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10 CFR 50.90
10 CFR 50.69

1CAN052102

May 26, 2021

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Application to Adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"

Arkansas Nuclear One, Unit 1
NRC Docket No. 50-313
Renewed Facility Operating License No. DPR-51

As required by Title 10 of the Code of Federal Regulations (10 CFR), Part 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit," and in accordance with the provisions of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," Entergy Operations, Inc. (Entergy) is submitting an application for an amendment to Renewed Facility Operating License (FOL) No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1).

The proposed amendment would modify the ANO-1 licensing basis by the addition of a license condition to allow for the implementation of the provisions of 10 CFR 50.69. These provisions allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the ANO-1 FOL. The categorization process being implemented through this change is consistent with NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, dated July 2005, which was endorsed by the NRC in Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, dated May 2006. Attachment 1 of the enclosure provides a list of categorization prerequisites. Use of the categorization process on a plant system will only occur after these prerequisites are met.

Once approved, the amendment shall be implemented within 60 days.

No new regulatory commitments are made in this submittal.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), a copy of this license amendment request, with enclosure, is being provided to the designated State Officials.

If there are any questions or if additional information is needed, please contact Riley Keele, Manager, Regulatory Assurance, Arkansas Nuclear One, at 479-858-7826.

I declare under penalty of perjury, that the foregoing is true and correct.
Executed on May 26, 2021.

Respectfully,

Ron Gaston

RWG/rwc

Enclosure 1: Evaluation of the Proposed Change

Attachments to Enclosure 1:

1. List of Categorization Prerequisites
2. Description of PRA Models Used in Categorization
3. Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items
4. External Hazards Screening
5. Progressive Screening Approach for Addressing External Hazards
6. Disposition of Key Assumptions/Sources of Uncertainty

Enclosure 2: Proposed Operating License (markup)

Enclosure 3: Revised Operating License

cc: NRC Region IV Regional Administrator

NRC Senior Resident Inspector – Arkansas Nuclear One

NRC Project Manager – Arkansas Nuclear One

Designated Arkansas State Official

Enclosure 1

1CAN052102

Evaluation of the Proposed Change

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EVALUATION OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit," and in accordance with the provisions of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," Entergy Operations, Inc. (Entergy) is submitting an application for an amendment to Renewed Facility Operating License (FOL) DPR-51 for Arkansas Nuclear One (ANO), Unit 1 (ANO-1). The proposed amendment requests U.S. Nuclear Regulatory Commission (NRC) approval to modify the ANO-1 licensing basis by the addition of a license condition to allow for the implementation of the provisions of 10 CFR 50.69. These provisions allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

2.0 DETAILED DESCRIPTION

2.1 Current Regulatory Requirements

The NRC has established a set of regulatory requirements for commercial nuclear reactors to ensure that a reactor facility does not impose an undue risk to the health and safety of the public, thereby providing reasonable assurance of adequate protection to public health and safety. The current body of NRC regulations and their implementation are largely based on a "deterministic" approach.

This deterministic approach establishes requirements for engineering margin and quality assurance in design, manufacture, and construction. In addition, it assumes that adverse conditions can exist (e.g., equipment failures and human errors) and establishes a specific set of design basis events (DBEs). The deterministic approach then requires that the facility include safety systems capable of preventing or mitigating the consequences of those DBEs to protect public health and safety. The Structures, Systems and Components (SSCs) necessary to defend against the DBEs are defined as "safety-related," and these SSCs are the subject of many regulatory requirements, herein referred to as "special treatments," designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. The distinction between "treatment" and "special treatment" is the degree of NRC specification as to what must be implemented for particular SSCs or for particular conditions. Typically, the regulations establish the scope of SSCs that receive special treatment using one of three different terms: "safety-related," "important to safety," or "basic component." The terms "safety-related "and "basic component" are defined in the regulations, while "important to safety," used principally in the general design criteria (GDC) of Appendix A to 10 CFR Part 50, is not explicitly defined.

2.2 Reason for the Proposed Change

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, 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. In contrast to the deterministic approach, 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. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner.

To take advantage of the safety enhancements available through the use of PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

The rule contains requirements on 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, as described by NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference [1]), 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 make adjustments to the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule 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, the rule enables licensees to focus resources on SSCs that make a significant contribution to plant safety. For SSCs that are categorized as high safety significant, existing treatment requirements are maintained or enhanced. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows an alternative risk-informed approach to treatment that provides reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow Entergy to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 Description of the Proposed Change

Entergy proposes the addition of the following condition to the ANO-1 renewed FOL in order to document the NRC's approval of the use 10 CFR 50.69.

> Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, and internal fire; the high wind / tornado safe shutdown equipment list to evaluate high wind / tornado missile events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 1 (ANO-1) passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and their associated supports; the results of the non-PRA evaluations that are based on the IPEEE Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in the Entergy submittal letter dated Date, and all its subsequent associated supplements, as specified in License Amendment No. [XXX] dated [DATE].
>
> Prior NRC approval, under 10 CFR 50.90, will be requested if ANO-1's feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69.

3.0 TECHNICAL EVALUATION

10 CFR 50.69 specifies the information to be provided by a licensee requesting adoption of the regulation. This request conforms to the requirements of 10 CFR 50.69(b)(2), which states:

> A licensee voluntarily choosing to implement this section shall submit an application for license amendment under § 50.90 that contains the following information:
>
> (i) A description of the process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs.
>
> (ii) A description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.
>
> (iii) Results of the PRA review process conducted to meet §50.69(c)(1)(i).
>
> (iv) A description of, and basis for acceptability of, the evaluations to be conducted to satisfy §50.69(c)(1)(iv). The evaluations must include the effects of common cause interaction susceptibility, and the potential impacts from known degradation mechanisms for both active and passive functions and address internally and externally initiated events and plant operating modes (e.g., full power and shutdown conditions).

Each of these submittal requirements are addressed in the following sections.

3.1 Categorization Process Description (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

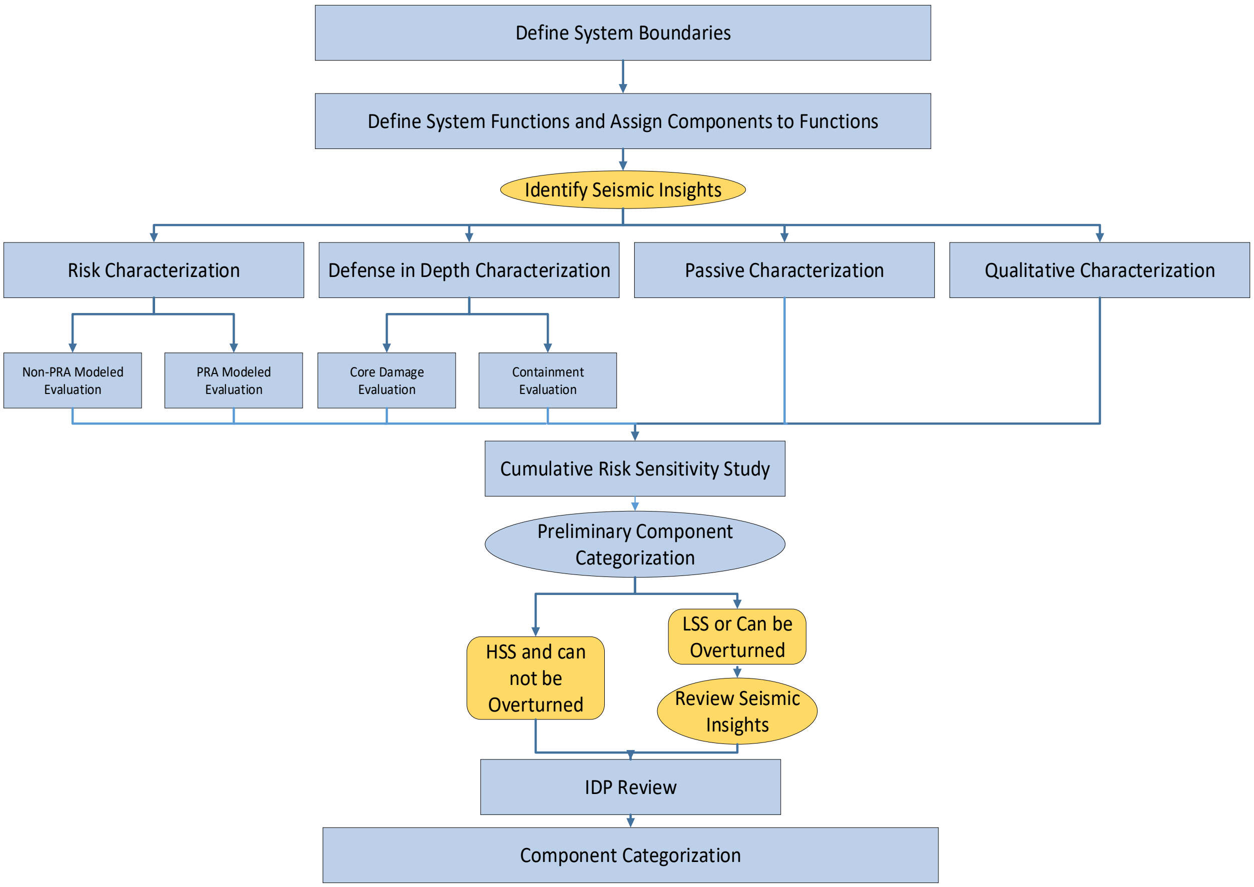
Entergy will implement the risk categorization process in accordance with NEI 00-04, Revision 0, as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" (Reference [2]). NEI 00-04 Section 1.5 states: "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety- significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04 as endorsed by RG 1.201, with the exception of the evaluation of impact of the seismic hazard, which will use the EPRI 3002017583 (Reference [3]) approach for seismic Tier 2 sites, which includes ANO-1, to assess seismic hazard risk for 10 CFR 50.69. Inclusion of additional process steps discussed below to address seismic considerations will ensure that reasonable confidence in the evaluations required by 10 CFR 50.69(c)(1)(iv) is achieved. RG 1.201 states 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)." However, neither RG 1.201 or NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, the order in which each of the elements of the categorization process (listed below) is completed is flexible and as long as all are completed, the elements may be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as Low Safety Significant (LSS) by all other elements. Similarly, NEI 00-04 only requires Item 4 to be completed for safety-related active components/functions categorized as LSS by all other elements.

1. PRA-based evaluations (e.g., the internal events, internal flooding, and Fire PRAs)
2. Non-PRA approaches (e.g., Fire Safe Shutdown Equipment List, Seismic Safe Shutdown Equipment List, other external events screening, and shutdown assessment)
3. Seven qualitative criteria in Section 9.2 of NEI 00-04
4. The defense-in-depth assessment
5. The passive categorization methodology

Figure 3-1 is an example of the major steps of the categorization process described in NEI 00-04; two steps (represented by four blocks on the Figure) have been included to highlight review of seismic insights as pertains to this application, as explained further in Section 3.2.3.

Figure 3-1: Categorization Process Overview



Categorization of SSCs will be completed per the NEI 00-04 process, as endorsed by RG 1.201, which includes the determination of safety significance through the various elements identified above. The results of these elements are used as inputs to arrive at a preliminary component categorization (i.e., HSS or LSS that is presented to the Integrated Decision-Making Panel (IDP). Note: the term "preliminary HSS or LSS" is synonymous with the NEI 00-04 term "candidate HSS or LSS." A component or function is preliminarily categorized as HSS if any element of the process results in a preliminary HSS determination in accordance with Table 3-1 below. The safety significance determination of each element, identified above, is independent of each other and therefore the sequence of the elements does not impact the resulting preliminary categorization of each component or function. Consistent with NEI 00-04, the categorization of a component or function will only be "preliminary" until it has been confirmed by the IDP. Once the IDP confirms that the categorization process was followed appropriately, the final RISC category can be assigned.

The IDP may direct and approve detailed categorization of components in accordance with NEI 00-04 Section 10.2. The IDP may always elect to change a preliminary LSS component or function to HSS, however the ability to change component categorization from preliminary HSS to LSS is limited. This ability is only available to the IDP for select process steps as described in NEI 00-04 and endorsed by RG 1.201. Table 3-1 summarizes these IDP limitations in NEI 00-04. The steps of the process are performed at either the function level, component level, or both. This is also summarized in the Table 3-1. A component is assigned its final RISC category upon approval by the IDP.

