10 CFR 50.90
10 CFR 50.69

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July 19, 2017

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

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station Units 1, 2, and 3**
Docket Nos. STN 50-528, 50-529, and 50-530
Renewed Operating License Nos. NPF-41, NPF-51, NPF-74
License Amendment Request to Adopt 10 CFR 50.69*, Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors*

In accordance with the provisions of Section 50.69(b)(2) and 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Arizona Public Service Company (APS) is requesting an amendment to the renewed operating license of Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3.

The proposed license amendment request (LAR) would modify the licensing basis, by the addition of a License Condition, to allow for the implementation of the provisions of 10 CFR, Part 50.69, *Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors*. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has high safety significance resulting in improved plant safety.

The enclosure to this letter provides the basis for the proposed change to the PVNGS Units 1, 2, and 3 renewed operating licenses. The categorization process being implemented through this change is consistent with Nuclear Energy Institute (NEI) 00-04, *10 CFR 50.69 SSC Categorization Guideline*, Revision 0, dated July 2005, which was endorsed by the Nuclear Regulatory Commission (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 structure, system, or component will only occur after these prerequisites are met.

The PVNGS Probabilistic Risk Assessment (PRA) models are described in Attachments 2 through 5 of the enclosure. The PRA models described within this LAR are the same as those described within the APS submittal of the LAR dated July 31, 2015, to revise the PVNGS Technical Specifications (TS) to allow risk-informed completion times [Agencywide Document Access and Management System (ADAMS) Accession Number ML15218A300], with routine maintenance updates applied. APS has also recently conducted a facts and observations (F&O) closure review of PRA peer review findings in accordance with an NRC letter dated May 3, 2017 (ADAMS Accession Number ML17079A427).

APS requests that the NRC staff utilize insights from their on-going review of the technical adequacy of PRA models in the risk-informed completion times LAR as well as the F&O closure review results to inform their review of the same PRA models for 10 CFR 50.69. This would reduce the number of APS 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 can be independently reviewed on their own merits without regard to the results from the review of the other.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and the Offsite Safety Review Committee have reviewed and approved this LAR. By copy of this letter, this LAR is being forwarded to the Arizona Radiation Regulatory Agency in accordance with 10 CFR 50.91(b)(1).

APS requests approval of the proposed license amendment within one year of the date of this letter, with the amendment being implemented within 90 days of issuance. No new commitments are made by this letter. Should you have any questions concerning the content of this letter, please contact Michael DiLorenzo, Licensing Section Leader, at (623) 393-3495.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: July 19, 2017
(Date)

Sincerely,

Lacal, Maria L(Z06149)

Digitally signed by Lacal, Maria L(Z06149)
DN: cn=Lacal, Maria L(Z06149)
Date: 2017.07.19 16:13:27 -07'00'

MLL/TNW/NTA

Enclosure: Description and Assessment of Proposed License Amendment for Categorization and Treatment of Structures, Systems, and Components

cc:	K. M. Kennedy	NRC Region IV Regional Administrator
	S. P. Lingam	NRC NRR Project Manager for PVNGS
	M. M. Watford O'Banion	NRC NRR Project Manager
	C. A. Peabody	NRC Senior Resident Inspector for PVNGS
	T. Morales	Arizona Radiation Regulatory Agency (ARRA)

Enclosure

Description and Assessment of Proposed License Amendment for Categorization and Treatment of Structures, Systems, and Components

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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 for Nuclear Power Reactors*. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance (LSS), alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance (HSS), requirements will not be changed or will be enhanced. This allows improved focus on equipment that has high 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 structures, systems and components (SSCs) necessary to defend against the DBEs are defined as *safety-related*, and these SSCs are the subject of many regulatory requirements, herein referred to as *special treatments*, designed to ensure that they are of high quality and high reliability, and have the capability to perform during postulated design basis conditions. Treatments include, but are 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 high 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 Nuclear Energy Institute (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 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 functional 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 a reasonable, though reduced, level of confidence that these SSCs will satisfy functional requirements.

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

2.3 Description of the Proposed Change

APS proposes the addition of the following condition to the PVNGS renewed operating licenses of Units 1, 2, and 3 to document NRC approval to use 10 CFR 50.69.

> APS 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 10 CFR 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. (See section 3.1 of this enclosure)
>
> (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. (See section 3.2 of this enclosure)
>
> (iii) Results of the PRA review process conducted to meet § 50.69(c)(1)(i). (See section 3.3 of this enclosure)
>
> (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). (See section 3.4 of this enclosure)

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

The PRA models described within this LAR are the same as those described within the APS submittal of the LAR dated July 31, 2015, to revise the PVNGS Technical Specifications (TS) to allow risk-informed completion times (Reference 9), with routine maintenance updates applied. APS has also recently conducted a F&O closure review of PRA peer review findings in accordance with an NRC letter dated May 3, 2017 (Reference 17).

APS requests that the NRC staff utilize insights from their on-going review of the technical adequacy of PRA models in the risk-informed completion times LAR as well as the F&O closure review results to inform their review of the same PRA models for 10 CFR 50.69. This would reduce the number of APS 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 can be independently reviewed on their own merits without regard to the results from the review of the other.

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

3.1.1 Overall Categorization Process

APS will implement the risk categorization process in accordance with NEI 00-04, Revision 0 (Reference 2), as endorsed by Regulatory Guide (RG) 1.201, Revision 1, *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 following are the clarifications 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 HSS or LSS pursuant to § 50.69(f)(1) will be documented in APS 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 HSS.

- Passive characterization will be performed using the processes described in Section 3.1.2 of this enclosure.

