consistent with the NEI 00-04 guidance, therefore assuring the SSC has been appropriately assigned candidate LSS.

In Section 3.2.2 of the LAR (Reference 1), TVA stated, in part, "the Fire SSEL and the identification of additional equipment relied upon to establish and maintain safe shutdown reflects the current as-built, as-operated plant and that changes to the plant will be evaluated to determine their impact to the equipment list and the categorization process." The FSSEL is an active document that supports the Watts Bar Appendix R program. Changes are managed within the Fire Protection Program consistent with License Conditions 2.F and 2.C.8 for Units 1 and 2, respectively. Therefore, the NRC staff finds that future changes to the Watts Bar Appendix R SSEL will be evaluated to determine their impact on the FSSEL and risk categorization process.

Based upon the NRC staff's review of TVA's approach to using the FSSEL provided in Section 3.2.2 of the LAR (Reference 1) and supplemented in response to DRA RAI 06 (Reference 2), the NRC staff finds the licensee's approach of using the Watts Bar FSSEL to assess the risk for internal fires, when integrated with the other steps/elements provided in the NEI 00-04 guidance as endorsed by the NRC, is acceptable for use in the 10 CFR 50.69 SSC categorization program.

3.5.3.2 External Hazards and Other Hazards (Non-Seismic)

External hazards were initially evaluated by the licensee during the IPEEE (Reference 14). This hazard category includes all non-seismic external hazards such as high winds, external floods, transportation and nearby facility accidents, and other hazards.

In Section 3.2.4 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 WBN [Watts Bar Nuclear Generating Plant] per a plant-specific evaluation in accordance with Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEEs) for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," Supplement 4 (Reference 14), and updated to use the criteria in ASME/ANS PRA Standard RA-Sa-2009. In Attachment 5 of the LAR (Reference 1), the licensee cited NUREG-1407 (Reference 33) and the ASME/ANS PRA Standard RA-Sa-2009 (Reference 11), as the bases for its screening criteria. In RAI 03b, the NRC staff requested

TVA to identify the external hazards that will be evaluated according to the flow chart in NEI 00-04, Section 5.4, Figure 5-6 (Reference 8). In response to DRA RAI 03b (Reference 3), the licensee stated that TVA will subject the external hazards (excluding internal fires and seismic hazards) to the process described by the flow chart in NEI 00-04, Figure 5-6. NEI 00-04, Figure 5-6 provides guidance to be used to determine SSC safety significance for other external hazards (excluding internal fires and seismic hazards). The NRC staff finds that TVA will assess the other external hazards consistent with Figure 5-6 of NEI 00-04 as endorsed in RG 1.201, Revision 1.

In Attachment 4 of the LAR (Reference 1), the licensee provided the results of the plant-specific evaluation that assessed the IPEEE results to the updated endorsed criteria in the ASME/ANS RA-Sa-2009 PRA Standard where external flooding was screened using PS1 and PS2. The NRC notes the staff's assessments of TVA's response to 10 CFR 50.54(f) (References 34 and 35) identified three flooding hazards not bounded by current design basis (local intense precipitation (LIP), flooding in streams and rivers, and the combined effects flood caused by probable maximum flood (PMF) and maximum wind-wave activity. Therefore, in RAI 03a, the NRC staff requested the licensee to provide a detailed discussion if the hazard screening included recent information and an updated reassessment had been conducted. In response to RAI 03.a.(i) (Reference 3), the licensee stated that based on further evaluation, TVA has determined that the more appropriate screening criteria should be C5 for external flooding. Criterion C5 states:

> The event is slow in developing, and it can be demonstrated that there is sufficient time to eliminate the source of the threat or to provide an adequate response.