Table 3-1: Categorization Evaluation Summary

Element	Categorization Step - NEI 00-04 Section	Evaluation Level	IDP Change HSS to LSS	Drives Associated Functions
Risk (PRA Modeled)	Internal Events Base Case – Section 5.1	Component	Not Allowed	Yes
	Fire, Seismic and Other External Events Base Case		Allowable	No
	PRA Sensitivity Studies		Allowable	No
	Integral PRA Assessment – Section 5.6		Not Allowed	Yes
Risk (Non-modeled)	Fire and Other External Hazards	Component	Not Allowed	No
	Seismic[1]	Function/Component	Allowed	No
	Shutdown – Section 5.5	Function/Component	Not Allowed	No
Defense-in-Depth	Core Damage – Section 6.1	Function/Component	Not Allowed	Yes
	Containment – Section 6.2	Component	Not Allowed	Yes
Qualitative Criteria	Considerations – Section 9.2	Function	Allowable[2]	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

<u>*Notes:*</u>

[1] *This non-modeled seismic categorization element refers to ANO-1; in particular, use of EPRI 3002017583 with EPRI Markups provided in Attachment 2 of References* [4]*and [5]..*

[2] *The assessments of the qualitative considerations are agreed upon by the IDP in accordance with Section 9.2. In some cases, a 10 CFR 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration, however the final assessments of the seven considerations are the direct responsibility of the IDP.*

The seven considerations are addressed preliminarily by the 10 CFR 50.69 categorization team for at least the system functions that are not found to be HSS due to any other categorization step. Each of the seven considerations requires a supporting justification for confirming (true response) or not confirming (false response) that consideration. If the 10 CFR 50.69 categorization team determines that one or more of the seven considerations cannot be confirmed, then that function is presented to the IDP as preliminary HSS. Conversely, if all the seven considerations are confirmed, then the function is presented to the IDP as preliminary LSS.

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 10 CFR 50.69 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. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal Events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04, Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g. Passive, Non-PRA-modeled hazards – see Table 3-1). Except for seismic, these components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Components having seismic functions may be HSS or LSS based on the IDP's consideration of the seismic insights applicable to the system being categorized. Therefore, if an HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component may be driven HSS based on Table 3-1 above or may remain LSS. For the seismic hazard, given that ANO-1 is a seismic Tier 2 (moderate seismic hazard) plant as defined in Reference [3], seismic considerations are not required to drive an HSS determination at the component level, but the IDP will consider available seismic information pertinent to the components being categorized and can, at its discretion, determine that a component should be HSS based on that information.

The following are clarifications to be applied to the NEI 00-04 categorization process:

- 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 probabilistic risk assessment. 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 having a minimum of three years of experience in the modeling and updating of the plant-specific PRA.

- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address at a minimum the purpose of the categorization; present treatment requirements for SSCs including requirements for design basis events; PRA fundamentals; details of the plant specific PRA including the modeling, scope, and assumptions, the interpretation of risk importance measures, and the role of sensitivity studies and the change-in-risk evaluations; and the defense-in-depth philosophy and requirements to maintain this philosophy.
- The decision criteria for the IDP for categorizing SSCs as HSS or LSS pursuant to §50.69(f)(1) will be documented in Entergy procedures.
- Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding safety significant and LSS.
- Passive characterization will be performed using the processes described in Section 3.1.2. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.
- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.
- NEI 00-04 Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5 but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Reference [5] Vogtle Electric Generating Plant, Units 1 and 2 (Vogtle) Safety Evaluation (SE) which states: "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."
- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS Function components to LSS.
- With regard to the criteria that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, Entergy will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.
- ANO-1 proposes to apply an alternative seismic approach to those listed in NEI 00-04 Sections 1.5 and 5.3. This approach is specified in EPRI 3002017583 (Reference [3]) for Tier 2 plants and is discussed in Section 3.2.3.

The risk analysis to be implemented for each modeled hazard is described below.

- Internal Event Risks: Internal events including internal flooding PRA model version 6p0 dated January 20, 2020.

- Fire Risks: Fire PRA model version 5p0 dated February 19, 2019.
- Seismic Risks: EPRI Alternative Approach in EPRI 3002017583 (Reference [3]) for Tier 2 plants with the markups provided in Attachment 2 of References [4] and [5] and additional considerations discussed in Section 3.2.3 of this license amendment request (LAR).
- Extreme Wind or Tornado (Missiles only): Tornado Safe Shutdown Equipment List as discussed in Section 3.2.4 of this LAR.
- Other External Risks (e.g., external floods): Using the IPEEE screening process as approved by NRC SE dated February 27, 2001 (Reference [6]). The other external hazards were determined to be insignificant contributors to plant risk.
- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown Configuration Risk Management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference [7]), which provides guidance for assessing and enhancing safety during shutdown operations.

Prior NRC approval, under 10 CFR 50.90, will be requested if ANO-1 determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69.

The SSC categorization process documentation will include the following elements:

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

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the ANO-2 Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference [8] consistent with the related SE issued by the Office of Nuclear Reactor Regulation.

The RI-RRA methodology is a risk-informed safety classification and treatment program for repair/replacement activities (RI-RRA methodology) for pressure retaining items and the associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., defense in depth, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.

The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in a final SE for Vogtle (Reference [5]). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The passive categorization process is intended to apply the same risk-informed process accepted by the NRC in the ANO-2-R&R-004 relief request (Reference [8]) for the passive categorization of Class 2, 3, and non-class components.

This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities, Section XI, Division 1," and N-662, "Alternative Repair/Replacement Requirements for Items Classified in Accordance with Risk-Informed Processes, Section IX, Division 1" as published in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15. Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair / replacement requirements of the affected LSS components.

Since ANO-1 was designed and built to the ANSI / USAS B31.7 standard which predated the ASME III, Class 1, 2 and 3 piping classification system, for any system chosen for categorization, ANO-1 will categorize all ANSI / USAS B31.7 Class 1 piping to be HSS for pressure retention. The other ANSI / USAS B31.7 piping classifications will then be candidate for categorization under the ANO2-R&R-004, Revision 1, methodology.

All ASME Section III Code Class 1 equivalent SSCs with a pressure retaining function, as well as supports, will be assigned high safety-significant, HSS, for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at ANO-1 for 10 CFR 50.69 SSC categorization.

3.2 Technical Adequacy Evaluation (10 CFR 50.69(b)(2)(ii))

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

The ANO-1 categorization process for the internal events and flooding hazard will use a peer reviewed plant-specific PRA model. The Entergy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for ANO-1. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

The ANO-1 categorization process for fire hazards will use a peer reviewed plant-specific fire PRA model. The internal Fire PRA model was developed consistent with NUREG/CR-6850 and only utilizes methods previously accepted by the NRC. The Entergy risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for ANO-1. Attachment 2 at the end of this enclosure identifies the applicable Fire PRA model.

3.2.3 Seismic Hazards

10 CFR 50.69(c)(1) requires the use of PRA to assess risk from internal events. For other risk hazards, such as seismic, 10 CFR 50.69(b)(2) allows, and NEI 00-04 (Reference [1]) summarizes, the use of other methods for determining SSC functional importance in the absence of a quantifiable PRA (such as Seismic Margin Analysis or IPEEE Screening) as part of an integrated, systematic process. For the ANO-1 seismic hazard assessment, Entergy proposes to use a risk informed graded approach that meets the requirements of 10 CFR 50.69(b)(2) as an alternative to those listed in NEI 00-04, Sections 1.5 and 5.3. This approach is specified in Reference [3] with the EPRI markups provided in Attachment 2 of References [4] and [5] and includes additional considerations that are discussed in this section.

(Note: The discussion below pertaining to Reference [3] includes the markups provided in Attachment 2 of References [4] and [5]).

The proposed categorization approach for ANO-1 is a risk-informed graded approach that is demonstrated to produce categorization insights equivalent to a seismic PRA. This approach relies on the insights gained from the seismic PRAs examined in Reference [3] and plant specific insights considering seismic correlation effects and seismic interactions. Following the criteria in Reference [3], the ANO-1 site is considered a Tier 2 site because the site Ground Motion Response Spectrum (GMRS) to safe shutdown earthquake (SSE) comparison is above the Tier 1 threshold but not high enough that the NRC required the plant to perform a seismic probabilistic risk assessment (SPRA) to respond to Recommendation 2.1 of the Near Term Task Force 50.54(f) letter (Reference [9]). Reference [3] also demonstrates that seismic risk is adequately addressed for Tier 2 sites by the results of additional qualitative assessments discussed in this section and existing elements of the §50.69 categorization process specified in NEI 00-04.

For example, the §50.69 categorization process as defined in NEI 00-04 includes an Integral Assessment that weighs the hazard-specific relative importance of a component (e.g., internal events, internal fire, seismic) by the fraction of the total Core Damage Frequency (CDF) contributed by that hazard. The risk from an external hazard can be reduced from the default condition of HSS if the results of the integral assessment meets the importance measure criteria for LSS. In applying the EPRI 3002017583 (Reference [3]) process to the 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the EPRI 3002017583 guidance and informed of plant SSC-specific seismic insights for the consideration in the HSS/LSS deliberations.

The trial studies in Reference [3], as amended by their RAI responses and amendments (References [10], [11], [12], [13], and [14]), show that seismic categorization insights are overlaid by other risk insights even at plants where the GMRS is far beyond the seismic design basis. Therefore, the basis for the Tier 2 classification and resulting criteria is that consideration of the full range of the seismic hazard produces limited unique insights to the categorization process. That is the basis for the following statements in Table 4-1 of Reference [3].

> *"At Tier 2 sites, there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special seismic risk evaluation process recommended using a Common Cause impact approach in the FPIE PRA can identify the appropriate seismic insights to be considered with the other categorization insights by the Integrated Decision-making Panel for the final HSS determinations."*

At sites with moderate seismic demands (i.e., Tier 2 range) such as ANO-1, there is no need to perform more detailed evaluations to demonstrate the inherent seismic capacities documented in industry sources such as Reference [15]. Tier 2 seismic demand sites have a lower likelihood of seismically induced failures and less challenges to plant systems. This, therefore, provides the technical basis for allowing use of a graded approach for addressing seismic hazards at ANO-1.

Test cases described in Section 3 of Reference [3], as amended by their RAI responses and amendments (References [10], [11], [12], [13], and [14]), indicate that there are very few, if any, SSCs that would be designated HSS for unique seismic reasons. The test cases identified that the unique seismic insights were typically associated with seismically correlated failures and led to unique HSS SSCs. While it would be unusual even for moderate hazard plants to exhibit any unique seismic insights, it is prudent and recommended by Reference [3] to perform additional evaluations to identify the conditions where correlated failures and seismic interactions may occur and determine their impact in the 10 CFR 50.69 categorization process. The special sensitivity study recommended in Reference [3] uses common cause failures, similar to the approach taken in a Full Power Internal Events (FPIE) PRA and can identify the appropriate seismic insights to be considered with the other categorization insights by the IDP for the final HSS determinations.

Entergy is using test case information from Reference [3], developed by other licensees. The test case information is being incorporated by reference into this application, specifically Case Study A (Reference [16]), Case Study C (Reference [17]), and Case Study D (Reference [18]) as well as, RAI responses and amendments (References [10], [11], [12], [13], and [14]), clarifying aspects these case studies.

Basis for ANO-1 being a Tier 2 Plant

As defined in Reference [3] ANO-1 meets the Tier 2 criteria for a "Moderate Seismic Hazard / Moderate Seismic Margin" site. The Tier 2 criteria are as follows:

> *"Tier 2: Plants where the GMRS [Ground Motion Response Spectrum] to SSE [Safe Shutdown Earthquake] comparison between 1.0 Hz and 10 Hz is greater than in Tier 1 but not high enough to be treated as Tier 3. At these sites, the unique seismic categorization insights are expected to be limited."*

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

For comparison, Tier 1 plants are defined as having a GMRS peak acceleration at or below approximately 0.2g or where the GMRS is below or approximately equal to the SSE between 1.0 Hz and 10 Hz. Tier 3 plants are defined where the GMRS to SSE comparison between 1.0 Hz and 10 Hz is high enough that the NRC required the plant to perform an SPRA to respond to the Fukushima 50.54(f) letter (Reference [9]).

As shown in Figure A4-1, comparing the ANO-1 GMRS (derived from the seismic hazard) to the SSE (i.e. seismic design basis capability), the GMRS is below the SSE up through 3 Hz and exceeds the SSE above 3 Hz (Reference [19]). The NRC screened out ANO-1 from performing an SPRA in response to the Near-Term Task Force (NTTF) 2.1 50.54(f) letter (Reference [20]). As such, it is appropriate that ANO-1 is considered a Tier 2 plant. The basis for ANO-1 being Tier 2 will be documented and presented to the IDP for each system categorized.