- 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.

- APS will require that if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6 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 preliminarily identified as HSS. The IDP must intervene to assign any of these HSS function components to LSS.

- With regard to the criterion that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, APS will not take credit for alternate means unless the alternate means are proceduralized and included in licensed operator training.

The risk analysis being implemented for each hazard is described below:

- Internal Event Risks: Internal events including internal flooding PRA model (same as described in Reference 9).

- Fire Risks: Internal Fire PRA model consistent with NUREG/CR-6850 (Reference 10) methodology (same as described in Reference 9).

- Seismic Risks: Seismic PRA model (same as described in Reference 9).

- Other External Risks (e.g., tornados, external floods, etc.): Screened out as not requiring PRA models as described in Reference 9, as 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, *Guidelines for Industry Actions to Assess Shutdown Management* (Reference 3), which provides guidance for assessing and enhancing safety during shutdown operations.

A change to the categorization process that is outside the bounds specified above (e.g., change from a seismic margins approach to a seismic probabilistic risk assessment approach) will not be used without prior NRC approval. The SSC categorization process documentation will include the following elements:

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

3.1.2 Passive Categorization Process

For the purposes of 10 CFR 50.69 categorization, passive components are those components that have a pressure retaining function. Passive components and the passive function of active components will be evaluated using the Risk-Informed Repair/Replacement Activities (RI-RRA) methodology consistent with the Safety Evaluation (SE) by the Office of Nuclear Reactor Regulation for Arkansas Nuclear One,

Unit 2, regarding their *Request to Use Risk-informed Safety Classification and Treatment for Repair/Replacement Activities in Class 2 and 3 Moderate and High Energy Systems,* dated April 22, 2009 (Reference 5).

The RI-RRA methodology is a risk-informed safety classification and treatment program previously used for repair/replacement activities (RI-RRA methodology) for pressure retaining items and their associated supports. In this method, the component failure is assumed with a probability of 1.0 and only the consequence evaluation is performed. It additionally applies deterministic considerations (e.g., defense in depth, safety margins) in determining safety significance. Component supports are assigned the same safety significance as the highest passively ranked component within the bounds of the associated analytical pipe stress model.

The use of this method was previously approved to be used for a 10 CFR 50.69 application by the NRC in the final SE for Vogtle Electric Generating Plant dated December 17, 2014 (Reference 6). The RI-RRA method as approved for use at Vogtle Electric Generating Plant for 10 CFR 50.69 does not have any plant specific aspects and is generic. It relies on the conditional core damage and large early release probabilities associated with postulated ruptures. Safety significance is generally measured by the frequency and the consequence of the event. However, this RI-RRA process categorizes components solely based on consequence, which measures the safety significance of the passive component given that it ruptures. This approach is conservative compared to including the rupture frequency in the categorization as this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. Therefore, the RI-RRA methodology for passive categorization is acceptable and appropriate for use at PVNGS for 10 CFR 50.69.

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

The following sections demonstrate that the quality and level of detail of the processes used in categorization of SSCs are adequate. The PRA models described below have been peer reviewed and there are no PRA upgrades that have not been peer reviewed. The PRA models described within this LAR are the same as those described within the APS submittal of the LAR dated July 31, 2015, to revise the PVNGS TS to allow risk-informed completion times (Reference 9), with routine maintenance updates applied. Changes and plant modifications previously identified in Reference 9 that were required to achieve an overall Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) consistent with NRC Regulatory Guide 1.174 (Reference 16) have been completed.

3.2.1 Internal Events and Internal Flooding

The PVNGS categorization process for the internal events and internal flooding hazards will use peer reviewed plant-specific Internal Events and Internal Flooding PRA models in accordance with RG 1.200, Revision 2 (Reference 7). The APS risk management process ensures that the PRA models used in this application reflects the as-built and as-operated plant for each of the PVNGS units. Only industry consensus methods were utilized in the development of the Internal Events and Internal Flooding PRA models. Attachment 2 of this enclosure identifies the Baseline Average Annual CDF and LERF for the Internal Events and Internal Flooding PRA models.

3.2.2 Fire Hazards

The PVNGS categorization process for fire hazards will use a peer reviewed plant-specific Internal Fire PRA model in accordance with RG 1.200, Revision 2 (Reference 7). The APS risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the PVNGS units. Industry consensus methods were utilized in the development of the Internal Fire PRA model. While APS was not an applicant to implement National Fire Protection Association Standard (NFPA) 805 in accordance with 10 CFR 50.48, the Internal Fire PRA model was developed consistent with NUREG/CR-6850 and only utilizes NRC approved methods. As part of the ongoing PRA maintenance and update process described in Section 3.2.6, APS will address Internal Fire PRA methods approved by the NRC since the development of the Internal Fire PRA. Note that APS does not credit incipient fire detection systems in the Internal Fire PRA model. Attachment 2 of this enclosure identifies the Baseline Average Annual CDF and LERF for the Internal Fire PRA model.

3.2.3 Seismic Hazards

The PVNGS categorization process for seismic hazards will use a peer reviewed plant-specific Seismic PRA model in accordance with RG 1.200, Revision 2 (Reference 7). The APS risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for each of the PVNGS units. Only industry consensus methods were utilized in the development of the seismic hazards for the seismic PRA. Attachment 2 of this enclosure identifies the Baseline Average Annual CDF and LERF for the Seismic PRA model.

