NRR-DMPSPEm Resource

From: Miller, Ed
Sent: Monday, January 22, 2018 10:11 AM
To: Miller, Ed
Subject: Slides for Jan 18, 2018, Public Meeting
Attachments: 50.69 License Amendment Request Template R4.pdf

Attached is the current version of the template LAR provided for the January 28, 2018, public meeting to discuss 10 CFR 50.69 LARs.

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10 CFR 50.69 LICENSE AMENDMENT REQUEST TEMPLATE

The purpose of this document is to provide a streamlined template for licensees to utilize when preparing a 10 CFR 50.69 application submittal. It is intended that a license amendment request (LAR) that follows this template conforms to the requirements of 10 CFR 50.69(b)(2) and 50.90. 10 CFR 50.69(b)(2) 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).

The above requirements are detailed and addressed in the technical evaluation section of this template. The intent of this template is to be concise but comprehensive as well as flexible. Below is an explanation of the different levels of guidance provided by this template, their intent and how they are formatted throughout the document.

Generically Applicable Text: This text is intended to be used in all cases

Optional Text: This text intended to be used optionally depending on whether it reflects the situation of the licensee.

[Licensee To Insert Text]: This text is intended to identify where the licensee should insert plant specific information. These place holders should be deleted prior to the completion of the submittal.

Example Text: This text is intended to only provide guidance on the level of detail expected in the plant specific information. This text should be deleted prior to the completion of the submittal.

Preparer Notes: This text is intended to provide additional guidance to the preparer of the license amendment request. This text should be deleted prior to the completion of the submittal.

[DATE] 10 CFR 50.90
10 CFR 50.69

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
[PLANT NAME] / [UNIT NOS.]
[Docket No(s) [50-[xxx], 50-[xxx]]

SUBJECT: Application to Adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"

In accordance with the provisions of 10 CFR 50.69 and 10 CFR 50.90, [LICENSEE] is requesting an amendment to the license of [PLANT NAME, UNIT NOS.].

The proposed amendment would modify the [PLANT NAME] licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, "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.

The enclosure to this letter provides the basis for the proposed change to the [PLANT NAME, UNIT NOS.] Operating Licenses. 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.

[PREPARER'S NOTE: If applicable, one of the three following paragraphs is recommended in order to provide assurance to the NRC that the submittal of 10 CFR 50.69 is NOT "linked" to other submittals. Also, this can also serve as a suggestion to the NRC that it is possible to streamline the review of the PRA model in this application

using the approval from a previous risk-informed application such as TSTF-505 or TSTF 425 or streamline the review of the PRA model for a future submittal that will be utilizing the same models. This discussion is also included in Section 3 of the Enclosure.]

Though routine maintenance updates have been applied, the NRC has previously reviewed the technical adequacy of the [PLANT NAME] Probabilistic Risk Assessment (PRA) model identified in this application for:

- [purpose] in [identify previous application where the PRA model technical adequacy was reviewed by the NRC, including date and ADAMS Accession Number].
- [List any additional applications using these models]

[LICENSEE] requests that the NRC utilize the review of the PRA technical adequacy for that application when performing the review for this application.

Or

[LICENSEE] intends to submit a separate license amendment request for [identify application] within the next [X months] using the same PRA model[s] described in this Enclosure. [LICENSEE] requests that the NRC coordinate their review of the PRA technical adequacy description in Section 3.2 and 3.3 of this enclosure for both applications. This would reduce the number of [LICENSEE] and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

Or

The PRA model[s] described within this LAR are the same as those described within the [LICENSEE] submittal of the LAR dated [DATE] for [identify application] (ADAMS Accession Number [ML NUMBER]), with routine maintenance updates applied. [LICENSEE] requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of [LICENSEE] and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

[LICENSEE] requests approval of the proposed license amendment by [DATE], with the amendment being implemented [BY DATE OR WITHIN X DAYS].

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the [designated STATE Official].

In accordance with 10 CFR 50.30(b), a license amendment request must be executed in a signed original under oath or affirmation. This can be accomplished by attaching a notarized affidavit confirming the signature authority of the signatory, or by including the following statement in the cover letter. The alternative statement is pursuant to 28 USC 1746. It does not require notarization.

This letter contains no regulatory commitments *OR* a License Condition and/or regulatory commitments described in Attachment 1 to the Enclosure."

If you should have any questions regarding this submittal, please contact [NAME, TELEPHONE NUMBER].

I declare under penalty of perjury that the foregoing is true and correct. Executed on [DATE].

Sincerely,

Signature

Enclosure:

1. Evaluation of the Proposed Change

cc: [NRC Project Manager
NRC Regional Office
NRC Resident Inspector
State Contact]

Enclosure
Evaluation of the Proposed Change

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1 SUMMARY DESCRIPTION

The proposed amendment modifies the licensing basis to allow for the implementation of the provisions of 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." 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.

2 DETAILED DESCRIPTION

2.1 CURRENT REGULATORY REQUIREMENTS

The Nuclear Regulatory Commission (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 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 their 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 [LICENSEE] to improve focus on equipment that has safety significance resulting in improved plant safety.

2.3 DESCRIPTION OF THE PROPOSED CHANGE

[LICENSEE] proposes the addition of the following condition to the renewed operating license[s] of [PLANT/UNIT] to document the NRC's approval of the use 10 CFR 50.69.

> [LICENSEE] 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) specified in the license amendment dated [DATE].
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach).
>
> *If any items from Attachment 1 should apply as license conditions*
> [LICENSEE] shall complete the numbered items [Identify Items] listed in Attachment 1, List of Categorization Prerequisites, of [LICENSEE] letter [ML Number], dated [DATE], prior to implementation.

3 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.

[PREPARER'S NOTE: If applicable, one of the three following paragraphs is recommended in order to provide assurance to the NRC that the submittal of 10 CFR 50.69 is NOT "linked" to other submittals. Also, this can serve as a suggestion to the NRC that it is possible to streamline the review of the PRA model in this application using the approval from a previous risk-informed application such as TSTF-505 or TSTF 425 or streamline the review of the PRA model for a future submittal that will be utilizing the same models. This is a duplicate of the preparer's note in the cover letter].

