Enclosure 5

50.59 License Amendment Request Submittal Template
Prepared by NEI Risk-Informed Engineering Programs
(50.69) Task Force

Meeting Summary of the 8/18/2016 Public Meeting

DATED September 12, 2016

50.69 License Amendment Request Submittal Template

Prepared by the NEI Risk Informed Engineering Programs (50.69) Task Force

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.

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.

[DATE] 10 CFR 50.90

U. S. Nuclear Regular 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 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.

Enclosure 1 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."

[PREPARER'S NOTE: If applicable, include one of the two following paragraphs if it is possible to streamline the review of the PRA model in this application using the approval 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 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.

Or

[LICENSEE] intends to submit a license amendment request for [identify application] within the next [X months] and requests that the NRC review the PRA technical adequacy description in this application for both applications. This would reduce the number of [LICENSEE] and NRC resources necessary to complete the review of the applications.]

[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.

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

This letter contains no NRC commitments.

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

Sincerely,

Signature

Enclosure:

1. Basis for Proposed Change

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

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1 SUMMARY DESCRIPTION

The proposed amendment would 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.

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. Special treatment includes, but is not limited to, quality assurance, testing, inspection, condition monitoring, assessment, evaluation, and resolution of deviations. 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 Assessment (PRA) addresses 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 2), 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 how SSC is categorized. Finally, assessment activities are conducted to make adjustments to the categorization and 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 by restructuring the regulations to allow an alternative risk-informed approach to special treatment. Conversely, for SSCs that do not significantly contribute to plant safety on an individual basis, the rule allows a 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, include one of the two following paragraphs if it is possible to streamline the review of the PRA model in this application using the approval 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.]

The NRC has previously reviewed the technical adequacy of the [PLANT NAME] Probabilistic Risk Assessment (PRA) model for [purpose] in [identify previous application where the PRA model technical adequacy was reviewed by the NRC, including date and ADAMS Accession Number]. The risk analyses described in this application utilize the same PRA model[s] described in the other application. Therefore, [LICENSEE] requests that the NRC utilize the review of the PRA technical adequacy for that application when performing the review for compliance with 10 CFR 50.69(b)(2)(ii) and 10 CFR 50.69(b)(2)(iii) in 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 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 it is possible to issue them in any order and without regard to the results of 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 current revision or any subsequent revisions of NEI 00-04, as endorsed by RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," (Reference 1). 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 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: Safe shutdown equipment list 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.

3.1.2 Passive Categorization Process

Passive components are defined in NEI 00-04 as SSCs having only a pressure retaining function. Active components can also have a passive or pressure retaining function. Therefore, the term "passive component" also refers to the passive function of active components, if applicable. NEI 00-04 states that passive component categorization should be performed using the guidance of ASME Code Case N-660 Revision 0, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," (Reference 4) or subsequent versions approved by ASME, in lieu of this guidance.

An updated methodology was approved by NRC for use in risk informed repair/replacement activities for Arkansas Nuclear One, Unit 2, as documented in the Safety Evaluation, dated April 22, 2009 (Reference 5). Subsequently, the same process was approved by NRC for use in 10 CFR 50.69 as documented in the final Safety Evaluation for Vogtle dated December 17, 2014 (Reference 6).

[LICENSEE] will use the same passive categorization method as described and approved in Arkansas Nuclear One, Unit 2, Safety Evaluation described above.

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.

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. Table 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 7) 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 7) 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 Table X. In addition, screened scenarios were reviewed and changes to credited functions or SSCs required to perform those functions are also identified in Table 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. Table 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 7) for evaluation of safety significance related to seismic hazards. An evaluation was performed of the as-built, as-operated plant against the seismic success paths 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 7) for evaluation of safety significance related to seismic hazards. An evaluation was performed of the as-built, as-operated plant against the seismic success paths identified in the SMA and changes to the success paths are identified in Table 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. Table 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 7) for evaluation of safety significance related to other external 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. Table[s] X at the end of this enclosure identifies the applicable other external hazard PRA model[s].

3.2.5 Low Power & Shutdown

The [PLANT NAME] categorization process will use the shutdown safety management plan described in NUMARC 91-06 (Reference 3), 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.

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 process discussed in Section 8 of NEI 00-04.

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 8). 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 NEI 00-02 (Reference 9), NEI 05-04 (Reference 10), and NEI 07-12 (Reference 11).

Table 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 including open findings in which NRC's review for closure is requested.
- Identification of and basis for any sensitivity analysis needed to address open findings.

Not required if Peer Review has been completed to 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 Table X.

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 Section 8 of the guidance 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. NRC Regulatory Guide 1.201 (for Trial Use), "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006.
2. NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline," Revision 0, Nuclear Energy Institute, July 2005.
3. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December, 1991.
4. ASME, "Risk-Informed Safety Classification for Use in Risk-Informed Repair/Replacement Activities," Code Case N-660, Revision 0, Section XI, Division 1, American Society of Mechanical Engineers, 2004.
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) dated April 22, 2009.
6. Vogtle Electric Generating Plant, Units 1 And 2 -Issuance Of Amendments Re: Use Of 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473) (ML14237A034) dated December 17, 2014.
7. 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.
8. 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.
9. NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Process Guidance," Nuclear Energy Institute, 2000.
10. NEI 05-04," Process for Performing Internal Events Peer Reviews Using the ASME/ANS PRA Standard," Revision 2, Nuclear Energy Institute, November 2008.
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.

Table 1: Internal Events and Internal Flooding

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)

Similar tables should be used for any additional PRA models (e.g. Fire PRA and Seismic PRA)

Table 2: 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

Table 3: Changes to Seismic Success Paths in SMA

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 success path

Table 4: 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.
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 table

Table 5: 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 not

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
			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:**	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	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
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.	uncertainty distributions, ensuring that the "state-of-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**	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	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
may not be transferred between event trees.	be transferred between event trees.		
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). **<u>CC-II:</u>** **ESTIMATE the uncertainty interval of the CDF results.** ESTIMATE the uncertainty intervals associated with parameter uncertainties (DA-	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> ESTIMATE the uncertainty interval of the CDF (and LERF) results. **ESTIMATE the uncertainty intervals**	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
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)**	**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-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.**	LE-F3: <u>CC-I/II/III:</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
(b) Identify conservative treatment of phenomena. **<u>CC-II</u>:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.** **<u>CC-III</u>:** **PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies.**			

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: **<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 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

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
			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,**	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

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
condensate and steam systems)			[The fire protection and RCS were added as sources of 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**	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	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
... **(f) screening criteria used in the analysis** **....** **(j) calculations or other analyses used to support or refine the flooding evaluation**	calculations or other analyses used to support or refine the flooding evaluation any walkdowns performed in support of identification or 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...** **...**	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...	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	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
(d) accident mitigating features and barriers credited… **…** **(e) assumptions or calculations used in the determination of …flood-induced effects on 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…** **….**	… (c) assumptions or calculations used in the determination of …flood-induced 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 …	PRA Standard, it was *not reviewed as part of the 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
(j) 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		
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 process to identify flood related features considered in flood	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

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	sequence quantification.	areas that are considered in flood sequence definition.