3.2.4 Other External Hazards

The PVNGS categorization process for the external hazards will use a peer reviewed plant-specific screening in accordance with RG 1.200, Revision 2 (Reference 7). Each external hazard was evaluated with respect to applicability and/or risk. The ASME PRA Standard RA-Sa-2009, *Standard for Level l/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications* (Reference 15) outlines preliminary and progressive screening approaches that are acceptable for this task. The screening started with the top approach and progressed downward until the hazard in question screened with respect to risk. If none of the screening approaches were successful, then the hazard was analyzed using a detailed PRA approach that meets applicable requirements in the ASME PRA Standard RA-Sa-2009. Implicit in these screening criteria (ones that do not present a quantitative measure) is the assumption that successfully meeting a criterion for screening indicates that the bounding CDF from that hazard is considered to be lower than 1E-6 per year. Attachment 4 provides a summary of the other external hazards screening results. Attachment 5 provides a summary of the progressive screening approach for addressing external hazards.

3.2.5 Low Power & Shutdown

The PVNGS 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 APS risk management process ensures that the applicable PRA models used in this application continue to reflect the as-built and as-operated plant for each of the PVNGS units. The process delineates the responsibilities and guidelines for updating the PRA model, and includes criteria for both regularly scheduled and interim PRA model updates. The process includes provisions for monitoring potential areas affecting the PRA model (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, APS will implement a process that addresses the requirements in NEI 00-04, Section 11, *Program Documentation and Change Control* (Reference 2). 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 models 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 processes are addressed using the process discussed in Section 8 and in the prescribed sensitivity studies discussed in Section 5 of NEI 00-04 (Reference 2).

In the overall risk sensitivity studies, APS will utilize a factor of 3 to increase the unavailability or unreliability of low safety significant (LSS) components consistent with that approved by the NRC in the Vogtle Electric Generating Plant 10 CFR 50.69 License Amendment Safety Evaluation Report (Reference 6). Consistent with the NEI 00-04 guidance, APS 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 PVNGS PRA models using the guidance of NUREG-1855, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making* (Reference 11) and

EPRI TR-1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments* (Reference 12).

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 PVNGS 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 risk ranking calculations were considered key for this application.

Key PVNGS PRA model specific assumptions and sources of uncertainty for this application were evaluated and documented. These key assumptions and sources of uncertainty reviewed were previously submitted to the NRC in the application dated July 31, 2015 (Reference 9) for risk-informed completion times. The conclusion of the review for this application is that no additional sensitivity analyses are required to address PVNGS PRA model specific assumptions or sources of uncertainty except for the following:

- In the process of categorizing SSCs into risk-informed safety classifications, APS will include in the risk sensitivity study 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 models described in Section 3.2 have been assessed against RG 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2 (Reference 7) consistent with NRC RIS 2007-06, *Regulatory Guide 1.200 Implementation* (Reference 24).

The Internal Events PRA model was peer reviewed in July 1999 by the Combustion Engineering Owners Group (CEOG) prior to the issuance of Regulatory Guide 1.200 (Reference 19). As a result, a self-assessment of the Internal Events PRA model was conducted by APS in March 2011 (Reference 20) in accordance with Appendix B of RG 1.200, Revision 2 (Reference 7), to address the PRA quality requirements not considered in the CEOG peer review. APS conducted a full scope Internal Flooding PRA model peer review (Reference 21) in November 2010, in accordance with RG 1.200, Revision 2 (Reference 7).

The Internal Events PRA quality (including the CEOG peer review and self-assessment results) has previously been reviewed by the NRC in requests to extend the Inverter Technical Specification Completion Time dated September 29, 2010 (Reference 13), and to implement TSTF-425, *Relocate Surveillance Frequencies to Licensee Control RITSTF Initiative 5b*, December 15, 2011 (Reference 14). All PRA upgrades (as defined by the ASME PRA Standard RA-Sa-2009 [Reference 15]) implemented since conduct of the CEOG peer review in 1999 have been peer reviewed.

APS conducted a full scope Internal Fire PRA model peer review (Reference 22) in December 2012 in accordance with RG 1.200, Revision 2 (Reference 7). APS conducted a second focused scope peer review of the Internal Fire PRA in December 2014 (Reference 23), to address ASME PRA Standard (Reference 15) supporting requirements determined not met to Capability Category II in the first peer review, not just the associated facts and observations (F&Os) from the first peer review. Thus, the second peer review generated new F&Os which replaced in their entirety the finding level F&Os from the first peer review.

APS conducted a full scope Seismic PRA model peer review (Reference 4) in February 2013, in accordance with RG 1.200, Revision 2 (Reference 7). APS conducted a full scope External Hazards screening peer review (Reference 25) in December 2011, in accordance with RG 1.200, Revision 2 (Reference 7).

An F&O closure peer review was performed in June 2017, in accordance with NRC letter dated May 3, 2017 (Reference 17) to assess the closure of all finding level F&Os from these peer reviews (Reference 18) that were not otherwise addressed by focused scope peer reviews that re-reviewed the associated ASME PRA supporting requirements in their entirety. The F&O closure review was conducted to ensure the findings had been satisfactorily resolved to meet the ASME PRA Standard RA-Sa-2009 (Reference 15) to Capability Category II, the sub-element criteria for the CEOG from Internal Events PRA peer review (Reference 19), and RG 1.200, Revision 2 (Reference 7).

The F&O closure peer review assessed the sixty finding level F&Os from the prior peer reviews and concluded that all were closed with the exception of eight findings. Of the eight not closed findings, six were assessed as partially closed, and two were assessed as open. The eight not closed findings and their dispositions are described in Attachment 3.