Though routine maintenance updates have been applied, the NRC has previously reviewed the technical adequacy of the [PLANT NAME] PRA model identified in this application for:

- [purpose] in [identify previous application where the PRA model technical adequacy was reviewed by the NRC, including date and ADAMS Accession

Number]. [LICENSEE] requests that the NRC utilize the review of the PRA technical adequacy for that application when performing the review for this application.

- [List any additional applications using these models]

Or

[LICENSEE] intends to submit a separate license amendment request for [identify application] within the next [X months] using the same PRA model[s] described in this Enclosure. [LICENSEE] requests that the NRC coordinate their review of the PRA technical adequacy description in Section 3.2 and 3.3 of this enclosure for both applications. This would reduce the number of [LICENSEE] and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

Or

The PRA model[s] described within this LAR are the same as those described within the [LICENSEE] submittal of the LAR dated [DATE] for [identify application] (ADAMS Accession Number [ML NUMBER]), with the same routine maintenance updates applied. [LICENSEE] requests that the NRC conduct their review of the PRA technical adequacy details for this application in coordination with the review of the application currently in-process. This would reduce the number of [LICENSEE] and NRC resources necessary to complete the review of the applications. This request should not be considered a linked requested licensing action (RLA), as the details of the PRA models in each LAR are complete which will allow the NRC staff to independently review and approve each LAR on their own merits without regard to the results from the review of the other.

3.1 CATEGORIZATION PROCESS DESCRIPTION (10 CFR 50.69(b)(2)(i))

3.1.1 Overall Categorization Process

[LICENSEE] 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, "10 CFR 50.69 SSC Categorization Guideline," as endorsed by RG 1.201. 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 nor 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 they are all completed they may even be performed in parallel. Note that NEI 00-04 only requires Item 3 to be completed for components/functions categorized as 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 (SSEL), seismic safe shutdown equipment list (SSEL), 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

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., High Safety Significant (HSS) or Low Safety Significant (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 Risk Informed Safety Class (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: IDP Changes from Preliminary HSS to LSS

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, Seismic and Other External Hazards –	Component	Not 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	N/A
Passive	Passive – Section 4	Segment/Component	Not Allowed	No

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). 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. Therefore, if a HSS component is mapped to a 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.

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

- The Integrated Decision-Making Panel (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 who has 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 safety significant or low safety-significant pursuant to § 50.69(f)(1) will be documented in [LICENSEE] procedures. Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. If a resolution cannot be achieved concerning the safety significance of an SSC, then the SSC will be classified as safety-significant.
- 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 Vogtle SER (Reference X) 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 Low Safety Significant (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, [LICENSEE] will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

Deleted: <#>[LICENSEE] will require that if any SSC is identified as high safety significant (HSS) from either the integra PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6 of NEI 00-04), the associated system function(s) would be identified as HSS.¶

The following are the exceptions taken to the NEI 00-04 categorization process:

- NEI 00-04, Section 5.2 states that the fire safety significance process takes one of two forms. Either the use of Fire Induced Vulnerability Evaluation (FIVE) or a Fire PRA. However, Section 3.2.2 of this LAR describes an alternate approach, which implements the Appendix R Safety Shutdown analysis that will be used in the [LICENSEE] categorization process to evaluate safety significance related to the fire hazard.

- [Example: Use of another of risk approach not described in NEI 00-04]

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

- Internal Event Risks: Internal events including internal flooding PRA model version [utility version and date] [accepted by NRC for TSTF 505 or other application, date, ML # (Reference X)].
- Fire Risks: Fire Safe Shutdown Equipment List (SSEL). *OR* Fire PRA model version [utility version and date] [accepted by NRC for NFPA 805 or other application dated xx, ML # (Reference X)].
- Seismic Risks: Success Path Component List (SPCL) from the IPEEE seismic analysis [accepted by NRC SER dated xx, ML # (Reference X)] *OR* Seismic PRA model version [utility version and date].

- Other External Risks (e.g., tornados, external floods): External [hazard] PRA model version [utility version and date]. *AND/OR* Using the IPEEE screening process as approved by NRC SER dated [dated xx, ML # (Reference X)] 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 3), which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach) will not be used without prior NRC approval. 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 Arkansas Nuclear One (ANO) Risk-Informed Repair/Replacement Activities (RI-RRA) methodology contained in Reference 5 (ML090930246) consistent with the related Safety Evaluation Report (SER) issued by the Office of Nuclear Reactor Regulation.

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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 their 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., DID, 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.

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The use of this method was previously approved to be used for a 10 CFR 50.69 application by NRC in the final Safety Evaluation for Vogtle dated December 17, 2014 (Reference X). 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. Therefore, the RI-RRA methodology for passive categorization is acceptable and appropriate for use at [PLANT NAME] for 10 CFR 50.69.

The methodology does not require modification in order to appropriately categorize Class 1 SSCs. The ASME classification of the SSC does not impact the methodology as it only evaluates the consequence of a rupture of the SSC's pressure boundary. As stated in the Vogtle SER, "categorizing solely based on consequence which measures the safety significance of the pipe given that it ruptures is conservative compared to including the rupture frequency in the categorization and the categorization will not be affected by changes in frequency arising from changes to the treatment." Therefore, this methodology is appropriate to apply to ASME Class 1 SSCs, as the consequence evaluation and deterministic considerations are independent of the ASME classification when determining the SSC's safety significance and will maintain this acceptable level of conservatism. The passive categorization process is intended to apply the same risk-informed process accepted in the ANO2-R&R-004 for the passive categorization of Class 2 and 3 components, to Class 1 pressure retaining SSCs in the scope of the system being categorized.

Deleted: The requirements of 10 CFR 50.69 are consistent with the ANO-2 RI-RRA License Amendment as the rule does not remove the repair and replacement provisions of the ASME Code required by § 50.55a(g) for ASME Class 1 SSCs, even if they are categorized as RISC-3, since those SSCs constitute principal fission product barriers as part of the reactor coolant system or containment. This is further clarified in the rule's Statement of Considerations. However, since the scope of 10 CFR 50.69 addresses additional requirements, this methodology will be applied to determine the safety significance of ASME Class 1 SSCs, some of which may be evaluated to be RISC-3.