The following paragraphs describe additional background and the process to be utilized for the graded approach to categorize the seismic hazard for a Tier 2 plant.

Implementation of the Recommended Process

Reference [3] recommends a risk-informed graded approach for addressing the seismic hazard in the 10 CFR 50.69 categorization process. There are a number of seismic fragility fundamental concepts that support a graded approach and there are important characteristics about the comparison of the seismic design basis (represented by the SSE) to the site-specific seismic hazard (represented by the GMRS) that support the selected thresholds between the three evaluation Tiers in the report. The coupling of these concepts with the categorization process in NEI 00-04 are the key elements of the approach defined in Reference [3] for identifying unique seismic insights.

The seismic fragility of an SSC is a function of the margin between an SSC's seismic capacity and the site-specific seismic demand. References such as EPRI NP-6041 (Reference [15]) provide inherent seismic capacities for most SSCs that are not directly related to the site-specific seismic demand. This inherent seismic capacity is based on the non-seismic design loads (pressure, thermal, dead weight, etc.) and the required functions for the SSC. For example, a pump has a relatively high inherent seismic capacity based on its design and that same seismic capacity applies at a site with a very low demand and at a site with a very high demand.

There are some plant features such as equipment anchorage that have seismic capacities more closely associated with the site-specific seismic demand since those specific features are specifically designed to meet that demand. However, even for these features, the design basis criteria have intended conservatisms that result in significant seismic margins within SSCs. These conservatisms are reflected in key aspects of the seismic design process. The SSCs used in nuclear power plants are intentionally designed using conservative methods and criteria to ensure that they have margins well above the required design bases. Experience has shown that design practices result in margins to realistic seismic capacities of 1.5 or more.

In applying the Reference [3] process for Tier 2 sites to the ANO-1 10 CFR 50.69 categorization process, the IDP will be provided with the rationale for applying the Reference [3] guidance and informed of plant SSC-specific seismic insights that the IDP may choose to consider in their HSS/LSS deliberations. As part of the categorization team's preparation of the System Categorization document (SCD) that is presented to the IDP, a section will be included that provides identified plant seismic insights as well as the basis for applicability of the Reference [3] study and the bases for ANO-1 being a Tier 2 plant. The discussion of the Tier 2 bases will include such factors as:

- The moderate seismic hazard for the plant,
- The definition of Tier 2 in the EPRI study, and
- The basis for concluding ANO-1 is a Tier 2 plant.

At several steps of the categorization process, (i.e., as noted in Figure 3-1 and Table 3-1) the categorization team will consider the available seismic insights relative to the system being categorized and document conclusions in the SCD. Integrated importance measures over all modeled hazards (i.e., internal events, including internal flooding, and internal fire for ANO-1) are calculated per Section 5.6 of NEI 00-04, and components for which these measures exceed the specified criteria are preliminary HSS which cannot be changed to LSS. For HSS SSCs uniquely identified by the ANO-1 PRA models but having design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events, will be addressed using non-PRA based qualitative assessments in conjunction with any seismic insights provided by the PRA.

For components that are HSS due to Fire PRA (FPRA) PRA but not HSS due to internal events PRA, the categorization team will review design-basis functions during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events and characterize these for presentation to the IDP as additional qualitative inputs, which will also be described in the SCD.

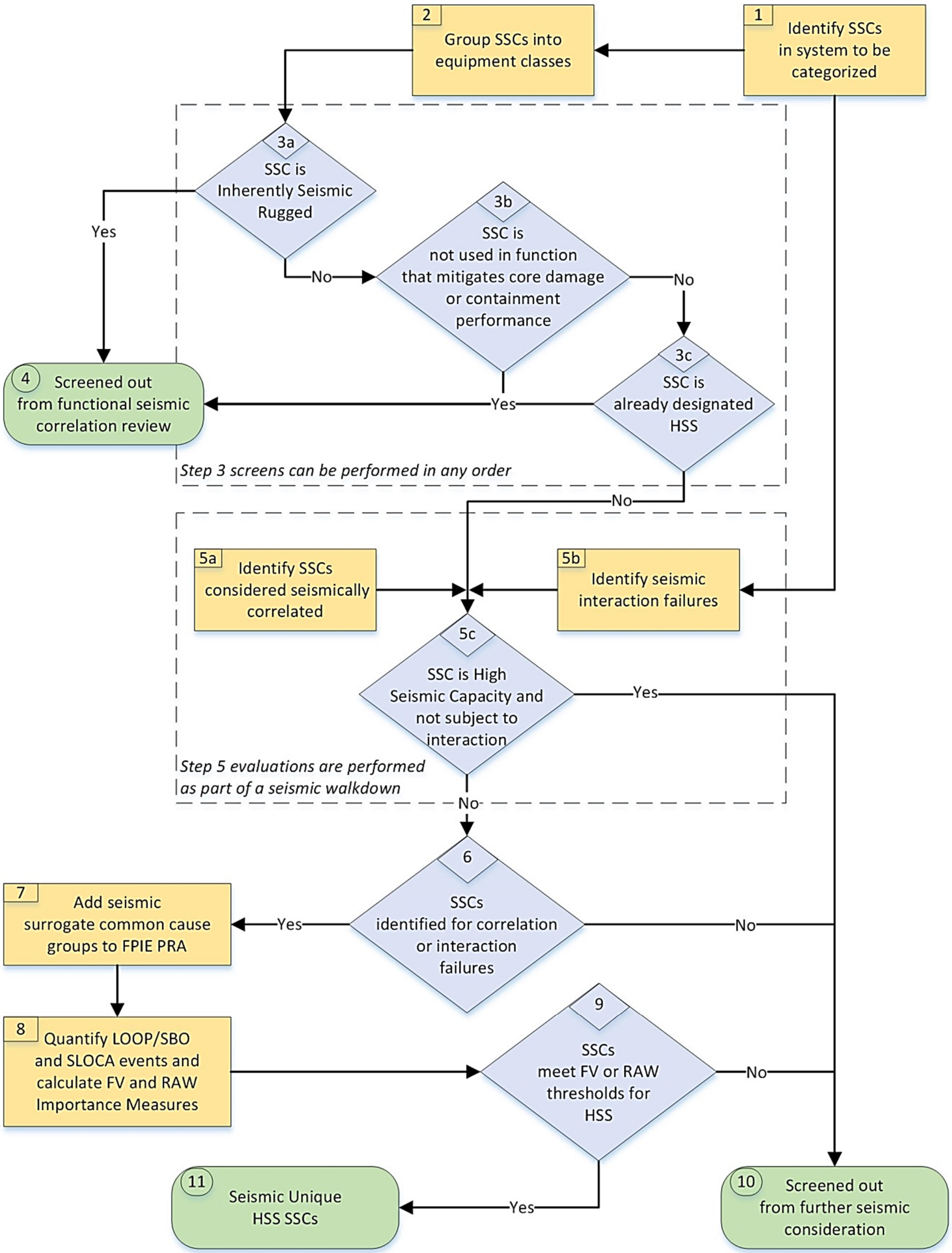
The categorization team will review available ANO-1 plant-specific seismic reviews and other resources such as those identified above. The objective of the seismic review is to identify plant-specific seismic insights that might include potentially important impacts such as:

- Impact of relay chatter
- Implications related to potential seismic interactions such as with block walls

- Seismic failures of passive SSCs such as tanks and heat exchangers
- Any known structural or anchorage issues with a particular SSC
- Components implicitly part of PRA-modeled functions (including relays)

For each system categorized, the categorization team will evaluate correlated seismic failures and seismic interactions between SSCs. This process is detailed in Reference [3], Section 2.3.1 and is summarized below in Figure 3-2. Determination of seismic insights will make use of the full power internal events PRA model supplemented by focused seismic walkdowns and will be utilized on a system basis.

Figure 3-2: Seismic Correlated Failure Assessment for Tier 2 Plants[1]



[1] Reproduced from Reference [3] Figure 2-3

Seismic impacts would be compiled on an SSC basis. As each system is categorized, the system-specific seismic insights will be documented in the categorization report and provided to the IDP for consideration as part of the IDP review process (e.g., Figure 3-1). The IDP can challenge any candidate HSS recommendation for any SSC from a seismic perspective. Any decision by the IDP to downgrade preliminary HSS components to LSS will consider the applicable seismic insights in that decision. SSCs identified from the FPRA as candidate HSS, which are not HSS from the internal events PRA or integrated importance measure assessment, will be reviewed for their design basis function during seismic events or functions credited for mitigation and prevention of severe accidents caused by seismic events. These insights will provide the IDP a means to consider potential impacts of seismic events in the categorization process.

If the ANO-1 seismic hazard changes from medium risk (i.e., Tier 2) at some future time, Entergy will follow its categorization review and adjustment process to review the changes to the plant and update, as appropriate, the SSC categorization in accordance with 10 CFR 50.69(e) and the EPRI 3002017583 SSC categorization criteria for the updated Tier. This includes use of the Entergy corrective action process (CAP).

If the seismic hazard is reduced such that it meets the criteria for Tier 1 in EPRI 3002017583, Entergy will implement the following process.

a) For previously completed system categorizations, Entergy may review the categorization results to determine if use of the criteria in EPRI 3002017583, Section 2.2, "Low Seismic Hazard / High Seismic Margin Sites" would lead to categorization changes. If changes are warranted, the changes will be implemented through the Entergy design control and corrective action programs and NEI 00-04, Section 12.

b) Seismic considerations for subsequent system categorization activities will be performed in accordance with the guidance in EPRI 3002017583, Section 2.2, "Low Seismic Hazard / High Seismic Margin Sites."

If the seismic hazard increases to the degree that an SPRA becomes necessary to demonstrate adequate seismic safety, Entergy will implement the following process following completion of the SPRA, including adequate closure of Peer Review Findings and Observations (F&Os).

a) For previously completed system categorizations, Entergy will review the categorization results using the SPRA insights as prescribed in NEI 00-04, Section 5.3, "Seismic Assessment" and Section 5.6, "Integral Assessment". If changes are warranted, the changes will be implemented through the Entergy design control and corrective action programs and NEI 00-04 Section 12.

b) Seismic considerations for subsequent system categorization activities will follow the guidance be performed in accordance with NEI 00-04 criteria, as recommended in EPRI 3002017583, Section 2.4, "High Seismic Hazard / Low Seismic Margin Sites."

In all cases, prior NRC approval, under 10 CFR 50.90, will be requested if ANO-1's feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69.

Historical Seismic References for ANO-1

The ANO-1 SSE and GMRS curves from the seismic hazard and screening response are shown in Section 2.4 and 3.1 in the seismic hazard and screening report (Unit 1 section) of Reference [21]. The ANO-1 SSE and GMRS curves from the seismic hazard and screening response are shown in Figure A4-1 of Attachment 4 of this request. The NRC's Staff assessment of the ANO-1 seismic hazard and screening response is documented in Reference [20]. In the Staff Confirmatory Analysis (Section 3.3.3) of Reference [20], the NRC concluded that the methodology used by Entergy in determining the GMRS was acceptable and that the GMRS determined by Entergy adequately characterizes the reevaluated hazard for the ANO-1 site.

Section 1.1.3 of Reference [3] cites various post-Fukushima seismic reviews performed for the U.S. fleet of nuclear power plants. For ANO-1, the specific seismic reviews prepared by the licensee and the NRC's staff assessments are provided here. These licensee documents were submitted under oath and affirmation to the NRC.

1. NTTF Recommendation 2.1 seismic hazard screening (Reference [21] and [20]).

2. NTTF Recommendation 2.1 spent fuel pool assessment (References [22] and [23]).

3. NTTF Recommendation 2.3 seismic walkdowns (References [24] and [25]).

4. NTTF Recommendation 4.2 seismic mitigation strategy assessment (S-MSA) (References [26], and [27]).

The following additional post-Fukushima seismic reviews were performed for ANO-1:

5. NTTF Recommendation 2.1 seismic Expedited Seismic Evaluation Process (ESEP) (References [28], and [29]).

6. NTTF Recommendation 2.1 seismic High Frequency Evaluation (References [30] and [31].

Summary

Based on the above, the Summary from Section 2.3.3 of Reference [3] applies to ANO-1; namely, ANO-1 is a Tier 2 plant for which there may be a limited number of unique seismic insights, most likely attributed to the possibility of seismically correlated failures, appropriate for consideration in determining HSS SSCs. The special sensitivity study recommended using common cause failures, similar to the approach taken in a FPIE PRA, can identify the appropriate seismic insights to be considered with the other categorization insights by the IDP for the final HSS determinations. Use of the EPRI approach outlined in Reference [3] to assess seismic hazard risk for §50.69 with the additional reviews discussed above will provide a process for categorization of RISC-1, RISC-2, RISC-3, and RISC-4 SSCs that satisfies the requirements of §50.69(c).