In RAI 03a(ii) (Reference 3), the licensee provided further clarification that they expect this criterion to be appropriate once the updated analysis is completed. However, if a beyond design basis External Flooding hazard does not screen out using Criterion C5, then the criteria of Section 6-2-3 of the ASME/ANS RA-Sa-2009 (Reference 11) will be used. The licensee included re-confirmation that there is sufficient time to eliminate the source of the threat or to provide an adequate response in accordance with Criterion C5, as part of categorization prerequisites that need to be completed prior to 50.69 categorization per the proposed license condition.

The licensee also stated that active and passive SSCs that are required to mitigate the consequence of external wind or tornado hazards consist of doors, missile shields, hatches, manhole covers, and sumps. The licensee explained that same process as described in the response to RAI 03a.iii, i.e., flow chart in NEI 00-04, Figure 5-6, will also be applied for screening for protection against external wind or tornado hazards.

In summary, the licensee confirmed that the categorization process will not deviate from the guidance presented in NEI 00-04 for the evaluation of non-seismic external hazards and other hazards. The licensee clarified that as part of the categorization process assessment of all hazards except internal fire and seismic events would be performed to determine if there are components being categorized that participate in screened scenarios and whose failure would result in an unscreened scenario. The NRC staff finds that the licensee's categorization process will evaluate the safety significance of SSCs for non-seismic external hazards and other hazards consistent with the guidance provided in Figure 5-6 of NEI 00-04, Revision 0, as endorsed by the NRC in RG 1.201, Revision 1. The NRC staff concludes that the licensee's treatment of other external hazards is acceptable and meets 10 CFR 50.69(c)(1)(ii).

3.5.3.3 Shutdown Risk

Consistent with the guidance in NEI 00-04, Revision 0 (Reference 8), the licensee proposed using the shutdown safety management plan based on NUMARC 91-06 (Reference 9). NUMARC 91-06 provides considerations for maintaining DID for the five key safety functions during shutdown, namely, decay heat removal capability, inventory control, power availability, reactivity control, and containment-primary/secondary. NUMARC 91-06 also specifies that a DID approach should be used with respect to each defined shutdown key safety function. This is accomplished by designating a running and an alternative system/train to accomplish the given key safety function.

The use of NUMARC 91-06 (Reference 9) described by the licensee in the submittal is consistent with the guidance in NEI 00-04, Revision 0 (Reference 8), as endorsed by the NRC in RG 1.201, Revision 1 (Reference 7). The approach uses an integrated and systematic process to identify HSS components, consistent with the shutdown evaluation process. Therefore, the NRC staff finds that the licensee's use of NUMARC 91-06 is acceptable, and meets the requirements set forth in 10 CFR 50.69(c)(1)(ii).

3.5.3.4 Component Safety Significance Assessment for Passive Components

Passive components are not modeled in the PRA; therefore, a different method is necessary to assess the safety significance of these components. Passive components are those components having only a pressure retaining function. This process also addresses the passive function of active components such as the pressure/liquid retention of the body of a motor-operated valve.

In Section 3.1.2 of the LAR (Reference 1), the licensee proposed using a categorization method for passive components not cited in NEI 00-04, Revision 0 (Reference 8), or RG 1.201, Revision 1 (Reference 7), for passive component categorization, which was approved by the NRC for Arkansas Nuclear One, Unit 2 (ANO-2) (Reference 15). 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 (concrete containment) and MC (metal containment) 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 36). 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 Unit 2 Approval of Request for Alternative (2-R&R-004) […] for the passive categorization of Class 2, 3, and non-class components." The NRC staff finds the licensee's proposed approach for passive categorization is acceptable for the 10 CFR 50.69 SSC categorization process.

3.5.3.5 Maintain Defense-in-Depth (NEI 00-04, Section 6)

Section 6 of NEI 00-04, Revision 0 (Reference 8), provides guidance on assessment of DID. Figure 6-1 in NEI 00-04, Revision 0, provides guidance to assess design basis DID based on the likelihood of the design-basis internal initiating event and the number of redundant and diverse trains nominally available to mitigate the initiating event. The likelihoods of the initiating events are binned. The bins for the different likelihoods consider whether HSS is assigned for SSCs that require fewer than the number of mitigating trains nominally available. Section 6 of NEI 00-04, Revision 0, also provides guidance to assess containment DID based on preserving containment isolation and long-term containment integrity and on preventing containment bypass and early hydrogen burns.