The ANO RI-RRA passive methodology implements the same risk-informed inservice inspection (RI-ISI) consequence evaluation process contained in EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Procedure" supplemented with additional qualitative considerations. The NRC SER of this EPRI topical report was issued by letter on October 28, 1999. Section 3.2.1 of the SER describes the scope of the RI-ISI methodology as:

The full-scope option includes ASME Code Class 1, 2, and 3 piping, piping whose failure could prevent safety-related structures, systems, or components (SSCs) from fulfilling their safety functions, and non-safety-related piping that is relied upon to mitigate accidents for whose failure could cause a reactor scram or actuation of a safety-related system.

While many pressure boundary components (passive components) are not "modeled" in a PRA, the consequence evaluation process of TR-112657, Rev B-A provides an explicit and robust process for determining the importance of pressure boundary components for both moderate and high energy systems. Consistent with the ASME/ANS PRA Standard, this supplementary analysis is used to augment the base PRA information. Further, as discussed above, the methodology uses the consequence portion of EPRI RI-ISI process enhanced with "additional considerations" which provide an additional layer of confidence for categorizing Class 1 SSCs as well as Class 2, 3 and non-class SSCs.

The same process as it pertains to inservice inspection has been approved for use on the full scope and code class designations of pressure retaining piping and welds in nuclear power plants. It has been determined to be sufficiently robust to assess the consequence risk of Class 1 piping and welds in the context of ISI even without the additional qualitative steps. The ANO RI-RRA has also determined to be sufficiently robust to assess the consequence of all Class 2 and Class 3 SSCs (with the additional qualitative steps) in the context of repair/replacement. Therefore, the ANO RI-RRA methodology should be sufficiently robust to assess the consequence of the full spectrum of pressure retaining components as well as active components with a pressure retaining function regardless of ASME classification.

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. The PRA models credited in this request are the same PRA models credited in the [TSTF-505-A application dated July 31, 2015 ADAMS Accession Number ML15218AXXX (Reference X)] with routine maintenance updates applied.

3.2.1 Internal Events and Internal Flooding

The [PLANT NAME] categorization process for the internal events and flooding hazard will use the plant-specific PRA model. The [LICENSEE] risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the [PLANT] units. Attachment 2 of this enclosure identifies the applicable internal events and internal flooding PRA models.

Note: Peer review information should only be discussed in Section 3.3

3.2.2 Fire Hazards

Option 1

Note: If the licensee uses a different name to identify the "Fire Safe Shutdown Program," replace term with the site-specific name.

The [PLANT NAME] categorization process will use the Fire Safe Shutdown Equipment List (SSEL) for evaluation of safety significance related to fire hazards. The Fire Safe Shutdown paths identify the safety functions and associated sets of equipment credited to achieve and maintain safe shutdown under postulated fire conditions as defined by 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979" (Reference X) and NRC Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power" (Reference Y). The Fire SSEL identifies the credited equipment on these Fire Safe Shutdown Paths. This approach also considers regulatory exemptions related to the Fire Safe Shutdown program and fire-induced Multiple Spurious Operations (MSOs) to identify any additional equipment.

The use of the Fire SSEL is a screening approach. There are no importance measures used in determining safety significance related to the fire hazard. Instead, using the Fire SSEL would identify all credited equipment as HSS regardless of their fire damage susceptibility or frequency of challenge. This approach ensures the SSCs that are credited to establish and maintain safe shutdown capability are retained as safety-significant. If a component is credited on the Fire SSEL, it is considered HSS. As stated in NEI 00-04, an SSC identified as HSS by a non-PRA method to address fire "may not be re-categorized by the IDP."

Furthermore, regulatory exemptions related to the Fire Safe Shutdown program and previously identified fire-induced MSOs were reviewed and it was concluded that no equipment in addition to the components on the Fire SSEL are relied upon to establish and maintain safe shutdown. Therefore, no additional components will be identified as HSS with regard to the fire hazard. The results of this review have been documented by the site and are available for NRC audit. Figure 3-1 illustrates how the Fire SSEL is reviewed to determine if the component being evaluated is HSS. *OR* Furthermore, regulatory exemptions related to the Fire Safe Shutdown program and previously identified fire-induced MSOs were reviewed and additional equipment that is relied upon to establish and maintain safe shutdown will be retained as HSS. The results of this review have been documented by the site and are available for NRC audit. Figure 3-1 illustrates how the Fire SSEL and the additional identified equipment are reviewed to determine if the component being evaluated is HSS.

This approach is an alternate process from the NEI 00-04 endorsed approaches. Similar to the NEI 00-04 FIVE approach, this approach uses the SSEL as a screening tool. However, the development of the Fire SSEL is not based on a successive screening methodology and is the starting point for the FIVE methodology. Therefore, industry assessments have shown that this Fire SSEL approach leads to many additional SSCs being identified as HSS making it more conservative in determining safety significance than the NEI 00-04 FIVE approach or a Fire PRA.

The [LICENSEE] Fire Safe Shutdown program is an active regulatory program that is routinely inspected by NRC. It was confirmed that this program ensures that the Fire SSEL and the identification of additional equipment relied upon to establish and maintain safe shutdown reflects the current as-built, as-operated plant and that changes to the plant will be evaluated to determine their impact to the equipment list and the categorization process.

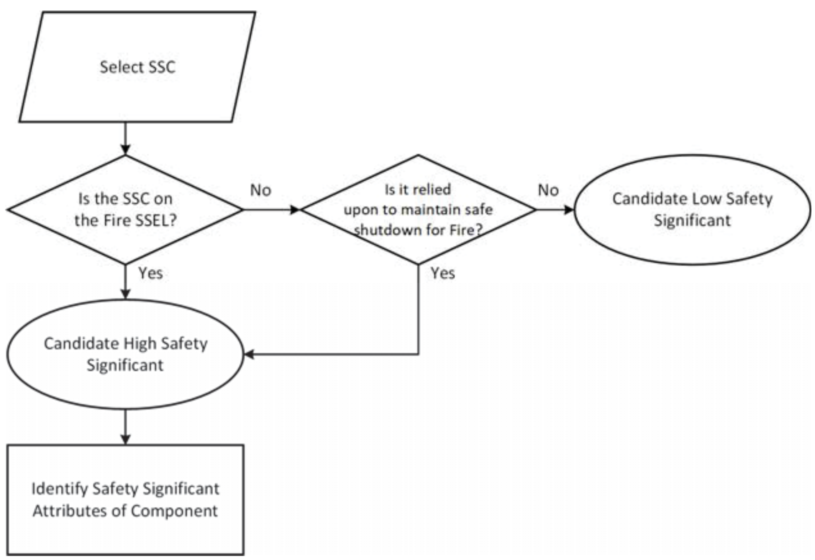

Figure 3-1: Safety Significance Process for Systems and Components Addressed in Fire Safe Shutdown Program

Option 2

The [PLANT NAME] 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 [LICENSEE] risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the [PLANT] units. Attachment X at the end of this enclosure identifies the applicable Fire PRA model.