3.2.4 Other External Hazards

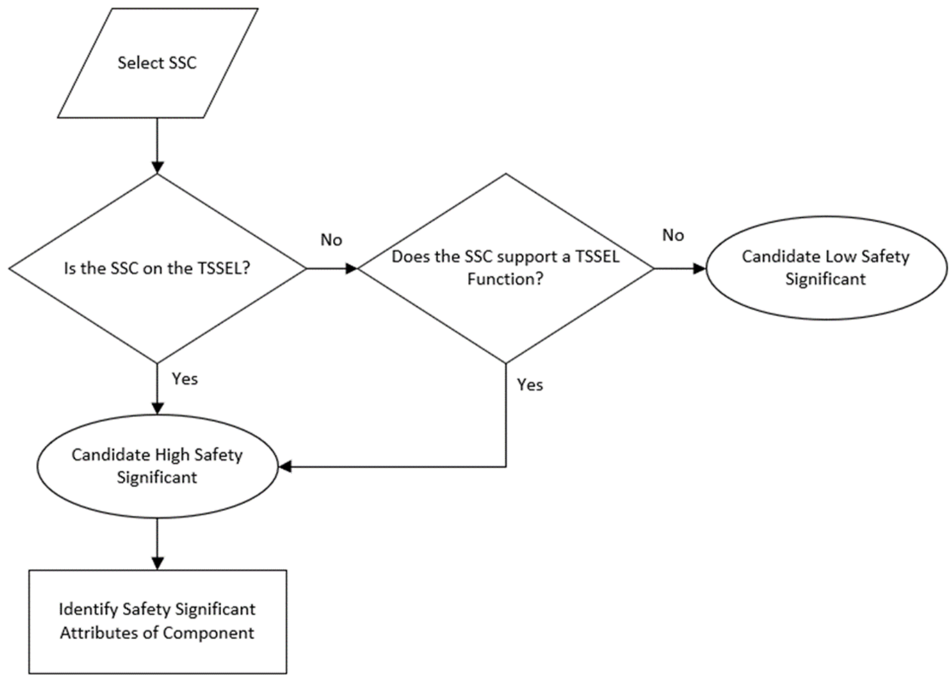
Tornado Missiles

Wind pressure effects from high winds and tornados are screened from further evaluation; see Attachment 4. Since the tornado missile hazard is not screened, the ANO-1 categorization process will use the safety significance process described below to determine the safety significance of SSCs for the tornado missile hazard. The hazard is assumed to be present during a tornado-induced loss of offsite power.

The tornado missile hazard safety significance process uses a Tornado Safe Shutdown Equipment List (TSSEL) of SSCs that was developed from a list of SSCs needed to achieve and maintain safe shutdown of the reactor assuming unavailability of offsite power. During categorization of systems, NEI 00-04 component to function mapping process will be applied to the safe shutdown function of (1) Decay Heat Removal; (2) Reactivity Control; (3) Inventory Control; (4) Power Availability, and (5) Reactor Pressure Control. The SSCs that fulfill the tornado missile safe shutdown functions, as well as any tornado missile barriers that are credited with protecting equipment that fulfills a TSSEL function, will be identified as candidate HSS for the system being categorized regardless of their tornado damage susceptibility or frequency of challenge. This approach ensures the SSCs that are credited to achieve and maintain the capability for safe shutdown are retained as safety-significant.

The safety significance process for the tornado missile hazard is shown in Figure 3-3. There are no importance measures used in determining safety significance of SSCs related to the tornado missile hazard. As stated in NEI 00-04, an SSC identified as HSS by a non-PRA method for external events "may not be re-categorized by the IDP."

Figure 3-3: Safety Significance Process for Systems and Components for the Tornado Missile Protection Program



<u>All other external hazards</u>

All other external hazards, except for seismic, were screened for applicability to ANO-1 per a plant-specific evaluation in accordance with GL 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4" (Reference [32]), and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power and Shutdown

Consistent with NEI 00-04, the ANO-1 categorization process will use the shutdown safety management plan described in NUMARC 91-06 for evaluation of safety significance related to low power and shutdown conditions. The overall process for addressing shutdown risk is illustrated in Figure 5-7 of NEI 00-04.

NUMARC 91-06 specifies that a defense-in-depth approach should be used with respect to each defined shutdown key safety function. The key safety functions defined in NUMARC 91-06 are evaluated for categorization of SSCs.

SSCs that meet either of the two criteria (i.e., considered part of a "primary shutdown safety system" or a failure would initiate an event during shutdown conditions) described in Section 5.5 of NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The Entergy risk management process ensures that the applicable PRA models used in this application continues to reflect the as-built and as-operated plant for ANO-1. The process delineates the responsibilities and guidelines for updating the PRA models, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, and industry operational experience) for assessing the risk impact of unincorporated changes, and for controlling the model and associated computer files. The process will assess the impact of these changes on the plant PRA model in a timely manner but no longer than once every two refueling outages. If there is a significant impact on the PRA model, the SSC categorization will be re-evaluated.

Entergy will also implement a process that addresses the requirements in NEI 00-04, Section 11, "Program Documentation and Change Control." The process will review the results of periodic and interim updates of the plant PRA that may affect the results of the categorization process. If the results are affected, adjustments will be made as necessary to the categorization or treatment processes to maintain the validity of the processes. In addition, any PRA model upgrades will be peer reviewed prior to implementing those changes in the PRA model used for categorization.

3.2.7 PRA Uncertainty Evaluations

Uncertainty evaluations associated with any applicable baseline PRA model(s) used in this application were evaluated during the assessment of PRA technical adequacy and confirmed through the self-assessment and peer review processes as discussed in Section 3.3 of this enclosure.

Uncertainty evaluations associated with the risk categorization process are addressed using the processes discussed in Section 8 of NEI 00-04 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04.

In the overall risk sensitivity studies, Entergy will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference [5]. Consistent with the NEI 00-04 guidance, Entergy will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures 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 detailed process of identifying, characterizing and qualitative screening of model uncertainties is found in Section 5.3 of NUREG-1855 and Section 3.1.1 of EPRI TR-1016737 (Reference [33]). The process in these references was mostly developed to evaluate the uncertainties associated with the internal events PRA model; however, the approach can be applied to other types of hazard groups.

The list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the evaluation of this application. If the ANO-1 PRA model used a non-conservative treatment, or methods that are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine its impact on this application. Only those assumptions or sources of uncertainty that could significantly impact the risk calculations were considered key for this application.

Key ANO-1 PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 6. The conclusion of this review is that no additional sensitivity analyses are required to address ANO-1 PRA model specific assumptions or sources of uncertainty.

3.3 PRA Review Process Results (10 CFR 50.69(b)(2)(iii))

The PRA models described in Section 3.2 have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference [34]), consistent with NRC Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation."

The Internal Events/Internal Flooding PRA model was subject to a self-assessment and a full peer review conducted in August 2009 against ASME/ANS RA-SA-2009 and RG 1.200, Revision 2. The Internal Flooding PRA model was subject to a self-assessment and a Focused-Scope Peer Review (FSPR) conducted in February-March 2017 against ASME/ANS RA-SA-2009 and RG 1.200, Revision 2. A LERF FSPR was conducted in August 2019 against ASME/ANS RA-SA-2009 and RG 1.200, Revision 2.

The FPRA model was subject to a self-assessment and a full-scope peer review was conducted in October 2009 against ASME/ANS RA-SA-2009 and RG 1.200, Revision 2. The ANO-1 FPRA has also been subject to three additional FSPRs in May 2012, October 2012, and June 2014. During the evolution of the NFPA 805 project, some changes to the fire scenario methodologies were applied to both refine the model and results, and to comply with approved methods more fully.

Closed findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12, and NEI 12-13, "Close-out of Facts and Observations" (Reference [35]) as accepted by NRC in the letter dated May 3, 2017 (Reference [36]). The results of this review have been documented and are available for NRC audit.

All findings were closed as shown in Attachment 3. There are no open peer review findings for the internal events, internal flooding, or FPRA models. However, there were three Supporting Requirements (SRs) that had Suggestions tied to the SR which led to a Capability Category I (CCI) in the peer review which were overlooked during the F&O closure process. Although there were no Findings associated with these, the SRs are listed in Attachment 3 for completeness and have been dispositioned. The attachments identified above demonstrate that the PRA is of sufficient quality and level of detail to support the categorization process and has been subjected to a peer review process assessed against a standard or set of acceptance criteria that is endorsed by the NRC as required by 10 CFR 50.69(c)(1)(i).

3.4 Risk Evaluations (10 CFR 50.69(b)(2)(iv))

The ANO-1 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of §50.69(b)(2)(iv). Sensitivity studies described in NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF. The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 Feedback and Adjustment Process

If significant changes to the plant risk profile are identified, or if it is identified that a RISC-3 or RISC-4 SSC can (or actually did) prevent a safety significant function from being satisfied, an immediate evaluation and review will be performed prior to the normally scheduled periodic review. Otherwise, the assessment of potential equipment performance changes and new technical information will be performed during the normally scheduled periodic review cycle.

To more specifically address the feedback and adjustment (i.e., performance monitoring) process as it pertains to the proposed ANO-1 Tier 2 approach discussed in Section 3.2.3, implementation of the Entergy design control and corrective action programs will ensure the inputs for the qualitative determinations for the seismic hazard continue to remain valid to maintain compliance with the requirements of 10 CFR 50.69(e).

The performance monitoring process is described in the Entergy 10 CFR 50.69 program documents. The program requires that the periodic reviews assess changes that could impact the categorization results and provides the IDP with an opportunity to recommend categorization and treatment adjustments. Station personnel from Engineering, Operations, Risk Management, Regulatory Assurance, and others have responsibilities for preparing and conducting various performance monitoring tasks that feed into this process. The intent of the performance monitoring reviews is to discover trends in component reliability; to help identify and reverse negative performance trends and take corrective action if necessary.

The Entergy configuration control process ensures that changes to the plant, including a physical change to the plant and changes to documents, are evaluated to determine the impact to drawings, design bases, licensing documents, programs, procedures, and training. The configuration control program has been updated to include a checklist of configuration activities to recognize those systems that have been categorized in accordance with 10 CFR 50.69 and to ensure that any physical change to the plant or change to plant documents is evaluated prior to implementing those changes.

The checklist includes:

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

- Steps to be performed if redundancy, diversity, or separation requirements are identified or affected. These steps include identifying any potential seismic interaction between added or modified components and new or existing safety related or safe shutdown components or structures.

- Review of impact to seismic loading, SSE seismic requirements, as well as the method of combining seismic components.

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

Entergy has a comprehensive problem identification and corrective action program that ensures issues are identified and resolved. Any issue that may impact the 10 CFR 50.69 categorization process will be identified and addressed through the problem identification and corrective action program, including seismic-related issues.

The Entergy 10 CFR 50.69 program requires that SCDs cannot be approved by the IDP until the panel's comments have been resolved to the satisfaction of the IDP. This includes issues related to system-specific seismic insights considered by the IDP during categorization.

Scheduled periodic reviews no longer than once every two refueling outages will evaluate new insights resulting from available risk information (i.e., PRA model or other analysis used in the categorization) changes, design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This scheduled review will include:

- A review of plant modifications since the last review that could impact the SSC categorization.

- A review of plant specific operating experience that could impact the SSC categorization.

- A review of the impact of the updated risk information on the categorization process results.

- A review of the importance measures used for screening in the categorization process.

- An update of the risk sensitivity study performed for the categorization.

In addition to the normally scheduled periodic reviews, if a PRA model or other risk information is updated, a review of the SSC categorization will be performed.

The periodic monitoring requirements of the 10 CFR 50.69 process will ensure that these issues are captured and addressed at a frequency commensurate with the issue severity. The 10 CFR 50.69 periodic monitoring program includes immediate and periodic reviews that include the requirements of the regulation to ensure that all issues that could affect 10 CFR 50.69 categorization are addressed. The periodic monitoring process also monitors the performance and condition of categorized SSCs to ensure that the assumptions for reliability in the categorization process are maintained.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

The following NRC requirements and guidance documents are applicable to the proposed change.