RG 1.201, Revision 1 (Reference 7), endorses the guidance in Section 6 of NEI 00-04 (Reference 8), 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 comply with alternative requirements to the Type B and C leakage testing requirements in both Options A and B of Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to 10 CFR Part 50. The criteria provided in section 10 CFR 50.69(b)(1)(x) are not to be used 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. The NRC staff finds that the licensee’s process is consistent with the NRC-endorsed guidance in NEI 00-04 and therefore fulfills the 10 CFR 50.69(c)(1)(iii) criterion that requires DID to be maintained.

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

All the information collected and evaluated in the licensee’s engineering evaluations is provided to the IDP as described in Section 7 of NEI 00-04, Revision 0 (Reference 8). The IDP will make the final decision about the safety significance of SSCs based on guidelines in NEI 00-04, Revision 0, the information the IDP receives, and the IDP’s expertise.

In Section 3.1.1 of the LAR (Reference 1), the licensee stated, in part, “…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the DID assessment (Section 6), the associated system function(s) would be identified as HSS.” The licensee also stated that, “[o]nce a system function is identified as HSS, then all the components that support that function are preliminary HSS.”

The NRC staff finds that the above description provided by the licensee for the preliminary categorization of functions is consistent with NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7), and is therefore acceptable.

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

Section 50.69(c)(1)(iv) of 10 CFR requires, in part, that any potential increases in CDF and LERF resulting from changes to treatment be small. The categorization process described in NEI 00-04, Revision 0, includes an overall risk sensitivity study for all the LSS components to

assure that if the unreliability of the components was increased, the increase in risk would be small (i.e., meet the acceptance guidelines of RG 1.174, Revision 3 (Reference 12)).

Section 3.1.1 of the LAR (Reference 1) states that an unreliability factor of 3 will be used for the sensitivity studies described in Section 8, "Risk Sensitivity Study," of NEI 00-04, Revision 0 (Reference 8). Section 3.2.7 of the LAR further confirms that a cumulative sensitivity study will be performed where the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in PRAs for all systems that have been categorized are increased by a factor of 3. The NRC staff finds the application of a factor of 3 for the sensitivities is consistent with the guidance in NEI 00-04, Revision 0, as endorsed by RG 1.201, Revision 1 (References 8 and 7).

In Section 3.1.1 of the LAR (Reference 1), for the "Overall Categorization Process," TVA specifically noted that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by § 50.69(c)(1)(iv)." This sensitivity study together with the periodic review process discussed in Section 3.6.1 of this SE, assure that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study. The NRC staff finds that the licensee will perform the risk sensitivity study consistent with the guidance in Section 8 of NEI 00-04, Revision 0 (Reference 8), and, therefore, will assure that the potential cumulative risk increase from the categorization is maintained acceptably low, as required by 10 CFR 50.69(c)(1)(iv).

3.5.6 IDP Review and Approval (NEI 00-04, Revision 0, Sections 9 and 10)

As required by 10 CFR 50.69(c)(2), the SSCs must be categorized by an IDP staffed with expert, plant knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operations, design engineering, and system engineering. In Section 3.1.1 of the LAR (Reference 1), the licensee stated that the IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design (mechanical and electrical) engineering, system engineering, safety analysis, and PRA. Therefore, the IDP will include the required expertise.