APS will resolve the eight not closed finding level F&Os listed in Attachment 3 and validate closed by a subsequent F&O closure review conducted in accordance with NRC letter dated May 3, 2017 (Reference 17). These not closed finding level F&Os will be closed prior to utilizing the PRA models for categorization. Resolution of the not closed finding level F&Os is not expected to have a significant impact on overall CDF or LERF based on the dispositions described in Attachment 3.

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

The PVNGS 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04 (Reference 2). 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 (Reference 2) will be used to confirm that the categorization process results in acceptably small increases to CDF and LERF. The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, 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

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 (Reference 1).
- 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 (Reference 16).
- Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, March 2009 (Reference 7).

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

4.2 No Significant Hazards Consideration

APS proposes to 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 Reactors*. The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements will not be changed or will be enhanced. This allows improved focus on equipment that has high safety significance resulting in improved plant safety.

APS 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(s). 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 the analyses of accidents are not affected by the proposed change. 10 CFR 50.69 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, APS 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, *Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance*, Revision 1, dated May 2006
2. NEI 00-04, *10 CFR 50.69 SSC Categorization Guideline*, Revision 0, Nuclear Energy Institute, dated July 2005
3. NUMARC 91-06, *Guidelines for Industry Actions to Assess Shutdown Management*, dated December 1991
4. Westinghouse Letter to APS CVER-13-028, *Transmittal of Palo Verde Seismic PRA – Final Peer Review Report*, February 14, 2013
5. NRC Safety Evaluation (SE) 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* (ADAMS Accession Number ML090930246), dated April 22, 2009
6. Vogtle Electric Generating Plant, Units 1 And 2 - Issuance of Amendments Re: *Use of 10 CFR 50.69* (ADAMS Accession Number ML14237A034), dated December 17, 2014
7. Regulatory Guide 1.200, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*, Revision 2, dated March 2009
8. NEI 00-02, *Probabilistic Risk Assessment (PRA) Peer Review Process Guidance*, Nuclear Energy Institute, dated 2000
9. License Amendment Request to Revise Technical Specifications to implement Risk-Informed Completion Times (ADAMS Accession Number ML15218A300), dated July 31, 2015
10. NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, dated September 2005
11. NUREG-1855, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making*, dated March 2009
12. EPRI TR-1016737, *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments*, dated December 2008

13. *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Issuance of Amendments Re: Changes To Technical Specification 3.8.7, "Inverters-Operating"* (ADAMS Accession Number ML102670352), dated September 29, 2010

14. *Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Issuance of Amendments Re: Adoption of TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control RITSTF Initiative 5b"* (ADAMS Accession Number ML112620293), dated December 15, 2011

15. 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

16. 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, dated April 2015

17. NRC letter (ADAMS Accession No. ML17079A427) dated May 3, 2017

18. ABS Consulting Report R-3882824-2037, *Palo Verde Generating Stations PRA Finding Level Fact and Observation Closure Review*, June 23, 2017

19. ABB Combustion Engineering letter to APS ST-99-542, *PSA Peer Review for Palo Verde Nuclear Generating Station*, July 12, 1999

20. Palo Verde Engineering Evaluation 3579223, *PRA input to the Risk-Informed Task Force (RITS) 5b license amendment,* March 10, 2011

21. Westinghouse Letter to APS LTR-RAM-II-10-082, Rev. 0, *Internal Flood Focused Scope RG 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements for the Palo Verde Nuclear Generating Station Probabilistic Risk Assessment*, November 2010

22. Westinghouse Letter to APS LTR-RAM-12-13, *Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS PRA Standard for the Palo Verde Nuclear Generating Station Fire Probabilistic Risk Assessment*, January 2, 2013

23. Hughes Associates Report 001014-RPT-01, *Palo Verde Nuclear Generating Station Fire PRA Focused-Scope Peer Review*, January 22, 2015

24. *NRC Regulatory Issue Summary 2007-06 Regulatory Guide 1.200 Implementation*, (ADAMS Accession No. ML070650428) dated March 22, 2007

25. Palo Verde Nuclear Generating Station Other External Hazards PRA Peer Review Report, December 2011

Attachment 1

List of Categorization Prerequisites

- APS will resolve the eight not closed finding level facts and observations (F&O) listed in Attachment 3 and validate them closed by a subsequent facts and observation (F&O) closure review conducted in accordance with NRC letter dated May 3, 2017 (Reference 17). The not closed finding level F&Os will be closed prior to utilizing the PRA models for categorization.

- APS 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:

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

  2. Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary high safety significant (HSS) or low safety significant (LSS) based on the seven 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.

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

  4. Assessment of 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.

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

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

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

  8. Documentation requirements per Section 3.1.1 of this enclosure.

Attachment 2

Total Unit 1/2/3 Baseline Average Annual CDF/LERF

Hazard	CDF (per reactor-year)	LERF (per reactor-year)
Internal events	1.3E-6	4.3E-8
Internal flooding	4.6E-7	2.1E-8
Seismic	3.1E-5	5.7E-6
Internal Fire	2.9E-5	2.4E-6
Total	6.2E-5	8.2E-6

Notes
1. Total CDF meets the RG 1.174 acceptance guideline of < 1E-4 per year
2. Total LERF meets the RG 1.174 acceptance guideline of < 1E-5 per year