- The regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."
- NRC Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, April 2015.
- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009.

The proposed change is consistent with the applicable regulations and regulatory guidance.

4.2 No Significant Hazards Consideration Analysis

Using the criteria in Title 10 of the Code of Federal Regulations (10 CFR), Part 50.92, "Issuance of amendment," Entergy Operations, Inc. (Entergy) has evaluated the proposed license amendment to Renewed Facility Operating License (FOL) DPR-51 for Arkansas Nuclear One (ANO) Unit 1 (ANO-1). The proposed amendment modifies the ANO-1 licensing basis, by the addition of a license condition to allow for the voluntary implementation of the provisions of Title 10 CFR 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 equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has safety significance resulting in improved plant safety.

Entergy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of Structures, Systems and Components (SSCs) subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The process used to evaluate SSCs for changes to NRC special treatment requirements and the use of alternative requirements ensures the ability of the SSCs to perform their design function. The potential change to special treatment requirements does not change the design and operation of the SSCs. As a result, the proposed change does not significantly affect any initiators to accidents previously evaluated or the ability to mitigate any accidents previously evaluated. The consequences of the accidents previously evaluated are not affected because the mitigation functions performed by the SSCs assumed in the safety analysis are not being modified. The SSCs required to safely shut down the reactor and maintain it in a safe shutdown condition following an accident will continue to perform their design functions.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not change the functional requirements, configuration, or method of operation of any SSC. Under the proposed change, no additional plant equipment will be installed.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change will permit the use of a risk-informed categorization process to modify the scope of SSCs subject to NRC special treatment requirements and to implement alternative treatments per the regulations. The proposed change does not affect any Safety Limits or operating parameters used to establish the safety margin. The safety margins included in analyses of accidents are not affected by the proposed change. The regulation requires that there be no significant effect on plant risk due to any change to the special treatment requirements for SSCs and that the SSCs continue to be capable of performing the associated design basis functions, as well as to perform any beyond design basis functions consistent with the categorization process and results.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

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[52] ASME/ANS RA-S-2009, Addenda to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2009.

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ATTACHMENTS

1. List of Categorization Prerequisites
2. Description of PRA Models Used in Categorization
3. Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items
4. External Hazards Screening
5. Progressive Screening Approach for Addressing External Hazards
6. Disposition of Key Assumptions/Sources of Uncertainty

Enclosure 1, Attachment 1

1CAN052102

List of Categorization Prerequisites

Attachment 1

List of Categorization Prerequisites

Entergy will establish procedure(s) prior to the use of the categorization process on a plant system. The procedure(s) will contain the elements/steps listed below.

- Integrated Decision-Making Panel (IDP) member qualification requirements.

- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven criteria in Section 9 of NEI 00-04 (see Section 3.2). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting, an LSS function are categorized as preliminary LSS.

- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.

- Assessment of defense-in-depth (DID) and safety margin. Safety-related components that are categorized as preliminary LSS are evaluated for any role in providing DID and safety margin and, if appropriate, upgraded to HSS.

- 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.

- 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 core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.

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

- Documentation requirements per Section 3.1.1 of the enclosure.

Enclosure 1, Attachment 2

1CAN052102

Description of PRA Models Used in Categorization

Attachment 2

Description of PRA Models Used in Categorization

Unit	Model	Baseline CDF	Baseline LERF	Comments
Full Power Internal Events / Internal Flooding (FPIE and IF) PRA Model				
1	Model: 6p0	6.5E-06	3.5E-08	2020 FPIE/IF Model of Record See Section 3.3 for Peer Review Discussion.
Fire PRA (FPRA) Model				
1	Model 5p0a	3.7E-05	6.9E-06	2019 Fire PRA Model of Record See Section 3.3 for Peer Review Discussion.

<u>List of Acronyms</u>

CDF	Core Damage Frequency
FPIE	Full Power Internal Events
IF	Internal Flooding
LERF	Large Early Release Frequency
PRA	Probabilistic Risk Assessment

Enclosure 1, Attachment 3

1CAN052102

Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Attachment 3

Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Fire Probabilistic Risk Assessment (FPRA) Model

Suggestion Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 10 CFR 50.69
CF-A2-01	CF-A2	N/A	The summary report, Appendix D for task 10 (Circuit Failure Model Likelihood Analysis) characterizes the uncertainty associated with method employed in determining failure probabilities. The conclusion of this report is that the "application of circuit failure probabilities is considered to have minimal impact on the results." Though it may be that a detailed analysis technique was followed for dominant scenarios, these failures are still in the dominant sequences. The accuracy and uncertainty associated with these values would have a significant impact on the results. Characterize the uncertainty with respect to how the method employed could introduce uncertainty into the final results.	PSA-AN01-03-FQ-01 utilizes a mean value and statistical representation of the uncertainty intervals associated with the hot short probabilities and duration terms from NUREG-7150. A type code was developed for each hot short probability and duration term utilized in the ANO-1 FPRA model similarly to the approach used for ignition frequencies which was met at CC-II or greater. A state of knowledge correlation has been performed for all sequences and cutsets and a mean value is calculated for the core damage frequency (CDF) / larger early release frequency (LERF) risk metric by propagating the uncertainty distributions using the Monte Carlo approach through the PRA model. Therefore, this Supporting Requirement (SR) has been resolved in the model of record and has no impact on the applications and has been self-assessed at CC-II or greater.

Suggestion Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 10 CFR 50.69
N/A- Not tied to Specific F&O or suggestion. Met CC-II when F&O HRA-D2-01 was resolved and closed.	HRA-D1	N/A	One recovery action is incorporated into the PRA model for multiple loss of DC breaker for 4160 bus events, which have an accident sequence associated with them. The event found indicates that recovery actions were incorporated for significant sequences rather than universally. The identification of all recovery actions used in the model is documented in Attachment D of ANO-1 Fire HRA Notebook (Report 0247060006.03-U1). Most fire-specific recoveries used screening values, so this was set as CC-I.	New fire human failure events (HFEs) were created for risk significant fire scenarios as identified through the review of the plant fire procedures and FPRA cutset model reviews. Detailed human reliability analysis (HRA) analysis was performed to develop these HFEs as outlined in Section 7.0 of the Fire HRA notebook (PSA-ANO1-03-HRA, R1). In summary, all fire HFEs have been developed following the process outlined in NUREG-1921 (Reference [37] of Enclosure 1), for which a focus scope peer review has been performed, and no longer implements a screening value for assuming the failure rate of operator actions. Therefore, this SR has been resolved in the model of record and has no impact on the applications and has been self-assessed at CC-II or greater.

Suggestion Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 10 CFR 50.69
FSS-B2-**01** FSS-B2-02	FSS-B2	N/A	Section 3.2 of the Fire Scenarios Report (0247-06-0006.05-U1) details MCR abandonment treatment. The scenario is 129-F, Scenario A. A CCDP of 0.1 is used based on an adequate evaluation of Appendix R, III.L requirements. The evaluation is based on adequate alternate shutdown procedures, validation of timing and manual action feasibility. Calculation ANO1-FP-09- 00011, Rev. 2, evaluates the abandonment times for the Unit 1 MCR. These arguments bound the fire risk contribution for MCR abandonment. A suggestion F&O (FSS-B2-01) is written for an incorrect calculation reference (Reference 3 in Section 8) of the Fire Scenarios Report (0247-06-0006.05-U1, Rev. 0). The calculation reference is ANO2-FP-09-00011. The correct reference should be ANO1-FP-09-00011.	The 0.1 screening conditional core damage probability (CCDP) is no longer implemented in the ANO FPRA for the Model Change Request (MCR) abandonment scenario. Significant plant modifications have been implemented at ANO Units 1 and 2 as part of the transition to NFPA-805 for the MCR abandonment strategy. The new common feed pump control center has been installed along with other plant modifications and have been incorporated into the FPRA model. The CCDP/CLERP is calculated based on the summation of actual system failures and accident sequences in conjugation with detailed operator actions specific to the MCR abandonment scenario. Therefore, the scenario CCDP and conditional large early release probability (CLERP) for the abandonment scenario is calculated using the same method as every other fire scenario developed in the FPRA model and no screening CCDP or CLERP values are assumed in the ANO-1 FPRA scenario development. Therefore, this SR has been resolved in the model of record and has no impact on the applications and has been self-assessed at CC-II or greater.

Enclosure 1, Attachment 4

1CAN052102

External Hazards Screening

Attachment 2

External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Aircraft Impact	Y	PS2, PS4	Acceptance criterion 1.A of Standard Review Plan (SRP) 3.5.1.6 (Reference [38] of Enclosure 1) states the probability is considered to be less than an order of magnitude of 10^{-7} per year by inspection if the plant-to-airport distance D is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D^2, or the plant-to-airport distance D is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D^2. Per the ANO-1 Safety Analysis Report (SAR) (Reference [39] of Enclosure 1), there is no major airport with a control tower within 50 miles of the plant site. The closest airports are the Russellville Municipal Airport (8 miles) and the Clarksville Municipal Airport (15 miles). None of these airports has any regularly scheduled air traffic. Based on this review, the Aircaft impact hazard is considered to be negligible.
Avalanche	Y	C3	Per the Individual Plant Examinations of External Events (IPEEE) Report (Reference [40] of Enclosure 1), the topography is such that no avalanche is possible. Based on this review, the Avalanche hazard is considered to be negligible.
Biological Event	Y	C5	Per ANO-1 SAR Section 9.3.2 (Reference [39] of Enclosure 1), to help limit biological fouling such as flow blockage from bivalve mollusks, Corbicula (Asiatic clams), a biocide is added at the intake structure in sufficient concentration to kill the mollusks. Station procedures provide for addition of biocide in the service water system and emergency cooling pond. The service water intake bays are also inspected and cleaned at least once every refueling outage to prevent clam buildup and fouling. Flow measurement orifices and instrumentation has been added to several of the auxiliary building coolers. Flow measurements are periodically taken and trended to detect any possible developing flow blockage from biological fouling. Based on this review, the Biological Event hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Coastal Erosion	Y	C1, C3	Per the IPEEE Report (Reference [40] of Enclosure 1), the site is located 6 miles W-NW of Russellville, Arkansas, on the peninsula formed by the Dardanelle Reservoir on the Arkansas River. There are several flood control dams upstream and downstream of the plant; therefore, erosion is not a significant concern. In addition, per ANO-1 SAR Section 1.7.3 (Reference [39] of Enclosure 1), the emergency cooling pond is excavated in natural soil; therefore, erosion is limited by the natural topography of the site. Based on this review, the Coastal Erosion hazard is considered to be negligible.
Drought	Y	C1, C5	Per the IPEEE Report (Reference [40] of Enclosure 1), drought is not a concern at ANO. Cooling water is provided by the Dardanelle Reservoir and emergency cooling pond. In addition, drought is a slowly developing hazard allowing time for orderly plant reductions, including shutdowns. Based on this review, the Drought hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
External Flooding	Y	C1	The evaluation of the impact of the external flooding hazard at the site was updated as a result of the NRC's post-Fukushima 50.54(f) Request for Information. The ANO Units 1 and 2 flood hazard reevaluation report (FHRR) was submitted to NRC for review on September 14, 2016 (Reference [41] of Enclosure 1). The FHRR determined that the only location where water ingress may have potential to impact key structures, systems, or components (SSCs) was via the turbine building train bay doors due to local intense precipitation (LIP). By letter dated May 31, 2017, ANO submitted its focused evaluation (FE) (Reference [42] of Enclosure 1) for ANO Units 1 and 2. The FE demonstrated that no doors, buildings, or propagation pathways that contain key SSCs are impacted by floodwaters during the LIP event. The calculated ponding levels were below the controlling current design bases (CDB) event, which is a probable maximum flood (PMF) from the Arkansas River coincident with dam failure and wind-generated waves. Any other buildings that are inundated by floodwaters or the propagation of floodwaters do not contain any SSCs or equipment that would affect the ability to maintain any of the key safety functions required to achieve and maintain safe shutdown. This includes the Turbine Building. All vulnerabilities due to the unbounded LIP mechanism were addressed by permanent flooding protection and available physical margin was demonstrated to be adequate to protect SSCs required to achieve and maintain safe shudown. After its review of ANO FE (Reference [43] of Enclosure 1), the NRC concluded that the station demonstrated effective flood protection from the reevaluated flood hazards. Based on this review, the External Flooding hazard is considered to be negligible.