The guidance in NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7), provides confidence that the IDP expertise is sufficient to perform the categorization and that the results of the different evaluations (PRA and non-PRA) are used in an integrated, systematic process as required by 10 CFR 50.69(c)(1)(ii). In Section 3.1.1 of the LAR (Reference 1), the licensee stated that at least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in modeling and updating of the plant-specific PRA. The licensee further stated that the IDP will be trained in the specific technical aspects and requirements related to the categorization process. This training will address, at a minimum, the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the DID philosophy and requirements to maintain this philosophy. In Attachment 1 of Enclosure 1 to the letter dated October 28, 2019 (Reference 4), the licensee confirmed that it

will provide 10 CFR 50.69 programmatic procedures prior to the use of the categorization process on a plant system. The licensee also confirmed the procedure(s) will specifically include an element for the IDP member qualification requirements. The NRC staff finds that the licensee's IDP areas of expertise meet the requirements in 10 CFR 50.69(c)(2) and the additional descriptions of the IDP characteristics, training, processes, and decision guidelines are consistent with NEI 00-04, Revision 0 (Reference 8), as endorsed in RG 1.201, Revision 1 (Reference 7).

Section 9.2.2, "Review of Safety Related Low Safety-Significant Functions/SSCs," of NEI 00-04, Revision 0 (Reference 8), which is performed by the IDP, states, in part, "[i]n making their assessment, the IDP should consider the impact of loss of the function/SSC against the remaining capability to perform the basic safety functions […]." This section also provides seven specific questions that should be considered by the IDP for making the final determination of the safety-significance for each function/SSC.

The IDP's authority to change component categorization from preliminary HSS to LSS is limited. Consistent with the guidance in NEI 00-04, Revision 0 (Reference 8), and Table 3-1 provided by the licensee in the LAR (Reference 1), components found to be HSS from the following aspects of the process cannot be re-categorized by the IDP: internal events PRA, non-PRA approaches (i.e., FSSEL, shutdown events, other hazards, external events (includes high winds), DID, and passive categorization. SSCs identified as HSS through sensitivity studies outlined in Section 5 of NEI 00-04, may be presented to the IDP for categorization as LSS, if this determination is supported by the integrated assessment process and other elements of the categorization process.

The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 10 CFR 50.69 application team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. The qualitative criteria are the direct responsibility of the IDP, as such changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. The licensee further confirmed that the final assessment of the seven qualitative questions in Section 9.2 of NEI 00-04 (Reference 8) is the IDP's responsibility and that the final categorization of the function will be HSS when any one of the seven questions cannot be confirmed (false response) for that function. The NRC staff finds this acceptable and consistent with the guidance in NEI 00-04, Revision 0, as endorsed in RG 1.201 (Reference 7).

The IDP may change the categorization of a component from LSS to HSS based on its assessment and decision-making. As outlined in Section 10.2, "Detailed SSC Categorization," of NEI 00-04, Revision 0 (Reference 8), and confirmed by the licensee in Section 3.1.1 of the LAR, the IDP's ability to re-categorize components supporting an HSS function from HSS to LSS is limited and only available to the IDP based upon the prescribed steps in the NEI 00-04 guidance as endorsed by RG 1.201, Revision 1 (Reference 7). The steps of the process are performed at either the functional level, component level, or both. For the Watts Bar SSC categorization process, the IDP can re-categorize components from HSS to LSS using the qualitative criteria outlined in Section 9.2 of the NEI 00-04 guidance and PRA sensitivities used to assess the results of the IEPRA (includes internal floods).

As discussed in NEI 00-04, Revision 0 (Reference 8), the only LSS SSC requirements that are relaxed for RISC-3 (LSS) SSCs are those related to treatment, not design or capability, and 10 CFR 50.69(d)(2)(i) requires the licensee ensures, with reasonable confidence, that RISC-3

SSCs remain capable of performing their safety-related functions under design basis conditions. Therefore, the NRC staff finds that the IDP for the Watts Bar categorization process is consistent with the endorsed guidance in NEI 00-04, Revision 0, and, therefore, fulfills 10 CFR 50.69(c)(1)(iv).