Note: Peer review information should only be discussed in Section 3.3

3.2.3 Seismic Hazards

Option 1

The [PLANT NAME] categorization process will use the seismic margins analysis (SMA) performed for the Individual Plant Evaluation-External Events (IPEEE) in response to GL 88-20 (Reference 6) for evaluation of safety significance related to seismic hazards. No plant specific approaches were utilized in development of the SMA. The NEI 00-04 approved use of the SMA [SSEL] as a screening process results in the identification of all system functions and associated SSCs that are involved in the seismic margin success path as HSS. Since the analysis is being used as a screening tool, importance measures are not used to determine safety significance. The NEI 00-04 approach using the [SSEL] identifies identify credited equipment as HSS regardless of their capacity, frequency of challenge, or level of functional diversity.

An evaluation was performed of the as-built, as-operated plant against the SMA [SSEL]. The evaluation compared of the as-built, as-operated plant to the plant configuration originally assessed by the SMA. Differences were reviewed to identify any potential impacts to the equipment credited on the [SSEL]. Appropriate changes to the credited equipment were identified and documented. This documentation is available for audit. The [LICENSEE] risk management program ensures that future changes to the plant will be evaluated to determine their impact on the SMA and risk categorization process.

Option 2

The [PLANT NAME] categorization process for seismic hazards will use a peer reviewed plant-specific seismic PRA model. The [LICENSEE] risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the [PLANT] units. Industry standard methods were utilized in the development of the seismic hazards for the SPRA. Updates to seismic hazard curves will be reflected in the PRA used for the categorization in accordance with the PRA model maintenance process.

If Addendum B is credited use the following paragraph

Because ASME/ANS PRA Standard Addendum B is credited for review or disposition of peer review findings, the following issues in NRC letter ML111720067 (Reference 13) are dispositioned as follows…]

The seismic analysis is based on the latest available hazard curves. Plant specific approaches to seismic hazards are described as follows:

[Summarize any plant-specific approaches used]

Attachment X at the end of this enclosure identifies the applicable Seismic PRA model.

Note: Peer review information should only be discussed in Section 3.3

3.2.4 Other External Hazards

Note: This first option is for other external hazards (i.e., external hazards that are not fire or seismic) where the licensee plans to use a PRA model.

The [PLANT NAME] categorization process for the following hazard[s] will use peer-reviewed PRA models (as applicable) in accordance with the ASME PRA Standard RA-Sa-2009:

[List Hazards]

The [LICENSEE] risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the [PLANT] units. Attachment[s] X at the end of this enclosure identifies the applicable other external hazard PRA model[s].

Note: This next option of using IPEEE screening results are for hazards that were NOT completely "screened" and instead used a screening process to identify SSCs that were necessary to protect against the hazard (e.g., tornado missile barrier). See NEI 00-04 for more information

The [PLANT NAME] categorization process will use screening results from the IPEEE in response to Generic Letter (GL) 88-20 (Reference 6) for evaluation of safety significance related to the following external hazards:

[List Hazards]

Figure 5-6 in Section 5.4 of NEI 00-04 illustrates the process that will be used to determine safety significance related to the above hazards.

All other external hazards were screened from applicability to [PLANT/UNIT] per a plant-specific evaluation in accordance with GL 88-20 (Reference 6) and updated to use the criteria in ASME PRA Standard RA-Sa-2009. Attachment 4 provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for external hazards.

Note: This next option should be used for all other external hazards that were screened due to plant design and credited specific components in that screening.

As part of the categorization assessment of other external hazard risk, an evaluation is performed to determine if there are components being categorized participate in screened scenarios and whose failure would result in an unscreened scenario. Consistent with the flow chart in Figure 5-6 in Section 5.4 of NEI 00-04, these components would be considered HSS.

[List Hazards]

Note: This next option should be used for all other external hazards that were screened by frequency. This language should be generically applicable.

All remaining hazards were screened from applicability and considered insignificant for every SSC and, therefore, will not be considered during the categorization process.

If any changes are implemented: Changes to the IPEEE analysis or incorporation of plant modifications in the updated IPEEE analysis are summarized as follows:

[Summarize any updates]

3.2.5 Low Power & Shutdown

Consistent with NEI 00-04, the [PLANT NAME] 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 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 NEI 00-04 will be considered preliminary HSS.

3.2.6 PRA Maintenance and Updates

The [LICENSEE] risk management process ensures that the applicable PRA model(s) used in this application continues to reflect the as-built and as-operated plant for each of the [PLANT] units. 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.

In addition, [LICENSEE] will 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.

In the overall risk sensitivity studies, [LICENSEE] will utilize a factor of 3 to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle in Reference 6. Consistent with the NEI 00-04 guidance, [LICENSEE] 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 9). 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 [PLANT] 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 [PLANT] 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 [PLANT] PRA model specific assumptions or sources of uncertainty except for the following:

- Perform a sensitivity increasing all the seismic PRA human events failures (HEFs) derived from the internal events PRA model by a factor of 3 to address the uncertainty associated with main control room actions that might take longer in a seismic event versus an internal initiating event.

3.3 PRA REVIEW PROCESS RESULTS (10 CFR 50.69(b)(2)(iii))

Note: Only describe the most recent full scope peer reviews and related focused scope peer reviews. Historical information of peer reviews should not be included.

The PRA model[s] described in Section 3.2 has 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 7) consistent with NRC RIS 2007-06.

The internal events PRA model was subject to a self-assessment and a full-scope peer review conducted in [Month Year]. *Describe any related focused scope peer reviews since the most recent full scope peer review.*

If Peer Review was not completed against RG 1.200 Rev 2.