Extreme Wind or Tornado	Y/N	C1, PS2, PS4	Section 5.1.1 of the IPEEE (Reference [40] of Enclosure 1) documents the screening of High Winds. It was determined that the ANO design basis is mostly consistent with the 1975 SRP requirements (Reference [44] of Enclosure 1) for non-tornadic winds. The differences with the SRP were determined to be insignificant, primarily due to the fact that the ANO design is controlled by tornadic and not straight winds. Table 5.1-1 of the IPEEE provides a comparison of the ANO design basis tornado parameters to the requirements in Regulatory Guide (RG) 1.76 (Reference [45] of Enclosure 1). Key equipment and structures are designed to withstand a maximum wind speed of 300 mph, external pressure drop of 3 psi, and rate of pressure drop of 1 psi/sec. Additionally, key Category I components outside of Category I structures (e.g., diesel exhausts and certain tanks) were determined to be capable of withstanding the tornado effects (Reference [40] of Enclosure 1). The RG 1.76 criteria are higher for wind speed (360 mph) and rate of pressure drop of (2 psi/sec). The ANO design considers all Category I structures unvented; therefore, the rate of pressure drop is not relevant to the design (Reference [40] of Enclosure 1). However, the ANO design does not meet the criteria for the maximum wind speed. Tornado wind speed hazard curve information for ANO is provided in Table 6-1 of NUREG/CR-4461, Rev. 2 (Reference [46] of Enclosure 1). The wind speed for the 1E-7 annual exceedance probability is 297 mph, using the F-Scale, and 227 mph using the more recent EF-Scale. Therefore, the frequency of the design tornado wind speed for ANO is approximately equal to the 1E-7/yr (based on the conservative F-Scale), which is much less than 1E-6/yr. Tropical storms (i.e., hurricanes) are not a concern at ANO due its location (i.e., approximately 400 miles inland). Straight winds (e.g., due to thunderstorms) are typically in the 50 – 70 mph range, although in rare cases may be over 100 mph. However, the hazard curve for straight winds tails off very quickly, such that below approximately 1.0E-03/yr, straight winds do not affect the overall wind hazard for areas with hurricane and/or tornado hazards (Reference [47] of Enclosure 1). Therefore, the core damage frequency (CDF) contribution from wind speeds greater than 300 mph is less than 1E-6/yr, and the effects of high winds and tornados, with the exception of tornado missiles, can be screened. See Section 3.2.4 and associated Figure 3-1 for tornado missile hazards, which are not screened.
Fog	Y	C4	Per the IPEEE Report (Reference [40] of Enclosure 1), fog can increase the frequency of occurrence for other events such as aircraft, railway and highway accidents.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			Fog is implicitly included in data for other events such as aircraft, railway, and highway accidents which are discussed elsewhere in this other external hazards evaluation. Based on this review, the Fog hazard is considered to be negligible.
Forest or Range Fire	Y	C3	Per the IPEEE Report (Reference [40] of Enclosure 1), the ANO site is is cleared of significant forestry and brush, and therefore, forest or brush fire do not pose any danger. Based on this review, the Forest or Range Fire hazard is considered to be negligible.
Frost	Y	C1, C4	There is negligible impact on the plant due to frost. The worst-case impact is frost induced freezing leading to a loss of off-site power event which is addressed in the weather-related Loss of Offsite Power (LOOP) initiating event in the Full Power Internal Events (FPIE) probabilistic risk assessment (PRA) model for ANO. Based on this review, the Frost hazard is considered to be negligible.
Hail	Y	C4	Hail is bounded by other events for which the plant is designed. Per the IPEEE Report (Reference [40] of Enclosure 1), hail is less damaging than the tornado missile hazard. In addition, the principal effects of such events would be to cause a LOOP and are addressed in the weather-related LOOP initiating event in the FPIE PRA model for ANO. Based on this review, the Hail hazard is considered to be negligible.
High Summer Temperature	Y	C1, C5	Per NUREG-1407 (Reference [48] of Enclosure 1), the capacity reduction of the ultimate heat sink would be a slow process that allows plant operators sufficient time to take proper actions such as reducing power output level or achieving and maintaining safe shutdown. In addition, should the emergency cooling pond (ECP) discharge reach 100 °F, then plant Technical Specification (TS) 3.7.8 require actions to shutdown the plant. Based on this review, the High Summer Temperature hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
High Tide, Lake Level, or River Stage	Y	C4	Per the IPEEE Report (Reference [40] of Enclosure 1), the site is located 6 miles W-NW of Russellville, Arkansas, on the peninsula formed by the Dardanelle Reservoir on the Arkansas River. There are several flood control dams upstream and downstream of the plant. See also "External Flooding." Based on this review, the High Tide, Lake Level, or River Stage hazard is considered to be negligible.
Hurricane	Y	C4	Per the IPEEE Report (Reference [40] of Enclosure 1), hurricanes are bounded by the external flooding hazard and the high winds or tornados hazard. Additionally, hurricanes, lose strength as they move inland and the greatest concern is possible damage from winds or flooding due to excessive rainfall. See External Flooding and Extreme Winds and Tornado Assessment. Based on this review, the Hurricane hazard is considered to be negligible.
Ice Cover	Y	C1	Per the IPEEE (Reference [40] of Enclosure 1), ice formation in this portion of the Arkansas River basin is light and infrequent. Per ANO-1 SAR Section 9.3.2.4 (Reference [39] of Enclosure 1), possible layers of ice on the emergency cooling pond surface would not cause flow blockage of the cooling water system. Based on this review, the Ice Cover hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Industrial or Military Facility Accident	Y	C3, PS2	SRP Chapters 2.2.1-2.2.2 (Reference [49] of Enclosure 1) describes acceptance criteria for this hazard and states that NRC reviews should include all identified facilities and activities within 8 kilometers (5 miles) of the plant and that facilities and activities at distances greater than 8 kilometers (5 miles) should be considered if they have the potential for affecting plant safety-related features. Per IPEEE (Reference [40] of Enclosure 1), there are no military bases, missile sites, chemical plants and storage facilities, oil pipelines, or airports within a 5-mile radius of the centerline of the containment of ANO sites. Stationary offsite sources of hazardous materials were recently evaluated. Based on communication with the four counties within the 5-mile radius of the plant site, Pope County, Johnson County, Yell County, and Logan County, four facilities storing hazardous chemicals were identified in Pope County and the chemical information was obtained. All chemicals screened out as being non toxic, non-volatile, or were solid materials. Based on this review, the Industrial or Military Facility Accident hazard is considered to be negligible.
Internal Flooding	N/A	N/A	The ANO-1 Internal Events and Internal Flooding PRA model addresses risk from internal flooding events.
Internal Fire	N/A	N/A	The ANO-1 Internal Fire PRA model addresses risk from internal fires.
Landslide	Y	C3	Per the ANO-1 SAR Section 2.6.7 (Reference [39] of Enclosure 1), slope stability evaluation of the intake and discharge canals were performed. The factor of safety is 1.5 for normal condition and 1.0 for seismic condition was considered acceptable. Potential landslides are not a problem at the plant site. Based on this review, the Landslide hazard is considered to be negligible.
Lightning	Y	C1, C4	Lightning strikes may result in a LOOP or plant trip. These events are addressed in the plant design basis and are modeled in the ANO-1 Internal Events PRA model. Based on this review, the Lightning hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Low Lake Level or River Stage	Y	C1, C5	Per the IPEEE (Reference [40] of Enclosure 1), the station can obtain the required minimum cooling water from the Dardanelle Reservoir through the canals based on the low water level of 336 ft. At a water level of 335 ft., the plant will be shut down and the water source shifted to the emergency cooling pond. In addition, per the ANO-1 SAR Section 9.3.2.4 (Reference [39] of Enclosure 1), the average water depth of the pond is monitored daily to insure that it is greater than or equal to the minimum depth specified in the TSs. The depth is read from a permanently installed device in the pond and recorded in a log by plant personnel. Since changes should be small from day to day, more than sufficient time is available to observe dangerous trends, e.g. decreasing water depth, and take appropriate action. Consequently, high and low level alarms are not necessary. Based on this review, the Low Lake Level or River Stage hazard is considered to be negligible.
Low Winter Temperature	Y	C1, C5	Per the IPEEE (Reference [40] of Enclosure 1), for winter operation, the ECP is designed to perform its safety function with an initial ice layer on the pond surface. Based on this review, the Low Winter Temperature hazard is considered to be negligible.
Meteorite or Satellite Impact	Y	PS4	Per the IPEEE Report (Reference [40] of Enclosure 1), this event has a very low annual probability of occurrence, less than 1E-9 (Section 2.10 NUREG-1407, Reference [48] of Enclosure 1); therefore, is eliminated on the basis of low frequency. Based on this review, the Meteorite or Satellite hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Pipeline Accident	Y	PS4	Per the IPEEE Report (Reference [40] of Enclosure 1), there are no military installations, chemical plants, oil pipeline, or airports within 5 miles of the centerline of containment. However, there is a natural gas pipeline located 600 feet from the ANO-1 reactor, which was evaluated. Per ANO-1 SAR Section 2.2.6 (Reference [39] of Enclosure 1), it has been concluded that the proximity of the gas line represents no safety hazard to the safe operation of the plant. Additionally, per the ANO-2 SAR Section 2.2.2, (Reference [50] of Enclosure 1), the probability of a rupture of this gas pipeline and subsequent ignition of the gas is less than 1E-7 per year. Based on this review, the Pipeline Accident hazard is considered to be negligible.
Release of Chemicals in Onsite Storage	Y	C4, PS1, PS2	Per the IPEEE Report (Reference [40] of Enclosure 1), chemicals stored onsite were evaluated and an updated chemical hazardous survey was completed in February 2020. As stated in the updated survey, in the original plant design, chlorine was stored onsite in one-ton cylinders for use as a water biocide. All chlorine has since been removed from the site since biocides based upon use of hypochlorite or bromine are used. All chemicals stored onsite were evaluated in the updated survey and screened out consistent with RG 1.78 (Reference [51] of Enclosure 1). See also "Toxic Gas." Based on this review, the Release of Chemicals in Onsite Storage hazard is considered to be negligible.
River Diversion	Y	C1, C3	Per the IPEEE Report (Reference [40] of Enclosure 1), Upstream diversion/damming by land slide, ice blockage or other cause is unlikely. In the unlikely event of upstream diversion or natural damming of the Arkansas River by landslide, ice blockage, or other causes, there would be sufficient storage in Dardanelle Reservoir to permit normal plant shutdown. Based on this review, the River Diversion hazard is considered to be negligible.
Sand or Dust Storm	Y	C3	Per IPEEE Report (Reference [40]), a sandstorm hazard is not relevant at ANO Site. Based on this review, the Sand or Dust Storm hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Seiche	Y	C3	Per IPEEE Report (Reference [40] of Enclosure 1), the Dardanelle Reservoir is not of sufficient size to be affected by surge or seiche flooding. See also "External Flooding." Based on this review, the Seiche hazard is considered to be negligible.
Seismic Activity	N/A	N/A	See Section 3.2.3 and Figure A4-1 in this Attachment.
Snow	Y	C1, C4	Per IPEEE Report (Reference [40] of Enclosure 1), the roofs of all structures are designed for a conservative snow load of 20 psf. Snow storms may also result in loss of offsite power or plant trip. These events are addressed in the plant design basis and are modeled in the ANO-1 Internal Events PRA model. See also "External Flooding." Based on this review, the Snow hazard is considered to be negligible.
Soil Shrink-Swell Consolidation	Y	C1	Per the ANO-1 SAR Section 2.6 (Reference [39] of Enclosure 1), various investigations were performed to define site foundation conditions and regional and site geologic, geohydrologic, and seismological conditions. As a result of the investigations performed, it was concluded that geologic, seismologic, and foundation conditions at the ANO site are adequate in all respects. Based on this review, the Soil Shrink-Swell Consolidation hazard can be considered to be negligible.
Storm Surge	Y	C3	The ANO site located on a peninsula of the Dardanelle Reservoir is not of sufficient size to be affected by surge or seiche flooding. Based on this review, the Storm Surge hazard is considered to be negligible.
Toxic Gas	Y	C4	Toxic gas is covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident. In addition, station procedures are established to address periodic control room habitability self-assessments. Based on this review, the Toxic Gas hazard is considered to be negligible.