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
IE-07 (B) / IE-12 *Internal Events*	The Interfacing Systems Loss of Coolant Accident (ISLOCA) treatment for the shutdown cooling suction line appears to have some questionable assumptions. First, it is assumed that the Low Temperature Over Pressure (LTOP) valve would always open. While this is the most likely scenario, the LTOP valve can fail to open. Qualitative arguments were made that should this happen, the resulting LOCA would be inside containment (primarily based on relative pipe lengths). This ignores the fact that the high stress points and stress concentration points are outside containment. Furthermore, the shutdown cooling warmup crossover piping was not considered.	Status: Partially Closed. Basis: Insufficient justification is provided in "Impact 200- 84.pdf" to demonstrate that the frequency of the scenario in question is negligible. The resolution of this finding only provides qualitative argument that this ISLOCA scenario will require failure of two Motor Operated Valves (MOVs), failure to open of the LTOP valve, and failure of the warmup piping or the bypass valve. No quantitative values and LTOP valve capacity were provided to demonstrate that the frequency of this scenario is negligible. Note that the likelihood of failure of the piping outside containment is relatively high. Furthermore, it is not clear if the capacity of the LTOP valve is sufficient to relieve the relatively large flow that may result from the catastrophic ruptures of the two upstream MOVs. Finally, ISLOCA may also result from leakage of both of the two upstream MOVs, or a combination of leakage and rupture of the two upstream MOVs, in conjunction with failure of the LTOP valve to open and failure of the downstream piping. This scenario would have a greater frequency than catastrophic ruptures of both MOVs because the frequency of MOV leakage is significantly greater than its catastrophic rupture. Recommendation: Provide additional justifications (including the capacity of the LTOP valve, all of the possible failure mode combinations and their probabilities of occurrence, LERF value, etc.) to demonstrate that the frequency of the scenario in question is indeed negligible compared to the LERF.	The Closure Review Team recommendation will be addressed by evaluating all ISLOCA failure modes of the shutdown cooling system piping. Justification for screening out any negligible scenarios will be provided. Leakage, spurious operation, and catastrophic failure modes of valves will be considered, as well as the LTOP relief valve failure to open or exceedance of its relief capacity. These changes are not expected to have a significant impact on total CDF or LERF since the current internal events contribution to total CDF and LERF is less than 2% and 0.5%, respectively. While the resulting LERF contribution from these scenarios may not be negligible, they are expected to be minimal based on industry operating experience. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
AS-03 (B) / AS-24 *Internal Events*	There are some differences between treatment of a small LOCA associated with a pipe break and an induced small LOCA (pressurizer safety valve reclosure) in the transient event trees. For example: • In the small LOCA event tree, successful high pressure injection and recirculation lead to questioning whether containment heat removal is successful. In the Transient Type 2 and Transient Type 3 event trees, RCS integrity can be lost if pressurizer safety valves do not reset after lifting. In the sequences from these event trees where high pressure injection and recirculation are successful, the question relating to containment heat removal is not asked.	Status: Open. Basis: Containment heat removal (CHR) is only asked in the small LOCA event tree when the success path is relying on high pressure sump recirculation (HPSR) with a failure of the operators to depressurize and cool down with successful SG heat removal. In this case the RCS remains at temperature so that there is substantial heat transfer to the containment. Table 4 of 13-NS-B065 R007 presents MAAP results for LOCA cases with failure of CHR from spray recirculation. Based on a reply to the reviewer's question, Table 4 indicates that <2" diameter breaks may just require CHR because s1_2_1a with SG cooling and failure of SG depressurization exceeds a containment pressure of 50 psig at just 6.3 hours. Smaller holes as represented by case s1_1_1a for a 1" break do not exceed even 50 psig until 22.6 hours. The ultimate containment failure pressure is 141 psig (i.e. 50% chance of failure) so assuming failure at 50 psig as a basis for success is conservative. It is therefore also conservative to assume that all small LOCA sizes (3/8" to 2.35") require CHR under these circumstances. The small LOCA event tree may also need to ask CHR in cases where the SGs are not depressurized; i.e. sequences 1 and 3. For a loss of main feedwater pumps (Type2), containment heat removal is not asked for any sequences. Either SG cooling prevents the PSV from lifting at all so there is no LOCA or the operators depressurize the RCS for alternate AFW (at low pressure) though the PSVs are assumed to lift and may fail to reseat. Failure of at least one PSV to reseat (equivalent hole size of 2.3") requires HPSI but the SGs are at low temperature in this scenario so CHR is currently assumed not required for the first 24 hours. Consideration should be given to containment failure at later times which may lead to subsequent core damage due to failure of sump recirculation at the time of containment failure.	The Closure Review Team recommendation will be addressed by modeling CHR in the small LOCA event tree, and event scenarios with failure of the PSV to reseat. MAAP analyses will be performed to include PSV failure to reseat in the small break sizes to determine the necessity of CHR for long-term stable end-state. These changes are not expected to have a significant impact on total CDF or LERF since the current internal events contribution to total CDF and LERF is less than 2% and 0.5%, respectively. The likelihood of a small LOCA with a loss of CHR for longer than 24 hours is very small. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
AS-03 (B) / AS-24 *Internal Events* (cont.)	• In the small LOCA event tree, RCS depressurization and use of low pressure injection and recirculation are considered if high pressure injection or recirculation fail. In the Transient Type 2 and Transient Type 3 event trees, consideration of RCS depressurization and use of low pressure systems is not included because the likelihood of high pressure injection or high pressure recirculation are small. It would seem that this assumption should apply to both cases, or not.	For a loss of condenser vacuum (Type3), it's similar to the type2 event tree. Therefore, for LOCA scenarios with a hole size no larger than 2.3" equivalent in diameter, with or without SG depressurization, further justification is needed to not require CHR to protect the containment. The following information is useful in reviewing the documents associated with this F&O. Page 337, Figure 3.27.4 of 13-NS-B061, Revision 5 is the small LOCA event tree. Type 2 initiators with RT before turbine trip only challenge the PSVs if all SG cooling is lost. Event tree for loss of main feedwater pumps in 3.9.4 on Page 130 shows that containment heat removal is not asked because if secondary heat removal is lost, core damage is assumed. Type 3 initiators are where turbine trips first and may challenge the PSVs. Figure 3.6.4 on page 102 shows the loss of condenser vacuum ET which is type 3 initiator. Containment heat removal is not asked. Type 2 and type 3 initiators presently have about the same contribution to CDF for internal events, as does small LOCA; i.e., around 12%. <u>Recommendation</u>: Perform a set of MAAP sensitivity analyses assuming a stuck open PSV with equivalent hole diameter of 2.3" to investigate the possibility of success without CHR. Expand the discussions in the MAAP report to better describe the basis for the 2" hole size as the critical break size. Further, the assumed mission time of 24 hours may be too short for consideration of containment failure. A loss of CHR that results in an exceedance of the pressure capacity at 48 hours is not a stable state at 24 hours and is still of concern. In other words, breaks with sizes smaller than 2" may also require CHR under the circumstances postulated in this F&O. One possibility is to add the events asking for CHR to the event trees for small LOCA, type2, and type3 initiators to see if the change in assumed success criteria makes any difference to CDF. The saturation temperatures corresponding to 50 psig is approximately 300°F. If it can be shown that SG cool-down limits the exiting RCS coolant temperature to less than these values prior to reaching 50 psig, then containment integrity should be assured.	