Since the peer review of the Internal Events PRA model was performed prior to the publication of RG 1.200 Rev 2, a self-assessment was conducted to assess the differences between RG 1.200 Rev 2 and RG 1.200 Rev X. That assessment confirmed that the PRA model meets the requirements of RG 1.200 Rev 2. Results from that assessment are documented in Attachment 7.

OR

The Internal Events PRA model was peer reviewed in [YEAR] by the [PWR or BWR] Owners Group (PWROG or BWROG) prior to the issuance of Regulatory Guide 1.200. As a result, a self-assessment was conducted by [LICENSEE] of the Internal Events PRA model in accordance with Appendix B of RG 1.200 Revision 2 (Reference X) to address the PRA technical adequacy requirements not considered in the [YEAR] peer review. The Internal Events PRA technical adequacy (including the [YEAR] peer review and self-assessment results) has previously been reviewed by the NRC in previous requests to [describe application] (Reference XX). Any PRA upgrades as defined by the ASME PRA Standard RA-Sa-2009 (Reference 10) since this review have been reviewed by a focused scope peer review. *OR* No PRA upgrades as defined by the ASME PRA Standard RA-Sa-2009 (Reference 10) have occurred to the Internal Events PRA model since conduct of the [PWROG or BWROG] peer review in [YEAR].

The Fire PRA model was subject to a self-assessment and a full-scope peer review conducted in [Month Year]. *Describe any related focused scope peer reviews since the most recent full scope peer review.*

The [Other Hazard (e.g., seismic)] PRA model was subject to a self-assessment and a full-scope peer review conducted in [Month Year]. The [Other Hazard] PRA Peer Review was conducted consistent with NEI 12-13, Revision 0, "External Hazards PRA Peer Review Process Guidelines," (Reference 13) and addressed all NRC comments documented in the associated NRC Letter ML12321A280 (Reference 14). *Describe any* *related focused scope peer reviews since the most recent full scope peer review. A similar paragraph should be repeated for each external hazard modeled.*

A finding closure review was conducted on the identified PRA models on [Date(s)]. 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" (F&Os) (Reference 11) as accepted by NRC in the letter dated May 3, 2017 (ML17079A427) (Reference 12). The results of this review have been documented and are available for NRC audit.

Attachment X provides a summary of the remaining findings and open items, including:

- Open items and disposition from the [PLANT NAME] RG 1.200 self-assessment.
- Open findings and disposition of the [PLANT NAME] peer reviews. This also includes those open peer review findings that are requested the NRC review for closure as part of this LAR.
- Identification of and basis for any sensitivity analysis needed to address open findings.

The attachment[s] 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 10 CFR 50.69(c)(1)(i).

3.4 RISK EVALUATIONS (10 CFR 50.69(b)(2)(iv))

The [PLANT NAME] 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 core damage frequency (CDF) and large early release frequency (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.

4 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 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
- Regulatory Guide 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.
- Regulatory Guide 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

[LICENSEE] proposes to modify the licensing basis to allow for the voluntary implementation of the provisions of 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." 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.

[LICENSEE] 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 their 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, [LICENSEE] 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 CONCLUSIONS

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 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 REFERENCES

Note that references may need to be renumbered based on optional and site specific text used

1. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.
2. NRC Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991.
4. EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, Final Report," Revision B-A, January 2000.
5. ANO SER Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-R&R-004, Revision 1, Request to Use Risk-Informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems (TAC NO. MD5250) (ML090930246), April 22, 2009.
6. Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Supplement 4," USNRC, June 1991.
7. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, , March 2009.
8. NUREG-1855, Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, March 2009
9. EPRI TR-1016737, Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, December 2008
10. ASME/ANS RA-Sa-2009, Standard for Level I/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, dated February 2009
11. NEI Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," February 21, 2017, Accession Number ML17086A431.
12. NRC Letter to Mr. Greg Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017, Accession Number ML17079A427.

13. NRC Letter to Mr. Oliver Martinez, "U.S. Nuclear Regulatory Commission (NRC) Comments on 'Addenda to a Current ABS: ASME RA-SB - 20XX, Standard for Level 1/Larege Early Release Frequency Probabilisitic Risk Assessment For Nuclear Power Plant Applications"

14. NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," Revision 0, Nuclear Energy Institute, August 2012.

15. NRC Letter to Mr. Biff Bradley (NEI), "U.S. Nuclear Regulatory Commission Comments on Nuclear Energy Institute 12-13, 'External Hazards PRA Peer Review Process Guidelines,' Dated August 2012," November 16, 2012, Accession Number ML12321A280.

16. Add any references to NRC review of plant specific FIVE, SMA, or IPEEE screening for Section 3.1.1

17. Add any optional references on previously approved applications with NRC review of PRA models for Section 3.1.1.

Attachment 1: List of Categorization Prerequisites

Licensing Note: The need for a License Condition and/or regulatory commitments should be reviewed by the Licensing Department. In general, plants should identify the need for plant modifications to meet the CDF and LERF risk values in RG 1.174 as a License Condition and the need for site procedures to address the items listed below as an NRC Commitment. However, the final decision is left to each individual plant.

The PRA model to be used for categorization credits the following modifications to achieve an overall Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) consistent with NRC Regulatory Guide 1.174 risk limits. Use of the categorization process on a plant system will only occur after the modifications are completed.

1. [Describe modification]
2. Install fuses in non-class DC motor circuits to prevent secondary fires due to multiple fire induced faults. This modification is complete in Unit 1, and is scheduled to be implemented in Unit 3 in the fall 2016 refueling outage and Unit 2 in the spring 2017 refueling outage.

[LICENSEE] 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 Signficant (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 their 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

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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.

