NRR-PMDAPEm Resource

From:	Miller, Ed
Sent:	Thursday, January 26, 2017 3:37 PM
To:	Miller, Ed
Subject:	Jan 25, 2017 Public Meeting With NEI on 10 CFR 50.69 - LAR Template
Attachments:	NEI 50_69 License Amendment Request Template.pdf

The NEI Template LAR from the meeting are attached

Hearing Identifier: NRR_PMDA
Email Number: 3297

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Subject: Jan 25, 2017 Public Meeting With NEI on 10 CFR 50.69 - LAR Template
Sent Date: 1/26/2017 3:37:11 PM
Received Date: 1/26/2017 3:37:13 PM
From: Miller, Ed

Created By: Ed.Miller@nrc.gov

Recipients:
"Miller, Ed" <Ed.Miller@nrc.gov>
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MESSAGE	52	1/26/2017 3:37:13 PM
NEI 50_69 License Amendment Request Template.pdf	599819	

Options
Priority: Standard
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Reply Requested: No
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APPENDIX A: 10 CFR 50.69 LICENSE AMENDMENT REQUEST TEMPLATE

The purpose of this appendix 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.

Boiler Plate 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.

10 CFR 50.90
10 CFR 50.69

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

SUBJECT: [PLANT NAME]
DOCKET NO.   50-[xxx]
              Application to adopt 10 CFR 50.69, "Risk-informed categorization
              and treatment of structures, system, and components (SSCs) for
              nuclear power plants"

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 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 (SSCs) for Nuclear Power Plants." 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.]

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]. [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 review 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 and allow the NRC staff to independently review and approve each LAR 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 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 and allow the NRC staff to independently review and approve each LAR 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 NRC commitments.

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].

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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LIST OF ATTACHMENTS

1   SUMMARY DESCRIPTION

The proposed amendment would modify 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 (SSCs) for Nuclear Power Plants." 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. Those 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 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).

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 proceeding 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 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 is a duplicate of the preparer's note in the cover letter].

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]. [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 review 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 and allow the NRC staff to independently review and approve each LAR 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 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 and allow the NRC staff to independently review and approve each LAR 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 the NEI 00-04, Revision 0, as endorsed by 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." Separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The following are the clarifications or deviations taken 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 worked on the modeling and updating of the plant-specific PRA for a minimum of three years.
- 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.
- 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 9.2.2 states that for those safety related functions that have been identified as candidate low safety significant, the IDP should perform a risk-informed assessment of the categorization by considering each of the seven listed criteria. [LICENSEE] intends to perform this assessment for all system functions at the beginning of the categorization process, after all system functions have been identified. Any system function that cannot be justified as LSS based on consideration of each the seven criteria will be categorized as HSS. 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.

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- [LICENSEE] will require that if any SSC is identified as high safety significant (HSS) from either the integrated 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.
- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS.

The risk analysis being implemented for each hazard is described:

- 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 induced vulnerability evaluation (FIVE) [accepted by NRC SER dated xx, ML # (Reference X)]. 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, etc.): 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.

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 are evaluated through a process that utilizes the guidance in EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, Final Report," Revision B-A, (Reference 4) as approved by the NRC in its Safety Evaluation. The passive component categorization process will implement the

EPRI TR-112657 method consistent with the Safety Evaluation Report by the Office of Nuclear Reactor Regulation "Request tor 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, Third and Fourth 10-Year In-service Inspection Intervals', dated April 22, 2009 (Reference 5). The following additional requirements will be applied to this process:

- Component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed.
- Additional deterministic considerations (e.g., defense in depth, safety margins) are applied as part of the overall categorization process.
- ASME Class 1 components are, by default, categorized as HSS with respect to passive risk.
- Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model.

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. All 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 [LICENSEE] 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 1 at the end of this enclosure identifies the applicable internal events and internal flooding PRA models.

3.2.2 Fire Hazards

Option 1

The [PLANT NAME] categorization process will use the Fire Induced Vulnerability Evaluation (FIVE) analysis performed for the Individual Plant Evaluation-External Events (IPEEE) in response to GL 88-20 (Reference 6) for evaluation of safety significance related to Internal Fire Hazards. An evaluation was performed of the as-built, as-operated plant against the fire scenarios identified in the FIVE analysis, which determined that there have been no changes in the mitigation function of equipment for any unscreened fire scenarios. In addition, screened scenarios were reviewed and

no credited functions or SSCs required to perform those functions have been affected. The [LICENSEE] risk management program ensures that future changes to the plant will be evaluated to determine their impact on the FIVE analysis and risk categorization process.