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
1-1 / IFSO-B2 *Internal Flooding*	As noted in SRs IFSO-A1, IFSO-A3, and IFSO-A5, some areas of the documentation do not provide sufficient detail about the process used. Specific items for which improved documentation is needed include: a. Documentation of sources in the Turbine Building. b. The basis for screening sources in the Fuel, Radwaste, and Turbine Buildings (i.e., the way in which the specified criteria are met for each source is not documented). For example, a walkdown during the peer review revealed that there is section of the wet pipe fire protection (FP) system running above the turbine cooling water (TC) pumps that could potentially spray both pumps. It is not clear based on 13-NS-C093 and 13-NS-C094 that this impact was considered and dispositioned. Likewise, feedline breaks in the turbine building are assumed to be bounded by the loss of main feedwater initiating event, but may have different impacts such as loss of instrument air due to humidity impacts. c. The temperature and pressure of flood sources.	Status: Partially Closed. Basis: The documentation (Sections 4.2.5 and 4.2.6 of Study 13- NS-C094, Revision 1, and footnote in Table C.1 of 13-NS-C093 Revision 1) have been revised to address the Findings. Section 4.2.6 of Study 13-NS-C094, Revision 1 discusses the flood sources in the TB and the impact of these flood sources, if any, on equipment in TB that are modeled in the PRA. The rationale for not including the temperature and pressure of fluid systems based on Assumption 2 of PRA Study 13-NS-C096 Revision 2 must be supported by the fact that there will be no propagation of steam (due to HELB) from the location of the piping system break to the adjacent location(s) and impacting PRA equipment in the adjacent location(s). Also, for feedwater line break in the TB, it must be verified that this event will not impact other PRA equipment such as the instrument air system due to steam and humidity. Recommendation: Verify and document the fact that propagation of steam (due to a HELB) will not occur from the location of piping system break to the adjacent location(s) and impacting PRA equipment in the adjacent location(s) and that a feedwater line break in the TB will not impact other PRA equipment such as the instrument air system due to steam and humidity.	The Closure Review Team recommendation will be implemented as written. These changes are not expected to have a significant impact on CDF or LERF since the current internal flood contribution to total CDF and LERF is less than 0.7% and 0.2%, respectively. Preliminary review indicates that steam propagation will have minimal impact on PRA equipment in adjacent locations. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
1-2 / IFEV-A7 *Internal Flooding*	Potential flooding mechanisms are primarily limited to failures of components. Human-induced flooding is screened based on plant maintenance practices (see 13 NS-C093, Section 3.2, Item 4 and 13-NS-C097, Section 3.5). This does not indicate that there was any search of plant operating experience and plant maintenance procedures to verify no potential for human-induced flood mechanisms.	Status: Partially Closed. Basis: Section 4.1 (pages 24 – 25) of study 13-NS-C097 Revision 2 addressed the finding with the discussion of the potential for human and maintenance induced flooding events. Maintenance activities/procedures were reviewed and a search of plant operating experience (APS PVAR/CRDR database, plant trip history and LERs) using flood-related keywords for flooding events was performed as documented in this section (4.1) of the System Study. It is stated in Section 3.5 of study 13-NS-C097 Revision 2 that maintenance activities, which involve the replacement of pumps or cleaning of heat exchangers, have the potential to induce a significant flooding event are not performed on-line at the plant. However, there was a PVNGS event that involved the plugging of the condenser tubes during plant operation. There is also a potential for on-line heat exchanger tube plugging if a heat exchanger tube leak is detected. Such events were not considered in the discussion of maintenance/human induced flooding in Section 3.5 of Study 13-NS-C097 Revision 2. Recommendation: To be consistent with the state of the industry practice, identify and evaluate human-induced floods during plant operation for scenarios that may result from risk-significant maintenance activities (e.g., plugging of the condenser tubes and heat exchanger tubes), and include applicable maintenance-induced failure modes in the next update of the internal flooding PRA to incorporate the revised system pipe break frequency values from EPRI Report 3002000079, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessment", Revision 3.	The Closure Review Team recommendation will be implemented as written. These changes are not expected to have a significant impact on CDF or LERF since the current internal flood contribution to total CDF and LERF is less than 0.7% and 0.2%, respectively. There have been a very limited number of human induced flood events that were screened out. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
1-3 / IFSN-A6 *Internal Flooding*	RG 1.200 Revision 2 documents a qualified acceptance of this supporting requirement (SR). The NRC resolution states that to meet Capability Category II, the impacts of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement) must be qualitatively assessed using conservative assumptions.	Status: Partially Closed. Basis: Assumption 2 in Section 3.1.3 of 13-NS-C096 states that "All components within a flood area where the flood originates were assumed susceptible and failed as a result of the flood, spray, steam, jet impingement, pipe whip, humidity, condensation and temperature concerns except when component design (e.g., water-proofing), spatial effects, low pressure source potential or other reasonable judgment could be used for limiting the effect." This assumption is appropriate and effectively bounds the potential impacts from jet impingement, pipe whip, spray, submergence, etc. However, this assumption of susceptible equipment failure in the flood originating room does not always bound the impact of humidity, condensation, and temperature concerns because of the potential propagation of the flooding effects (e.g., steam). From this consideration, the assumption is nonconservative. Recommendation: For the applicable flood scenarios in relevant locations, evaluate the effects of humidity, condensation, and temperature by considering the possible propagation from the initiating room to all connecting rooms. Also see F&O 1-1.	The Closure Review Team recommendation will be implemented as written. These changes are not expected to have a significant impact on CDF or LERF since the current internal flood contribution to total CDF and LERF is less than 0.7% and 0.2%, respectively. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