3.6 Programmatic Configuration Control (NEI 00-04, Revision 0, Sections 11 and 12)

Section 50.69(c)(1)(ii) of 10 CFR requires, in part, that 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 operating experience. Sections 11 and 12 of NEI 00-04, Revision 0 (Reference 8), includes discussion on Periodic Review and program documentation and change control. Maintaining change control and periodic review provides confidence that all aspects of the 10 CFR 50.69 program and risk categorization for SSCs continually reflect the Watts Bar as built, as-operated plant. A more detailed evaluation is provided in the sections below.

3.6.1 Periodic Review (NEI 00-04, Revision 0, Section 12)

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

Section 11.2 of NEI 00-04, Revision 0 (Reference 8), titled, “Following Initial Implementation,” states, in part, “[t]he periodic update of the plant PRA may affect the results of the categorization process. If the results are affected, the licensee must make adjustments as necessary to either the categorization or treatment processes to maintain the validity of the processes.” In Section 3.2.6 of the LAR (Reference 1), the licensee described the process for maintaining and updating the Watts Bar PRA models used for the 10 CFR 50.69 categorization process. Consistent with NEI 00-04, the licensee stated, “[t]he TVA risk management process ensures that the applicable PRA mode(s) used in this application continue to reflect the as-built and as-operated plant for each of the WBN units.” The licensee’s process includes provisions for: monitoring issues affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operational experience); assessing the risk impact of unincorporated changes; and controlling the model and associated computer files. The process also includes reevaluating previously categorized systems to ensure the continued validity of the categorization.

Regarding the propagation of any changes from the IEPRA to the SPRAs, in DRA RAI 12-01 (Reference 6) the NRC staff requested confirmation that the appropriate changes would be included in the SPRA used for categorization. In response to DRA RAI 12-01 (Reference 4), the licensee included an additional categorization prerequisite as part of the proposed license condition to assess the impact of changes to the IEPRA (including internal flooding) on the Watts Bar SPRA and subsequent risk importance measures, and if impacted, to propagate those model changes to the SPRA model. Based on its review, the NRC staff determined that the licensee’s approach is acceptable because it will ensure that the SPRA model used for categorization is appropriately updated.

Routine PRA updates at Watts Bar are performed every two refueling cycles at a minimum. The NRC staff finds the risk management process described by the licensee in the LAR is consistent with Section 12 of NEI 00-04, Revision 0 guidance as endorsed by the NRC.

3.6.2 Program Documentation and Change Control (NEI 00-04, Revision 0, Section 11)

Section 50.69(f) of 10 CFR requires, in part, program documentation, change control, and records. In Section 3.2.6 of the LAR (Reference 1), the licensee stated that it will implement a process that addresses the requirements in Section 11 of NEI 00-04, Revision 0 (Reference 8), pertaining to program documentation and change control records. Section 3.1.1 of the LAR states that the RISC categorization process documentation will include the following ten elements:

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

Attachment 1 of Enclosure 1 to the letter dated October 28, 2019 (Reference 4), provides the following steps/elements that the licensee will include in the 10 CFR 50.69 programmatic procedures prior to the use of the categorization process: (1) IDP member qualification requirements, (2) qualitative assessment of system functions (3) component safety significance assessment (4) assessment of DID and safety margin (5) review by the IDP (6) overall risk sensitivity study, (7) periodic review, and (8) documentation requirements identified in Section 3.1.1 of the LAR (Reference 1).

In addition, the NRC staff recognizes that for facilities licensed under 10 CFR Part 50, Appendix B Criterion VI, for Document Control, procedures are considered formal plant documents that require "[m]easures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality." The NRC staff finds that the elements provided in Section 3.1.1 of the LAR, in addition to the list of implementation items provided in Attachment 1 of Enclosure 1 to the letter dated October 28, 2019 (Reference 4), for the Watts Bar 10 CFR 50.69 categorization process will be documented in formal licensee procedures consistent with Section 11 of NEI 00-04, Revision 0 (Reference 8), as endorsed by the NRC in RG 1.201, Revision 1 (Reference 7), and are therefore sufficient for meeting the 10 CFR 50.69(f) requirement for program documentation, change control and records.