Option 2

The [PLANT NAME] categorization process will use the Fire Induced Vulnerability Evaluation (FIVE) analysis performed for the Individual Plant Evaluation-External Events (IPEEE) in response to GL 88-20 (Reference 6) for evaluation of safety significance related to internal fire hazards. An evaluation was performed of the as-built, as-operated plant against the fire scenarios identified in the FIVE analysis and changes to the mitigation features are identified in Attachment X. In addition, screened scenarios were reviewed and changes to credited functions or SSCs required to perform those functions are also identified in Attachment X. The [LICENSEE] risk management program ensures that future changes to the plant will be evaluated to determine their impact on the FIVE analysis and risk categorization process.

Option 3

The [PLANT NAME] categorization process for fire hazards will use a peer reviewed plant-specific fire 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 X at the end of this enclosure identifies the applicable Fire PRA model.

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. An evaluation was performed of the as-built, as-operated plant against the SPCL identified in the SMA which determined that there have been no changes to the success paths. 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 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.

An evaluation was performed of the as-built, as-operated plant against the SPCL identified in the SMA and changes to the success paths are identified in Attachment X. 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 3

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

3.2.4 Other External Hazards

Option 1 - for screened hazards

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

[List Hazards]

Option 2

The [PLANT NAME] categorization process for the following hazard[s] will use a peer reviewed plant-specific PRA model:

[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].

All other external hazards were screened from applicability to [PLANT/UNIT] per a plant-specific evaluation in accordance with GL 88-20 (Reference 5). Attachment 7 provides a summary of the other external hazards screening results. Attachment 8 provides a summary of the progressive screening approach for external hazards.

3.2.5 Low Power & Shutdown

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.

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, 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 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04.

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 PRAs 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.

Sources of model uncertainty and related assumptions have been identified for the [PLANT] PRA models using the guidance of NUREG-1855 (Reference 8) and EPRI TR-1016737 Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessment (Reference 9).

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. 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 which 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 configuration risk calculations were considered key for this application.

Key [PLANT] PRA model specific assumptions and sources of uncertainty for this application are identified and dispositioned in Attachment 9. 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))

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). Specifically, the model was subject to a self-assessment and a peer review conducted in [Month Year]. A summary disposition of open findings are provided in Section 3.3.2. Closed findings were closed using an NRC accepted closure process.

Attachment X provides a summary of:

- Open items and disposition from the [PLANT NAME] RG 1.200 self-assessment.
- Open findings and disposition of the [PLANT NAME] peer review. 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.

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

Since the peer review was performed prior to the publication of RG 1.200 Rev 2, the results of a self-assessment of the differences between RG 1.200 Rev 2 and RG 1.200 Rev X are documented in Attachment X.

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). 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 table[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, human errors, etc.). 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 at 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, US Nuclear Regulatory Commission, 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 (SSCs) for Nuclear Power Plants." 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 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

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, US Nuclear Regulatory Commission, 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 l/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. Add any references to NRC review of plant specific FIVE, SMA, or IPEEE screening for Section 3.1.1

12. 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

The PRA model to be used for categorization credits the following modifications to achieve an overall CDF and 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 HSS or LSS based on the seven questions 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 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. Components that are categorized· as preliminary LSS are evaluated for their role in providing defense-in-depth and safety margin and, if appropriate, upgraded to HSS.
- Review by the Integrated Decision-making Panel. The categorization results are presented to the lDP for review and approval. The lDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.
- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of RG 1.174.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.

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)

Information on additional PRA models (e.g. Fire PRA and Seismic PRA) should be added to this table

Attachment 3: Changes to Fire Induced Vulnerability Evaluation (FIVE)

Equipment/Function Credited	Description of Change	Impacts/Comments
[Describe equipment or function]	[Describe the change made to the equipment or function]	[Discuss the impacts and changes to the results of the analysis]
Diesel Driven Aux Feed Pump	Diesel driven pump replaced with steam driven pump	Steam driven pump is now credited mitigation equipment and is retained as safety significant

Attachment 4: Changes to Seismic SPCL

Equipment/Function Credited	Description of Change	Impacts/Comments
[Describe equipment or function]	[Describe the change made to the equipment or function]	[Discuss the impacts and changes to the results of the analysis]
Aux feed water cooling inlet AOV	Valve internals removed	Valve is no longer credited on the seismic SCPL

Attachment 5: 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	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.