1-4 / IFEV-A6 *Internal Flooding*	There is no evidence in 13-NS-C097 that a search was made for plant-specific operating experience, plant design features, and conditions that may impact flood likelihood and no Bayesian updating was performed. However, adjustments are made to some initiating event frequencies based on system run times to account for differences between impacts when the pumps are running or in standby.	Status: Partially Closed. Basis: Section 4.1 (pages 24 – 25) of study 13-NS-C097 Revision 2 addressed the finding on "lack of search for plant-specific operating experience, plant design features, and conditions that may impact flood likelihood" with a discussion of the search of flood type events in the PVNGS Site Work Management System database and License Event Reports. It also discusses the review of the PVNGS maintenance procedures on flood prevention guidelines and the potential of maintenance induced flooding. No Bayesian update was performed on the flood initiating event frequency due to insufficient flood data at PVNGS. However there were a few events associated with leaks/flooding that were screened based on the impact of those events on safety-related or PRA equipment. The criteria for screening of flooding events based on impact may not be applicable to the screening of flood events performed for the purpose of evaluating the likelihood of flooding. Recommendation: Develop criteria for screening of flood events for the purpose of evaluating the likelihood of flooding or flooding frequency. The criteria may be in terms of potential spatial impact of the flood event. Use the criteria to re-evaluate the flooding events at PVNGS for the purpose of flood frequency update. Unscreened flooding events should then be used for updating the applicable flooding frequency.	The Closure Review Team recommendation will be implemented as written. These changes are not expected to have a significant impact on CDF or LERF since the current internal flood contribution to total CDF and LERF is less than 0.7% and 0.2%, respectively. There have been a very limited number of flood events that were screened out. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
SHA-E2-01 / SHA-E2 *Seismic*	The evaluation and incorporation of uncertainties in the site response velocity profile may not be properly incorporated because of insufficient or unreviewable site-specific data and/or its documentation. Also, the site response evaluation was completed using a Senior Seismic Hazard Analysis Committee (SSHAC) Level 1 (L1) process which does not meet the ASME general Capability Category II guidelines.	Status: Open. Basis: Finding SHA-E2 is considered as Open. The technical basis for this conclusion is summarized below and is supported by the documentation as found in LCI Report 2211-PR-07-Rev. 4, Seismic Hazard Evaluation for Palo Verde Nuclear Generating Station, dated 8/27/2013 (referred to as LCI, 2013), and LCI Report PC No. PV-001-PC-05, Rev. 0, Soil Hazard and GMRS/FIRS Calculation for Palo Verde Nuclear Generating Station, dated 2/27/2015 (referred to as LCI, 2015b). LCI (2013) provides a set of soil hazard curves fractiles and for peak ground acceleration (pga) a set of 96 aggregate pga hazard curves. It is assumed that the aggregate pga hazard curves were derived for use as part of the seismic risk quantification. LCI (2013) provides an adequate summary of how the aggregate pga hazard curves were developed and compared to the pga hazard curve fractiles. LCI (2015b) provides an updated set of soil hazard curves in part based on the updated site response evaluation completed at the Palo Verde site. LCI (2015b) provides sufficient technical material to understand how the site response results were combined with the soil hazard curves for rock site conditions to produce an updated set of soil hazard curves for the Palo Verde site. Finding SHA-E2 is related to the assessment of aleatory and epistemic uncertainties in site response, and their subsequent impact on the derivation of soil hazard curves. The disposition to this finding states, "A SSHAC L3 analysis was performed subsequent to the seismic PRA development as part of the NTTF response to the NRC 50.54f letter on Fukushima. The SSHAC L3 analysis produced a site hazard curve which is bounded by the SSHAC L1 hazard curve developed and used in the Seismic PRA model. Therefore, the issue is resolved by the updated SSHAC L3 hazard analysis". While the updated probabilistic seismic hazard analysis completed for the Palo Verde site included SSHAC L3 seismic source characterization and ground motion characterization models, the site response analysis was not completed using an explicit SSHAC process. It is important to note that there is no mandate to complete the site response analysis following the SSHAC guidance; the approach and method documented in LCI (2015a and 2015b) is consistent with expectation documented as part of EPRI guidance (SPID) which has been endorsed by the NRC.	The updated seismic hazard curves provided by LCI (2015b) and associated impacts on fragilities will be incorporated into the seismic PRA. These changes are not expected to have a significant impact on CDF or LERF based on the NRC review of the updated seismic hazard curves LCI (2015b) which found the updated seismic hazard is bounded by the current design basis safe shutdown earthquake at most frequencies above 1 Hertz and minor exceedances above 1 Hertz to be considered "de minimis" (ADAMS Accession No. ML16221A604). These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3 **Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process**			
Finding F&O ID / Requirement	**Description**	**Closure Review Team Recommendation(s)**	**Disposition**
SHA-E2-01 / SHA-E2 *Seismic* (cont.)		However, the seismic hazard curves provided by LCI (2015b) were not explicitly compared to those derived in LCI (2013) to corroborate the disposition provided. LCI (2015b) does not include updated hazard curve fractiles for pga that can be compared to either the pga fractiles or the aggregate pga hazard curves as found in LCI (2013). Comparison of the mean pga hazard curves [Table 7.1 from LCI (2013) to Table 2 from LCI (2015b)] indicates that the updated pga hazard has increased, thus it is not clear if the seismic risk quantification using the pga hazard curves from LCI (2013) is appropriate. For example at a pga = 0.5g the mean hazard has increased from 1.72E-6 [Table 7.1 from LCI (2013)] to 6.53E-6 Table 2 from LCI (2015b)]. In summary, insufficient information has been provided to demonstrate that the updated pga soil seismic hazard curves is bounded by the pga soil hazard curve used in the Seismic PRA model. <u>Recommendation</u>: Demonstrate that the updated set of soil pga hazard curves fractiles (mean, and 5th, 16th, 50th, 84th, 95th) is bounded by the soil pga hazard curves used in the Seismic PGA model. If the updated set of soil pga hazard curves is greater than those used in the Seismic PRA model, the impact on Seismic risk quantification should be assessed.	