Attachment 2: Description of PRA Models Used in Categorization

Units	Model	Baseline CDF	Baseline LERF	Comments
1	[reference, review, date]	Core Damage Frequency	Large Early Release Frequency	[applicable prior approvals] [one model applicable to all units]
2	BB06F dated October 10, 2014 Peer Reviewed Against RG 1.200 R2 on June 9, 2015	1.2E-05	1.7E-06	NRC reviewed model for risk-informed completion times (MLXXXXXXXX)
3	BB07F dated October 10, 2014 Peer Reviewed Against RG 1.200 R2 on June 9, 2015	1.2E-05	1.7E-06	NRC reviewed model for risk-informed completion times (MLXXXXXXXX)

Attachment 3: Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69 [And for Other Applications]
Provide identifier from Peer Review Report *(If multiple models are being used, identify which model the finding applies to)*	ASME/ANS Identifier	Capability Category identified in peer review report	Write up of finding from peer review report	Identify whether the finding was resolved. Request NRC's review for closure if needed. Provide a description of the disposition of the finding. State that this disposition or closure is appropriate for all identified applications or specify which applications.
HR-G6-01	HR-G6	CC-I/II/III Not Met	Check of consistency and review for reasonableness is missing in the Revision 4 updated HRA draft and the prior revision document information related to these items is not appropriate to use in light of the updates performed and changes to the results. Section 8 includes a table of human failure events (HFEs) and human error probabilities (HEPs) but does not include HEP reasonableness check, as is documented in Section 8.3 of the November 2005 HRA update for Revision 3.	This F&O was resolved. It is requested that NRC review the resolution of this finding for closure against the base model. All HRAs were reviewed and were either determined to be reasonable or have been revised. This review is documented in Section 8.2.2 of the internal events PRA calculation (Reference X).

Finding Number	Supporting Requirement(s)	Capability Category (CC)	Description	Disposition for 50.69 [And for Other Applications]
CS-C2-02	CS-C2	CC I/II/III MET	A summary of the fire zone nomenclature (e.g. used in cable routing) and table associating fire zones with physical analysis units and referring to appropriate plant drawings and site maps would simplify review. Information is available but scattered, complicating review. Condense the information from the FSAR Chapter 9A (Fire Hazards Analysis) into a table. Add nomenclature description and appropriate plant drawings and site maps.	This F&O refers to a documentation enhancement. The resolution of this F&O has no impact on any technical element of the analysis.
QU-F2-01	QU-F2	CC-I/II/III Not Met	Asymmetry analysis was not performed in the quantification analysis. Insights from alternate alignments may not be adequately categorized or identified.	Alternate alignment runs were performed to identify if uncertainty or risk insights would be affected as a result of an assumed alignment. This included a review of the FV and RAW importance measures that will be used for the categorization of SSCs. It was determined that alternate alignments would no impact the categorization of any SSCs. Attachment 1 provides more details of the alternate alignment and sensitivity cases that were performed.

Attachment 4: External Hazards Screening

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Aircraft Impact	Y	PS2 PS4	Airport hazard meets 1975 SRP requirements. Additionally, airways hazard bounding analysis per NUREG-1855 is < 1E-6/y.
Avalanche	Y	C3	Not applicable to the site because of climate and topography.
Biological Event	Y	C3, C5	Sudden influxes not applicable to the plant design (closed loop systems for ECWS and CWS). Slowly developing growth can be detected and mitigated by surveillance.
Coastal Erosion	Y	C3	Not applicable to the site because of location.
Drought	Y	C5	Plant design eliminates drought as a concern. In addition, this event is slowly developing.
External Flooding	Y	PS2	Plant design meets 1975 SRP requirements.
Extreme Wind or Tornado	Y	PS2 PS4	The plant design basis tornado has a frequency < 1E-7/y. The spray pond nozzles (not protected against missiles) have a bounding median risk < 1E-7/y.
Fog	Y	C1	Limited occurrence because of arid climate and negligible impact on the plant.
Forest or Range Fire	Y	C3	Not applicable to the site because of limited vegetation.
Frost	Y	C1	Limited occurrence because of arid climate.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Hail	Y	C1 C4	Limited occurrence and bounded by other events for which the plant is designed. Flooding impacts covered under Intense Precipitation.
High Summer Temperature	Y	C1	Plant is designed for this hazard. Associated plant trips have not occurred and are not expected.
High Tide, Lake Level, or River Stage	Y	C3	Not applicable to the site because of location.
Hurricane	Y	C4	Covered under Extreme Wind or Tornado and Intense Precipitation.
Ice Cover	Y	C3 C1	Ice blockage causing flooding is not applicable to the site because of location (no nearby rivers and climate conditions). Plant is designed for freezing temperatures, which are infrequent and short in duration.
Industrial or Military Facility Accident	Y	PS2	Explosive hazard impacts and control room habitability impacts meet the 1975 SRP requirements (RGs 1.91 and 1.78).
Internal Flooding	N	None	PRAs addressing internal flooding have indicated this hazard typically results in CDFs ≥ 1E-6/y. Also, the ASME/ANS PRA Standard requires a detailed PRA for this hazard which is addressed in the [PLANT/UNIT] Internal Flooding PRA.
Internal Fire	N	None	PRAs addressing internal fire have indicated this hazard typically results in CDFs ≥ 1E-6/y. Also, the ASME/ANS PRA Standard requires a detailed PRA for this hazard which is addressed in the [PLANT/UNIT] Internal Fire PRA.
Landslide	Y	C3	Not applicable to the site because of topography.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Lightning	Y	C1	Lightning strikes causing loss of offsite power or turbine trip are contributors to the initiating event frequencies for these events. However, other causes are also included. The impacts are no greater than already modeled in the internal events PRA.
Low Lake Level or River Stage	Y	C3	Not applicable to the site because of location.
Low Winter Temperature	Y	C1 C5	Extended freezing temperatures are rare, the plant is designed for such events, and their impacts are slow to develop.
Meteorite or Satellite Impact	Y	PS4	The frequency of meteorites greater than 100 lb striking the plant is around 1E-8/y and corresponding satellite impacts is around 2E-9/y.
Pipeline Accident	Y	C3	Pipelines are not close enough to significantly impact plant structures.
Release of Chemicals in Onsite Storage	Y	PS2	Plant storage of chemicals meets 1975 SRP requirements.
River Diversion	Y	C3	Not applicable to the site because of location.
Sand or Dust Storm	Y	C1 C5	The plant is designed for such events. Also, a procedure instructs operators to replace filters before they become inoperable.
Seiche	Y	C3 C1	Not applicable to the site because of location. Onsite reservoirs and spray ponds designed for seiches.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Seismic Activity	N	None	PRAs addressing seismic activity have indicated this hazard typically results in CDFs ≥ 1E-6/y. Also, the ASME/ANS PRA Standard requires a detailed PRA or Seismic Margins Assessment (SMA) for this hazard which is addressed in the [PLANT/UNIT] Seismic PRA or SPCL.
Snow	Y	C1 C4	The event damage potential is less than other events for which the plant is designed. Potential flooding impacts covered under external flooding.
Soil Shrink-Swell Consolidation	Y	C1 C5	The potential for this hazard is low at the site, the plant design considers this hazard, and the hazard is slowly developing and can be mitigated.
Storm Surge	Y	C3	Not applicable to the site because of location.
Toxic Gas	Y	C4	Toxic gas covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident.
Transportation Accident	Y	PS2 PS4 C3 C4	Potential accidents meet the 1975 SRP requirements. Bounding analyses used for offsite rail shipment of chlorine gas and onsite truck shipment of ammonium hydroxide. Marine accident not applicable to the site because of location. Aviation and pipeline accidents covered under those specific categories.
Tsunami	Y	C3	Not applicable to the site because of location.
Turbine-Generated Missiles	Y	PS2	Potential accidents meet the 1975 SRP requirements.
Volcanic Activity	Y	C3	Not applicable to the site because of