External Hazard	Screening Result		
	Screened? (Y/N)	**Screening Criterion (Note a)**	**Comment**
Transportation Accident	Y	PS2, PS4	An updated evaluation was performed for transporation (mobile) accidents that could impact the site. Mobile offsite sources evaluated include barge traffic, rail traffic, and highway traffic. The total release frequency was less than 1E-6/yr. No specific plant vulnerabilities were identified. Based on this review, the Transportation Accidents hazard is considered to be negligible.
Tsunami	Y	C3	The location of ANO site located on a peninsula in Lake Dardanelle (Dardanelle Reservoir) precludes the possibility of a tsunami. Based on this review, the Tsunami hazard is considered to be negligible.
Turbine-Generated Missiles	Y	C1, PS4	Per the IPEEE Report (Reference [40] of Enclosure 1), the annual probability of turbine generated missiles is less than 1.1E-8. In addition, per SAR Section 14.1.2.9.5 (Reference [39] of Enclosure 1), any missile resulting from a turbine-generator overspeed incident is not considered credible. The Unit 2 SAR (Reference [50] of Enclosure 1) discusses the probabilites of missiles generated for the Unit 1 and Unit 2 turbines in Table 3.5-3. The annual probability of missile generation for the Unit 1 turbine is about 1E-08. Based on this review, the Turbine-Generated Missiles hazard is considered to be negligible.
Volcanic Activity	Y	C3	There are no active or dormant volcanoes located near the plant site. Based on this review, the Volcanic Activity hazard is considered to be negligible.
Waves	Y	C1, C4	Waves are bounded by other hazards that are considered and screen out (e.g., seiche). See also “External Flooding." Based on this review, the Waves hazard is considered to be negligible.
Note a – See Attachment 5 for descriptions of the screening criteria.			

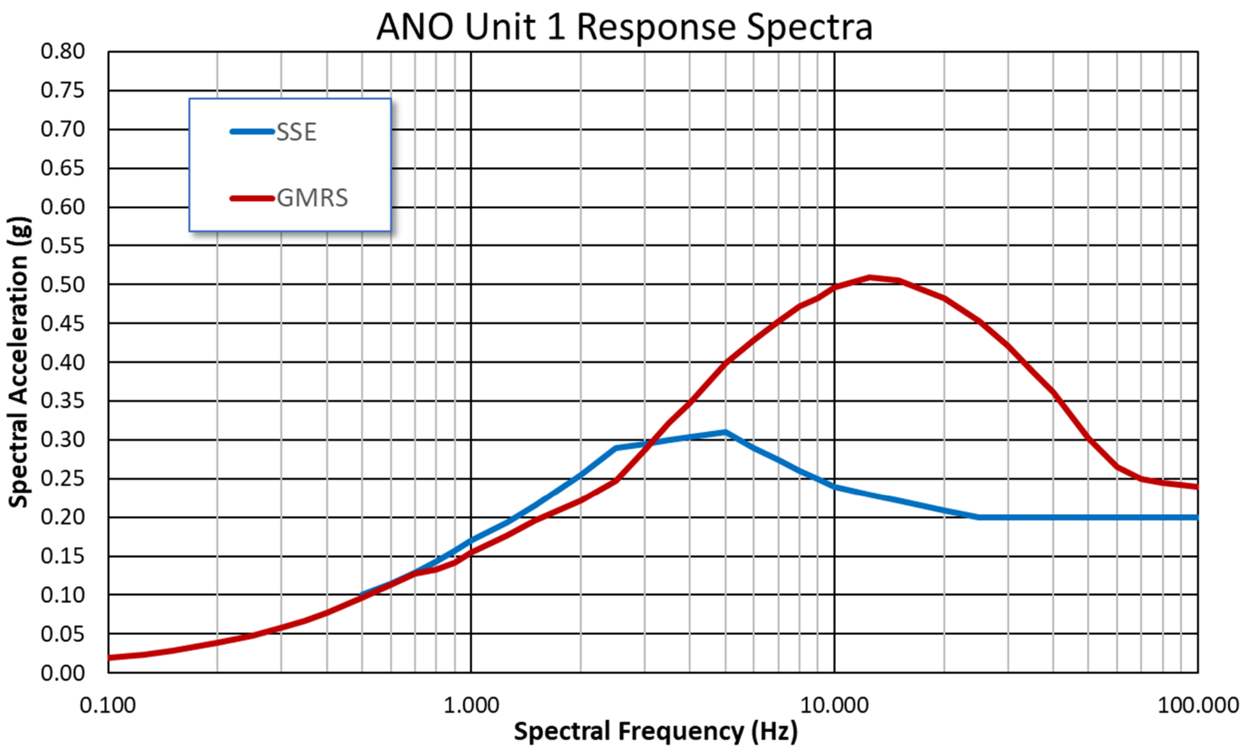

Figure A4-1

Ground Motion Response Spectra (GMRS) and Safe Shutdown Earthquake (SSE) Response Spectra for ANO-1

(From Reference [21] of Enclosure 1)

Enclosure 1, Attachment 5

1CAN052102

Progressive Screening Approach for Addressing External Hazards

Attachment 5

Progressive Screening Approach for Addressing External Hazards

Event Analysis	Criterion	Source	Comments
Initial Preliminary Screening	C1. Event damage potential is < events for which plant is designed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C2. Event has lower mean frequency and no worse consequences than other events analyzed.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C3. Event cannot occur close enough to the plant to affect it.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	
	C4. Event is included in the definition of another event.	NUREG/CR-2300 and ASME/ANS Standard RA-Sa-2009	Not used to screen. Used only to include within another event.
	C5. Event develops slowly, allowing adequate time to eliminate or mitigate the threat.	ASME/ANS Standard RA-Sa-2009	
Progressive Screening	PS1. Design basis hazard cannot cause a core damage accident.	ASME/ANS Standard RA-Sa-2009	
	PS2. Design basis for the event meets the criteria in the NRC 1975 Standard Review Plan (SRP).	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
	PS3. Design basis event mean frequency is < 1E-5/y and the mean conditional core damage probability is < 0.1.	NUREG-1407 as modified in ASME/ANS Standard RA-Sa-2009	
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Enclosure 1, Attachment 6

1CAN052102

Disposition of Key Assumptions/Sources of Uncertainty

Attachment 6

Disposition of Key Assumptions/Sources of Uncertainty

The ANO-1 internal events (IEs) and fire probabilistic risk assessment (PRA) models and documentation were reviewed for plant-specific modeling assumptions and related sources of uncertainty. The ANO-1 Sources of Uncertainty Reports document the sources of PRA modeling uncertainty. The reports identify assumptions and determine if those assumptions are related to sources of model uncertainty and characterize that uncertainty, as necessary. The identified uncertainties were reviewed for this application.

Each PRA model includes an evaluation of the potential sources of uncertainty for the base case models using the approach that is consistent with the ASME/ANS RA-Sa-2009 (Reference [52] of Enclosure 1) requirements for identification and characterization of uncertainties and assumptions. This evaluation identifies those sources of uncertainty that are important to the PRA results and may be important to PRA applications which meets the intent of steps C-1 and E-1 of NUREG-1855, Revision 1 (Reference [53] of Enclosure 1).

The results of the base PRA evaluations were reviewed to determine which potential uncertainties could impact the 10 CFR 50.69 categorization process results. This evaluation meets the intent of the screening portion of steps C-2 and E-2 of NUREG-1855, Revision 1.

For the 10 CFR 50.69 Program, the guidance in Nuclear Entergy Institute (NEI) 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0 (Reference [1] of Enclosure 1) specifies sensitivity studies to be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the structure, system, or components (SSCs) importance. Regulatory Guide (RG) 1.174, Revision 3 (Reference [54] of Enclosure 1), cites NUREG-1855, Revision 1, as related guidance. In Section B of RG 1.174, Revision 3, the guidance acknowledges specific revisions of NUREG-1855 to include changes associated with expanding the discussion of uncertainties. The results of the evaluation of PRA model sources of uncertainty as described above are evaluated relative to the 10 CFR 50.69 application in Attachment 6 to determine if additional sensitivity evaluations are needed.

> Note: The ANO-1 Fire PRA was developed using consensus methods outlined in NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities" (Reference [55] of Enclosure 1), and interpretations of technical approaches as required by NRC. Fire PRA methods were based on NUREG/CR-6850, other more recent NUREGs (e.g., NUREG-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)" (Reference [56] of Enclosure 1), and published "frequently asked questions" (FAQs) for the Fire PRA.

The key sources of uncertainties identified in this Attachment do not present a significant impact on the ANO-1 10 CFR 50.69 application, and therefore, the PRA models are capable of producing accurate 10 CFR 50.69 importance measure results.

Internal Events / Internal Flooding PRA Model Sources of Uncertainty

IE / Internal Flooding (IF) PRA Sources of Assumption/ Uncertainty	IE / IF PRA 10 CFR 50.69 Impact	IE / IF PRA Model Sensitivity and Disposition (10 CFR 50.69)
Detailed evaluations of human error probabilities (HEPs) are performed for the risk significant human failure events (HFEs) using industry consensus methods. Mean values are used for the modeled HEPs. Uncertainty associated with the mean values can have an impact on core damage frequency (CDF) and Large Early Release Frequency (LERF) results.	This uncertainty potentially affects all SSCs that use an operator action as a surrogate for a modeled component.	As directed by NEI 00-04, human failure events are increased to the 95^{th} percentile and also decreased to the 5^{th} percentile values as part of the required 10 CFR 50.69 PRA categorization sensitivity activities. These results are capable of driving a component and its respective functions to high safety significance (HSS) and, therefore, the uncertainty of the HFEs are accounted for in the categorization process.
Common cause failures are developed using available industry data	This uncertainty potentially affects all SSCs evaluated during 10 CFR 50.69 categorization.	As directed by NEI 00-04, common cause basic events are increased to the 95^{th} percentile and also decreased to the 5^{th} percentile values as part of the required 10 CFR 50.69 PRA categorization sensitivity activities. These results are capable of driving a component and its respective functions to HSS and, therefore, the uncertainty of the common cause failure probabilities are accounted for in the categorization process.
FLEX[2] Equipment Credit	The FLEX Feed Pump is credited in Unit 1 for extended loss of AC power (ELAP) cases in the Full Power internal Events Model. (Note: The FPRA has not incorporated FLEX in the Unit 1 model.)	A sensitivity study was performed removing credit for the FLEX Feed Pump. The effects of this sensitivity case only applied to loss of offsite power (LOOP) / loss of offsite power (LOSP) sequences. The results of the sensitivity showed a marginal increase in risk <2%.