3.7 Summary of 10 CFR 50.69 Categorization Process

The NRC staff finds the PRAs and use of the non-PRA methods described by the licensee in the submittal (Reference 1), as supplemented by letters dated July 15 and 29, and October 28, 2019 (References 2, 3, and 4 respectively), are acceptable for use in the SSC categorization process. The NRC staff approves the use of the following approaches and methods in the licensee's 10 CFR 50.69 categorization process:

- IEPRA (includes internal flood) to assess the risk from internal events and internal flood, respectively
- Watts Bar Fire SSEL to assess fire risk
- SPRA to assess seismic risk
- Screening analysis performed for the IPEEE to assess the risk for high winds, external floods, and other hazards
- Shutdown Safety Management Plan consistent with NUMARC 91-06 (Reference 9) to assess shutdown risk
- ANO-2 passive categorization method to assess passive components for Class 2 and 3 SSCs and their associated supports (Reference 15)

The NRC staff reviewed all the primary steps outlined in Section 3.2 of this SE used by the licensee in the 10 CFR 50.69 categorization process to assess the safety significance of active and passive components while ensuring the SSC's intended functions remain intact. The NRC staff concludes that the licensee's categorization process adequately implements 10 CFR 50.69 using models, methods, and approaches consistent with NEI 00-04, Revision 0 (Reference 8), as endorsed by the NRC, and therefore, satisfies the requirements of 10 CFR 50.69(c). The NRC staff finds the licensee's proposed categorization process acceptable for categorizing the safety significance of SSCs. Specifically, the NRC staff concludes that the licensee's categorization process:

(1) Considers results and insights from plant-specific internal events PRA that have been subjected to a peer review process against RG 1.200, Revision 2 (Reference 10), as reviewed in Section 3.5.1.1 through 3.5.1.6 of this SE, and with the completion of the implementation items provided Attachment 1 of Enclosure 2 to the letter dated October 28, 2019, will be of sufficient quality and level of detail to support the categorization process, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(i);

(2) Determines SSC functional importance using an integrated systematic process that reasonably reflects the current plant configuration, operating practices, and applicable plant and industry operational experience, as reviewed in Sections 3.5 and 3.6, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(ii);

(3) Maintains DID, as reviewed in Section 3.5.3.5 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iii);

(4) Includes evaluations that provide reasonable confidence that for SSCs categorized as RISC-3, sufficient safety margins are maintained and that any potential increases in CDF and LERF resulting from changes in treatment are small, as reviewed in Section 3.5.3.5 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(1)(iv);

(5) Is performed for entire systems and structures, rather than for selected components within a system or structure, as reviewed in Section 3.5 of this SE, and therefore, the requirements in 10 CFR 50.69(c)(1)(v) will be met upon implementation; and

(6) Includes categorization by IDP, staffed with expert, plant-knowledgeable members whose expertise includes, at a minimum, PRA, safety analysis, plant operation, design engineering and system engineering, as reviewed in Section 3.5.6 of this SE, and therefore, meets the requirements in 10 CFR 50.69(c)(2).

In Attachment 1 of Enclosure 2 to the letter dated October 28, 2019, the licensee provided a list of implementation items that will establish procedure(s) prior to the use of the categorization process on a plant system (Reference 4). These items are required to be implemented by the licensee as set forth in proposed license condition 12(b), discussed below. The list of the implementation items encompasses in its entirety the steps/elements described in the NEI 00-04, Revision 1 as endorsed by the NRC and reviewed by the NRC staff in this SE. Therefore, the NRC staff concludes that upon the completion of these implementation items, the Watts Bar IEPRA (includes internal floods), SPRA, and SSC categorization process will be updated and controlled consistent with 10 CFR 50.69(e) and (f) for ensuring that the categorization of SSCs continue to reflect the as-built-as-operated plant design.