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
			location.
Waves	Y	C3 C4	Waves associated with adjacent large bodies of water are not applicable to the site. Waves associated with external flooding are covered under that hazard.
Note a – See Attachment 5 for descriptions of the screening criteria.			

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	
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	

Event Analysis	Criterion	Source	Comments
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	

Attachment 6: Disposition of Key Assumptions/Sources of Uncertainty

The following table provides the structure of this attachment which should be populated with items of significance from the plant-specific evaluation.

Assumption/ Uncertainty	Discussion	Disposition

Attachment 7: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III

This attachment is optional and the following table can be adjusted if necessary

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
IE-C10:CC-I/IIIIII: **...** **An example of an acceptable generic data sources is NUREG/CR-5750 Note 1.**	IE-C12: CC-I/IIIIII: ... An example of an acceptable generic data sources is NUREG/CR-6928 Note 1.	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR cites a more recent example of an acceptable generic data source.	[identify if NUREG/CR-5750 data is used. If so, justify it's use or provide sensitivity study of impact of changing to more recent data source]
SY-B15: CC-I/II/III: **...** **(h) harsh environments induced by containment venting, or failure that may occur prior to the onset of core damage.**	SY-B14: CC-I/II/III: ... (h) harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage	The sentences were clarifications provided in RG 1.200 Revision 1 and Revision 2, respectively. The updated SR explicitly requires consideration of containment venting ducts and failure of the containment boundary prior to core damage.	[Confirm that additional failure modes were considered or perform sensitivity study of impact from additional failure modes]

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
DA-C1: CC-I/II/III: **...** **Examples of parameter estimates and associated sources include:** **(a) component failure rates and probabilities: NUREG/CR-4639 Note (1), NUREG/CR-4550 Note (2), NUREG-1715 Note 7**	DA-C1: CC-I/II/III: ... Examples of parameter estimates and associated sources include: (a) component failure rates and probabilities: NUREG/CR-4639 2-7, NUREG/CR-4550 2-3, NUREG-1715 2-21, NUREG/CR-6928 2-20	Reference NUREG-1715 was added by RG 1.200 Revision 1; References NUREG-1715 and NUREG/CR-6928 were included in the 2009 version of the PRA Standard. The updated SR cites more recent examples of acceptable generic data sources.	Though additional examples of generic data were identified, they don't super cede the previous data source and will not impact the technical adequacy of the PRA.
QU-A2a: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF ...	QU-A2: CC-I/II/III: PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF (and LERF) ...	The LERF requirement was added by RG 1.200 Revision 2. The updated SR explicitly requires consideration of LERF for sequence quantification.	Sequence quantification for LERF may identify enhancements to be made in the LERF model for a more realist ice estimate of LERF. However, as the sequence quantification is not used in the NEI 00-04 Risk Ranking methodology along with Defense-in-Depth considerations, not having LERF quantified at the sequence level will not impact the categorization results.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
QU-A2b: **CC-I:** **ESTIMATE the point estimate CDF from internal events.** **CC-II:** **ESTIMATE the mean CDF from internal events, accounting for the "state-of-knowledge" correlation between event probabilities Note (1).** **CC-III:** **CALCULATE the mean CDF from internal events by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.**	QU-A3: CC-I: ESTIMATE the point estimate CDF (and LERF). CC-II: ESTIMATE the mean CDF (and LERF) accounting for the "state-of-knowledge" correlation between event probabilities Note (1). CC-III: CALCULATE the mean CDF (and LERF) by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.	The phrase, "from internal events", was deleted from the 2009 version of the PRA Standard. The LERF requirement was added by RG 1 .200 Revision 2. The SR explicitly requires consideration of LERF.	Per the note in 2007 SR LE-E4 and LE-F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
QU-B6:CC I/II/III: ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the successes may not be transferred between event trees.	QU-B6:CC I/II/III: ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF or LERF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the successes may not be transferred between event trees.	The LERF requirement was added by RG 1.200 Revision 2.	The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE E-4 and LE F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF requirements.
QU-E3: **<u>CC-I:</u>** **ESTIMATE the uncertainty interval of CDF results. Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).**	QU-E3: <u>CC-I:</u> ESTIMATE the uncertainty interval of CDF (and LERF) results. **Provide a basis for the estimate consistent with the characterization parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15).** <u>CC-II:</u>	The LERF requirement was added by RG 1.200 Revision 2.	The SR explicitly requires consideration of LERF. However, per the note in 2007 SR LE E-4 and LE F3, LERF was addressed in applicable requirements of Table 4.5.8, which includes all QU SRs. Thus, the peer review using the 2007 version of the PRA Standard was addressed these LERF