[2] Diverse and **Flexible** Mitigation Capability, or **FLEX**

Fire PRA (FPRA) Model Sources of Uncertainty

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 1 - Analysis Boundary and Partitioning	This task poses a limited source of uncertainty beyond the credit taken for boundaries and partitions. Task 1 establishes the overall spatial scope of the analysis and provides a framework for organizing the data for the analysis. The partitioning features credited are required to satisfy established industry standards.	Based on a review of the assumptions and potential sources of sources of uncertainly associated with this element, it is concluded that the methodology for the Analysis Boundary and Partitioning task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 2 - Fire PRA Component Selection	This task involves the selection of components to be treated in the analysis in the context of initiating events and mitigation. The potential sources of uncertainty include those inherent in the internal events PRA model as that model provides the foundation for the FPRA. The mapping of basic events to components requires not only the consideration of failure modes (active versus passive) but an understanding of the fire function / PRA component functions not previously considered risk significant in the Full Power Internal Events (FPIE) model. When performed correctly, the only uncertainty not already captured in the FPIE model is related to the Multiple Spurious Operation (MSO) process.	In the context of the FPRA, the uncertainty that is unique to the analysis is related to initiating event identification. However, that impact is minimized through use of the Pressurized Water Reactor Owners Group (PWROG) Generic MSO list and the process used to identify and assess potential MSOs. Based on the discussion of sources of uncertainty and the discussion above, it is concluded that the methodology for the Component Selection task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 3 - Cable Selection	The selection of cables to be considered in the analysis is identified using industry guidance documents. No treatment of uncertainty is typically required for Task 3 beyond the understanding of the cable selection approach (i.e., mapping an active basic event to a passive component for which power cables were not selected). Additionally, PRA credited components for which cable routing information was not provided represent a source of uncertainty (conservatism) in that components whose cable locations are not explicitly modeled (i.e. "UNL" components) could be assumed failed unnecessarily.	The results of a sensitivity performed for the transition to National Fire Protection Association (NFPA) 805 (Reference [57] of Enclosure 1) showed that the methodology for this task did not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 4 - Qualitative Screening	Qualitative screening was not performed; however, structures were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the Fire PRA were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip.	In the event a structure (location) which could result in a plant trip was incorrectly excluded, its contribution to CDF would be small (with a conditional core damage probability (CCDP) commensurate with base risk). Such a location would have a negligible risk contribution to the overall FPRA. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Qualitative Screening task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 5 - Fire-Induced Risk Model	The methodology used to develop the FPRA plant response model is consistent with the standard that used for the internal events PRA model development and was subjected to industry Peer Review. The PRM model is applied in such a fashion that all postulated fires are assumed to generate a plant trip. This represents a source of uncertainty, as it is not necessarily clear that fires would result in a trip. In the event the fire results in damage to cables and/or equipment identified in Task 2, the PRA model includes structure to translate them into the appropriate induced initiator.	The identified source of uncertainty could result in the over-estimation of fire risk. In general, the FPRA development process would have reviewed significant fire initiating events and performed supplemental assessments to address this possible source of uncertainty. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire-Induced Risk Model task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 6 - Fire Ignition Frequencies	Ignition source counting is an area with inherent uncertainty; however, the results are not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the frequency values from NUREG-2169 (Reference [58] of Enclosure 1) which result in uncertainty due to variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates, based on limited fire events and fire test data.	The ANO-1 FPRA utilized the bin frequencies from NUREG-2169. Consensus approaches are employed in the model. Based on a review of the assumptions and potential sources of uncertainty related to this element it is concluded that the methodology for the Fire Ignition Frequency task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 7 - Quantitative Screening	Other than screening out potentially risk significant scenarios (ignition sources), this task is not a source of uncertainty.	Quantitative screening criteria was defined for the ANO-1 FPRA as the CDF / LERF contribution of zero, such that all quantified fire scenarios are retained. All of the results were retained in the cumulative CDF / LERF; therefore, no uncertainty was introduced as a result of this task. Based on the discussion above, it is concluded that the methodology for the Quantitative Screening task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 8 - Scoping Fire Modeling	The approach taken for this task included: 1) The use of generic fire modeling treatments in lieu of conservative scoping analysis techniques; and, 2) Limited detailed fire modeling was performed to refine the scenarios developed using the generic fire modeling solutions. The primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR-6850 (Reference [55] of Enclosure 1).	Detailed fire modeling was applied to risk significant scenarios where the reduction in conservatism was likely to have a measurable impact. Consensus modeling approach is used for the Fire Modeling tasks and it is concluded that the methodology does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the ANO-1 50.69 application.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 9 - Detailed Circuit Failure Analysis	The circuit analysis is performed using standard electrical engineering principles. However, the behavior of electrical insulation properties and the response of electrical circuits to fire induced failures is a potential source of uncertainty. This uncertainty is associated with the dynamics of fire and the inability to ascertain the relative timing of circuit failures. The analysis methodology assumes failures would occur in the worst possible configuration, or if multiple circuits are involved, at whatever relative timing is required to cause a bounding worst-case outcome. This results in a skewing of the risk estimates such that they are over-estimated.	Circuit analysis was performed as part of the deterministic post fire safe shutdown analysis. Refinements in the application of the circuit analysis results to the FPRA were performed on a case-by-case basis where the scenario risk quantification was large enough to warrant further detailed analysis. Hot short probabilities and hot short duration probabilities as defined in NUREG-7150, Volume 2, based on actual fire test data, were used in the ANO-1 FPRA. The uncertainty (conservatism) which may remain in the FPRA is associated with scenarios that do not contribute significantly to the overall fire risk. Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Detailed Circuit Failure Analysis task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 10 - Circuit Failure Model Likelihood Analysis	One of the failure modes for a circuit (cable) given fire induced failure is a hot short. A conditional probability and a hot short duration probability are assigned using industry guidance published in NUREG-7150, Volume 2 (Reference [59] of Enclosure 1). The uncertainty values specified in NUREG-7150, Volume 2 are based on fire test data.	The use of hot short failure probability and duration probability is based on fire test data and associated consensus methodology published in NUREG-7150, Volume 2 (Reference [59] of Enclosure 1). Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Circuit Failure Mode Likelihood Analysis task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 11 - Detailed Fire Modeling	The application of fire modeling technology is used in the FPRA to translate a fire initiating event into a set of consequences (fire induced failures). The performance of the analysis requires a number of key input parameters. These input parameters include the heat release rate (HRR) for the fire, the growth rate, the damage threshold for the targets, and response of plant staff (detection, fire control, fire suppression). The fire modeling methodology itself is largely empirical in some respects and consequently is another source of uncertainty. For a given set of input parameters, the fire modeling results (temperatures as a function of distance from the fire) are characterized as having some distribution (aleatory uncertainty). The epistemic uncertainty arises from the selection of the input parameters (specifically the HRR and growth rate) and how the parameters are related to the fire initiating event. While industry guidance is available, that guidance is derived from laboratory tests and may not necessarily be representative of randomly occurring events. The fire modeling results using these input parameters are used to identify a zone of influence (ZOI) for the fire and cables/equipment within that ZOI are assumed to be damaged. In general, the guidance provided for the treatment of fires is conservative and the application of that guidance retains that conservatism. The resulting risk estimates are also conservative.	Consensus modeling approach is used for Detailed Fire Modeling and it is concluded that the methodology for the Detailed Fire Modeling task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 12 - Post-Fire Human Reliability Analysis	HEPs represent a potentially large uncertainty for the FPRA given the importance of human actions in the base model. Since many of the HEP values were adjusted for fire, the joint dependency multipliers developed for the FPIE model also represent a potential for introducing a degree of conservatism. The HEPs included the consideration of degradation or loss of necessary cues due to fire. Given the methodology used, the impact of any remaining uncertainties is expected to be small.	The HEPs include the consideration of degradation or loss of necessary cues due to fire. The fire risk importance measures indicate that the results are somewhat sensitive to Human Reliability Analysis (HRA) model and parameter values. The ANO-1 FPRA model HRA is based on industry consensus modeling approaches for its HEP calculations, so this is not considered a significant source of epistemic uncertainty. Additionally, for the 10 CFR 50.69 program, the guidance in NEI 00-04 (Reference [1]) specifies that certain sensitivity studies be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, maintenance probabilities, and manual suppression probabilities for fire) do not mask the SSC(s) importance. It is concluded that the methodology for the Post-Fire Human Reliability Analysis task does not introduce any epistemic uncertainties that would require sensitivity treatment. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 13 - Seismic-Fire Interactions Assessment	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	The qualitative assessment of seismic-induced fires should not be a source of model uncertainty as it is not expected to provide changes to the quantified FPRA model. Based on the discussion above, it is concluded that the methodology for the Seismic-Fire Interactions Assessment task does not introduce any epistemic uncertainties that affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Task 14 - Fire Risk Quantification	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit.	The selected truncation was confirmed to be consistent with the requirements of the PRA Standard (Reference [52] of Enclosure 1). Based on a review of the assumptions and potential sources of uncertainty related to this element and the discussion above, it is concluded that the methodology for the Fire Risk Quantification task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 15 - Uncertainty and Sensitivity Analyses	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources - of uncertainty.	This task does not introduce any new uncertainties. This task is intended to address how the fire risk assessment could be impacted by the various sources of uncertainty. Additionally, for the 10 CFR 50.69 program, the guidance in NEI 00-04 (Reference [1] of Enclosure 1) specifies that certain sensitivity studies be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, maintenance probabilities, and manual suppression probabilities for fire) do not mask the SSC(s) importance. Based on the discussion above, it is concluded that the methodology for the Uncertainty and Sensitivity Analyses task does not introduce any epistemic uncertainties that would affect the 10 CFR 50.69 program. Therefore, this does not represent a key source of uncertainty for the ANO-1 10 CFR 50.69 application.
Task 16 - Fire PRA Documentation	The FPRA Documentation task does not introduce any new uncertainties to the fire risk.	This task does not introduce any new uncertainties to the fire risk as it outlines documentation requirements.

FPRA Description	FPRA Sources of Uncertainty	FPRA Disposition
Very Early Warning Fire Detection System (VEWFDS)	Installed in Unit 2 (only) in key electrical cabinets. Procedures are established to address system operation and response.	Credit in the FPRA was removed during the NFPA-805 approval process given that NUREG-2180 (Reference [60] of Enclosure 1) was not published.

Enclosure 2

1CAN052102

Proposed Operating License
(mark-up)
(3 pages)

(10) Upon implementation of Amendment 239 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.4, in accordance with Specifications 5.5.5.c.(i), 5.5.5.c.(ii), and 5.5.5.d, shall be considered met. Following implementation:

1. The first performance of SR 3.7.9.4, in accordance with Specification 5.5.5.c.(i), shall be within 15 months of the approval of TSTF-448. SR 3.0.2 will not be applicable to this first performance.

2. The first performance of the periodic assessment of CRE habitability, Specification 5.5.5.c.(ii), shall be within 15 months of the approval of TSTF-448. SR 3.0.2 will not be applicable to this first performance.

3. The first performance of the periodic measurement of CRE pressure, Specification 5.5.5.d, shall be within 15 months of the approval of TSTF-448. SR 3.0.2 will not be applicable to this first performance.

(11) 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors

Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire; and the high wind / tornado safe shutdown equipment list to evaluate high wind / tornado missile events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the ANO-1 passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and the associated supports; the results of the non-PRA evaluations that are based on the Individual Plant Examination for External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in the Entergy submittal letter dated [DATE], and all its subsequent associated supplements, as specified in License Amendment No. [XXX] dated [DATE].

Prior NRC approval, in accordance with 10 CFR 50.90, will be requested if ANO-1's feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69.

Move to new Pg 9

3. This renewed license is effective as of the date of issuance and shall expire at midnight, May 20, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

Renewed License No. DPR-51
Amendment No. ~~256~~,

Move to new Pg 9

Original Signed by:
Jon R. Johnson

Jon R. Johnson, Acting Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A - Technical Specifications and
Technical Specifications Bases (ML011710071 and ML011710100)

Date of Issuance: June 20, 2001

Moved from Pg 8

3. This renewed license is effective as of the date of issuance and shall expire at midnight, May 20, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Jon R. Johnson

Jon R. Johnson, Acting Director
Office of Nuclear Reactor Regulation

Attachment:
Appendix A - Technical Specifications and
Technical Specifications Bases (ML011710071 and ML011710100)

Date of Issuance: June 20, 2001

Enclosure 3

1CAN052102

Revised Operating License
(2 pages)

(10) Upon implementation of Amendment 239 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.9.4, in accordance with Specifications 5.5.5.c.(i), 5.5.5.c.(ii), and 5.5.5.d, shall be considered met. Following implementation:

1. The first performance of SR 3.7.9.4, in accordance with Specification 5.5.5.c.(i), shall be within 15 months of the approval of TSTF-448. SR 3.0.2 will not be applicable to this first performance.

2. The first performance of the periodic assessment of CRE habitability, Specification 5.5.5.c.(ii), shall be within 15 months of the approval of TSTF-448. SR 3.0.2 will not be applicable to this first performance.

3. The first performance of the periodic measurement of CRE pressure, Specification 5.5.5.d, shall be within 15 months of the approval of TSTF-448. SR 3.0.2 will not be applicable to this first performance.

(11) 10 CFR 50.69, Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors

Entergy is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 Structures, Systems, and Components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire; and the high wind / tornado safe shutdown equipment list to evaluate high wind / tornado missile events; the NUMARC 91-06 shutdown safety assessment process to assess shutdown risk; the ANO-1 passive categorization method to assess passive component risk for Class 2 and Class 3 SSCs and the associated supports; the results of the non-PRA evaluations that are based on the Individual Plant Examination for External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards except seismic; and the alternative seismic approach as described in the Entergy submittal letter dated [DATE], and all its subsequent associated supplements, as specified in License Amendment No. [XXX] dated [DATE].

Prior NRC approval, in accordance with 10 CFR 50.90, will be requested if ANO-1's feedback process determines that a process different from the proposed alternative seismic approach is warranted for seismic risk consideration in categorization under 10 CFR 50.69.

3. This renewed license is effective as of the date of issuance and shall expire at midnight, May 20, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Jon R. Johnson

Jon R. Johnson, Acting Director
Office of Nuclear Reactor Regulation

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Appendix A - Technical Specifications and
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Date of Issuance: June 20, 2001