The implementation items provided by the licensee in Attachment 1 of Enclosure 2 to the letter dated October 28, 2019 (Reference 4), are as follows:

1. IDP member qualification requirements.

2. Qualitative assessment of system functions: System functions are qualitatively categorized as preliminary HSS or LSS based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.1.1 of the LAR (Reference 1)). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting LSS function are categorized as preliminary LSS.

3. Component safety significance assessment: Safety significance of active components is assessed through a combination of PRA and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

4. Assessment of DID and safety margin: Components that are categorized as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.

5. Review by the IDP: The categorization results are presented to the IDP for review and approval: The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.

6. Risk sensitivity study: For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to CDF and LERF and meets the acceptance guidelines of RG 1.174 (Reference 12).

7. Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.

   - Consistent with the guidance in NEI 00-04, TVA shall establish in the categorization procedure requirements for periodic re-assessment. The PRA staff will reassess all previously categorized systems whenever a change is made to any key source of uncertainty or assumption listed in Attachment 6 of the LAR (CNL-18-068) (Reference 1). Additionally, the procedure will require evaluation of model changes to identify changes that introduce new key sources of uncertainty and assumptions with respect to the application.

8. TVA shall 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 Low Safety and High Safety Significance is still applicable, and reclassify, if necessary, in accordance with NEI 00-04 (i.e., PRA model update, and at least once per two fuel cycles in a unit).

9. Documentation requirements per Section 3.1.1 of enclosure 1 of the LAR (CNL-18-068) (Reference 1).

In addition to the procedure changes above, Attachment 1 of Enclosure 2 to the letter dated October 28, 2019 (Reference 4) states that TVA will also perform the following actions:

10. As documented in the F&O Closure Report, all changes initiated by the F&O resolutions were confirmed by the Integrated Assessment Team to have been incorporated into the living model and associated documentation. TVA shall update the Model of Record (MOR) with this information prior to system categorization.

11. TVA shall re-introduce the SOKC into the MOR prior to using the PRA model to support categorization of SSCs under 10 CFR 50.69.

12. With respect to the external flooding hazards, TVA shall re-confirm that there is sufficient time to eliminate the source of the threat or to provide an adequate response in accordance with screening criterion C5, prior to 50.69 categorization.

4.0 CHANGES TO THE OPERATING LICENSE

Based on the 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 7), and NEI 00-04, Revision 0 (Reference 8).

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/or list 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 8) 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 3.6 of this SE. In response to DRA RAI 12-01 (Reference 4), the licensee proposed the following amendment

to the FOLs for Watts Bar Nuclear Plant, Units 1 and 2. The proposed license condition states (example for Unit 1):

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

(a) 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, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards; fire hazards by use of the fire protection program (FPP) safe shutdown equipment list (SSEL), and a screening of other external hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009, as specified in Unit 1 License Amendment [Number].

(b) Prior to implementation of the provisions of 10 CFR 50.69, TVA shall complete the implementation items in Enclosure 2, Attachment 1, "List of Categorization Prerequisites," to TVA letter CNL-19-108, "Response to NRC Second Request for Additional Information Regarding Watts Bar Nuclear Plant, Units 1 and 2, Application to Adopt 10 CFR 50.69, 'Risk-informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors' (WBN-TS-17-24) (EPID L-2018-LLA-0493)," dated October 28, 2019.

(13) Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from using the FPP SSEL approach to an internal fire 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 Tennessee State official was notified of the proposed issuance of the amendment on January 14, 2020. 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 Commission previously issued a proposed finding that the amendment involves no significant hazards consideration, published in the Federal Register on July 30, 2019 (84 FR 36969), 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: April 30, 2020

SUBJECT: WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 134 AND 38 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-2018-LLA-0493) DATED APRIL 30, 2020

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