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
CC-II: **ESTIMATE the uncertainty interval of the CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** **CC-III:** **Propagate parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)....(no change)**	ESTIMATE the uncertainty interval of the CDF (and LERF) results. **ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15), taking into account the state-of-knowledge correlation.** CC-III: **Propagate parameter uncertainties (DA-D3, HR-D6, HR-G8, IE-C15)....(**no change)		requirements.
QU-E4: **CC-I:** **PROVIDE an assessment of the impact of the model uncertainties and assumptions on the results of the PRA.** **CC-II:** **EVALUATE the sensitivity of the results to model uncertainties and key assumptions using**	QU-E4: CC-I/II/III: For each source of model uncertainty and related assumption identified in QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).	Separate requirements for CC-I, II and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard. The reference to Note 1 was deleted by RG 1.200 Revision 2.	The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
sensitivity analyses Note (1). **<u>CC-III:</u>** **EVALUATE the sensitivity of the results to uncertain model boundary conditions and other assumptions using sensitivity analyses except where such sources of uncertainty have been adequately treated in the quantitative uncertainty analysis Note (1).**			
LE-F2: **<u>CC-I</u>:** **PROVIDE a qualitative assessment of the key sources of uncertainty.** **Examples:** **(a) Identify bounding assumptions.** **(b) Identify conservative treatment of phenomena.** **<u>CC-II</u>:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty**	LE-F3: <u>CC-I/IIIIII:</u> IDENTIFY and CHARACTERIZE the LERF sources of model uncertainty and related assumptions, in a manner consistent with the applicable requirements of Tables 2-2.7-2(d) and 2-2.7-2(e).	Separate requirements for CC-I, II, and III were collapsed into a single requirement for CC-I/II/III in the 2009 version of the PRA Standard.	The updated SR assigns the same requirement to all three CCs. Meeting CC-II: in the 2007 version of the PRA Standard assures that the new SR is met.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
and includes sensitivity studies for the significant contributors to LERF. **CC-III:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies.**			
IF-F2: **CC-I/II/III:** **DOCUMENT the process used to identify … flood areas… , For example, this documentation typically includes** **…** **(b) flood areas used in the analysis and the reason for eliminating areas from further analysis**	IFPP-B2: CC-I/II/III: DOCUMENT the process used to identify flood areas. For example, this documentation typically includes (a) flood areas used in the analysis and the reason for eliminating areas from further analysis (b) any walkdowns performed in support of the plant partitioning	The requirement to document walkdowns performed in support of plant partitioning was added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources.	Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was not reviewed as part of the peer review conducted using that version of the PRA Standard. A self-assessment against the 2009 version of the standard was performed and [it was determined that the documentation of flood walkdowns meets the requirement of the 2009 standard] OR [the flood walk down documentation was updated to meet the

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
			requirements of the standard and the new walk down information was evaluated to determine that it had no impact of the Flood PRA model] OR [the flood walk down documentation was updated to meet the requirements of the standard and the Flood PRA model was updated to account for new walkdown information]
IF-B1: <u>CC-I/II/III:</u> For each flood area, IDENTIFY the potential sources of flooding Note (1). INCLUDE: **(a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems)**	IFSO-A1 : <u>CC-I/II/III:</u> For each flood area, IDENTIFY the potential sources of flooding Note (1). INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems, and reactor coolant system) ...	The requirement to include the fire protection system in Item (a) as a potential flooding source was added by RG 1.200 Revision 1. The requirement to include the reactor coolant system in Item (a) as a potential flooding source was added to the 2009 version of the PRA Standard.	[This requirement was addressed in the peer review, which used the 2007 version of the PRA Standard amended by RG 1.200 Revision 1]. OR [The flood model was reviewed and it was confirmed that the fire protection and RCS systems are included in the flood model] OR [The fire protection and RCS were added as sources of

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
			flooding to the flood model]
IF-F2 **<u>CC-I/II/III:</u>** **DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes:** **flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined** **…** **(f) screening criteria used in the analysis** **….** **(j) calculations or other analyses used to support or**	IFSO-F2 <u>CC-I/II/III:</u> DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: Flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined Screening analysis used in the analysis calculations or other analyses used to support or refine the flooding evaluation any walkdowns performed in support of identification or	The requirement to document walkdowns performed in support of the identification or screening of flood sources as added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources.	The internal flood PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
refine the flooding evaluation	screening of flood sources		
IF-F2 **<u>CC-I/II/III:</u>** **DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes:** **...** **(c) propagation pathways...** **...** **(d) accident mitigating features and barriers credited...** **...** **(e) assumptions or calculations used in the determination of ...flood-induced effects on equipment operability**	IF-F2 <u>CC-I/II/III:</u> DOCUMENT the process used to identify applicable flood sources. For example, this documentation typically includes: ... (a) propagation pathways... ... (b) accident mitigating features and barriers credited... ... (c) assumptions or calculations used in the determination of ...flood-induced effects on equipment operability ... (d) screening criteria used in the	The requirement to document walkdowns performed in support of the identification or screening of flood sources as added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the process to identify flood sources. Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was *not reviewed as part of*	The internal flood PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs, mitigation and other flood related features in the flood areas that are considered in flood sequence definition.

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
... **(f) screening criteria used in the analysis** **...** **(g) flood scenarios considered, screened, and retained** **...** **(h) description of how the internal events analysis models were modified...** **....** **(j) calculations or other analyses used to support or refine the flooding evaluation** **...**	analysis ... (e) flood scenarios considered, screened, and retained ... (f) description of how the internal events analysis models were modified... .... (g) calculations or other analyses used to support or refine the flooding evaluation ... (h) any walkdowns performed in support of identification or screening of flood scenarios	*the peer review conducted using that version of the PRA Standard*.	
IF-F2 **<u>CC-I/II/III:</u>** **DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes:**	IF-F2 <u>CC-I/II/III:</u> DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes: ...	The requirement to document walkdowns performed in support of the identification or screening of flood sources as added to the 2009 version of the PRA Standard. The updated SR cites examples of acceptable documentation of the	Since documentation of walkdowns was not in the 2007 version of the PRA Standard, it was *not* reviewed as part of the peer review conducted using that version of the PRA Standard The internal flood PRA documents the walkdowns performed to validate information related to flood areas, flood sources, SSCs,

SR in 2007 PRA Standard as Amended by RG 1.200, Revision 1	SR in 2009 PRA Standard as Amended by RG 1.200, Revision 2	Description of Change	Resolution
... **(j) calculations or other analyses used to support or refine the flooding evaluation** **...** **(f) screening criteria used in the analysis** **...** **(i) flooding scenarios considered screened, and retained** **...** **(k) results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D**	(j) calculations or other analyses used to support or refine the flooding evaluation ... (f) screening criteria used in the analysis ... (i) flooding scenarios considered screened, and retained ... (k) results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D ... (e) any walkdowns performed in support of internal flood accident sequence quantification	process to identify flood related features considered in flood sequence quantification.	mitigation and other flood related features in the flood areas that are considered in flood sequence definition.