The following is an optional example attachment

Attachment 6: Comparison of RG 1.200 Revision 1 and Revision 2 SRs Applicable to CC-I/II, CC-II/III, and CC-I/II/III

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/II/III: … 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 supercede 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 realistic 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

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
			not impact the categorization results.
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-**	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 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
knowledge" correlation between event probabilities is taken into account.			
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-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 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-E3: **CC-I:** **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).** **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)**	QU-E3: CC-I: 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).** CC-II: 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,**	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-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 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
	IE-C15)....(no change)		
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 sensitivity analyses Note (1).** **CC-III:** **EVALUATE the sensitivity of the results to uncertain model boundary conditions and other assumptions using sensitivity**	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
analyses except where such sources of uncertainty have been adequately treated in the quantitative uncertainty analysis Note (1).			
LE-F2: **CC-I:** **PROVIDE a qualitative assessment of the key sources of uncertainty.** **Examples:** **(a) Identify bounding assumptions.** **(b) Identify conservative treatment of phenomena.** **CC-II:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the**	LE-F3: CC-I/II/III: 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
significant contributors to LERF. **CC-III:** **PROVIDE uncertainty analysis <u>that</u> identifies the key sources of uncertainty and includes sensitivity studies.**			
IF-F2: **<u>CC-I/II/III</u>:** **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: <u>CC-I/II/III</u>: 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

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
			walkdown documentation was updated to meet the requirements of the standard and the new walkdown information was evaluated to determine that it had no impact of the Flood PRA model] OR [the flood walkdown 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: CC-I/II/III: 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	IFSO-A1 : CC-I/II/III: 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	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	[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

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
water system, service water system, fire protection system, component cooling water system, feedwater system, condensate and steam systems)	system, fire protection system, component cooling water system, feedwater system, condensate and steam systems, and reactor coolant system) ...	flooding source was added to the 2009 version of the PRA Standard.	that the fire protection and RCS systems are included in the flood model] OR [The fire protection and RCS were added as sources of flooding to the flood model]
IF-F2 **CC-I/II/III:** **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**	IFSO-F2 CC-I/II/III: 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	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
.... **(j) calculations or other analyses used to support or refine the flooding evaluation**	support or refine the flooding evaluation any walkdowns performed in support of identification or screening of flood sources		
IF-F2 **CC-I/II/III:** **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**	IF-F2 CC-I/II/III: 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	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*	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
equipment operability **...** **(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** **...**	effects on equipment operability ... (d) screening criteria used in the 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	*peer review conducted using that version of the PRA Standard.*	

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
IF-F2 **CC-I/II/III:** **DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes:** **...** **(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**	IF-F2 CC-I/II/III: DOCUMENT the process used to define the applicable internal flood accident sequences and their associated quantification. For example, this documentation typically includes: ... (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	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 related features considered in flood sequence quantification.	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, 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
... **(k) results of the internal flood analysis, consistent with the quantification requirements provided in HLR-QU-D**	requirements provided in HLR-QU-D ... (e) any walkdowns performed in support of internal flood accident sequence quantification		

Attachment 7: 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 and 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.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Landslide	Y	C3	Not applicable to the site because of topography.
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.

External Hazard	Screening Result		
	Screened? (Y/N)	Screening Criterion (Note a)	Comment
Volcanic Activity	Y	C3	Not applicable to the site because of 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 Table 7 for descriptions of the screening criteria.			

Attachment 8: 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	

Event Analysis	Criterion	Source	Comments
	PS4. Bounding mean CDF is < 1E-6/y.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	
Detailed PRA	Screening not successful. PRA needs to meet requirements in the ASME/ANS PRA Standard.	NUREG-1407 and ASME/ANS Standard RA-Sa-2009	

Attachment 9: Disposition of Key Assumptions/Sources of Uncertainty

Assumption / Uncertainty	Discussion	Disposition
Loss of Coolant Accident (LOCA) Frequencies	NUREG/CR-6928 restated the results from NUREG-1829. The LOCA frequencies are based upon expert elicitations. The LOCA sizes identified by the NRC are different from those estimated for PVNGS.	The slight variance in the range of break sizes for different LOCAs is not significant and is judged to have minimal impact on LOCA frequencies, within the uncertainties associated with the expert elicitation values, and of insignificant impact. Therefore, no sensitivity analysis is required for this application.
The Seismic PRA HFE dependency analysis	The Seismic PRA dependency analysis assumes that once an accident sequence is initiated, the operator action timing for a seismically induced event is similar to that of an internally induced event for main control room actions.	The modification of the timing available due to seismic considerations may result in a longer response or identification time and consequently a higher HEP. A sensitivity analysis was performed in the seismic PRA quantification increasing the failure probability all HEPs to 1.0, resulting in a 39.36% increase in CDF. For this application, an additional sensitivity analysis will be performed increasing all HEPs derived from the internal events PRA model in the seismic PRA quantification by a factor of 3 to address this uncertainty.