Attachment 3
Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process

Finding F&O ID / Requirement	Description	Closure Review Team Recommendation(s)	Disposition
SFR-F3-01 / SFR-F3 *Seismic*	The draft report LTR-RAM-II-12-074 indicates that the draft relay assessment uses the IPEEE relay assessment as the starting point but accounts for the updated seismic hazard curve at the site. However, the report includes the following statement in Section 2.3 (Unaddressed Relays): *This list (unaddressed relays) included 69 such relays. Of the relays that have been included in the SPRA, their seismic fragility events are found in many of the dominant CDF cutsets.*	Status: Partially Closed. Basis: In response to F&O SFR-F3-01, two documents were provided: 1) Westinghouse Letter LTR-RAM-II-12-074 R002 - PV SPRA - Relay Assessment.pdf, and 2) 11c4043-cal-028 rev0 Seis Frag for Sel Relays.pdf. These two documents were reviewed to support the conclusion that the previously unaddressed relays have been addressed and are included in the SPRA model and quantification. The following paragraphs provide the basis for the conclusion that the reported fragilities are reasonable, and provide recommendations for completeness of documentation. Westinghouse Document LTR-RAM-II-12-074 R002 – PV SPRA incorporates into the existing relay database 288 additional relays including the 69 previously unaddressed in the IPEEE. This document develops the first round relay fragilities based on the relay analysis performed as part of the plant IPEEE. Several relays, including some of the 69 relays that were not addressed in IPEEE, are screened because they are not chatter sensitive. The relays for which chatter is not acceptable are assigned initial HCLPF values based on the IPEEE review level earthquake (RLE) PGA of 0.5 scaled in accordance with the location specific spectral acceleration ratios of the IPEEE ISRS and the SPRA ISRS. These initial HCLPF values are substantiated with walkdowns of host components. The reported initial quantification using these HCLPF and generic betas identified the dominant relay contributors. Westinghouse Document LTR-RAM-II-12-074 R002 - PV SPRA describes the modelling of the relays in the SPRA. S&A Calculation 11c4043-cal-028 documents the detailed fragility analysis using separation of variable for 13 relays selected from the top contributors. The median capacity is based on the in-cabinet seismic demand on the relays associated with the SPRA ISRS, and EPRI relay GERS. The associated uncertainty parameters are obtained by using the separation of variables method. The resulting fragility parameters are scaled to conform to the revised seismic hazard and the revised GMRS. We concur with the analysis and feel that the resulting fragilities are sufficiently realistic. Finding SFR-F3-01 could be resolved on the basis that APS will take actions to implement the recommendations provided below and update WEC Document LTRRAM-II-12-074 Rev. 2, and S&A Calculation 11c4043-cal-028 Rev. 0.	The Closure Review Team recommendation will be implemented as written. These changes are not expected to have any impact on CDF or LERF since the recommendations are associated with documentation changes to better explain modeling rationale. These changes will be implemented and the finding verified closed by a subsequent F&O Closure Review as a pre-requisite to categorization.

Attachment 3 **Disposition and Resolution of Open Peer Review Findings and Self-Assessment Open Items from Facts and Observation Closure Review Process**			
Finding F&O ID / Requirement	**Description**	**Closure Review Team Recommendation(s)**	**Disposition**
SFR-F3-01 / SFR-F3 *Seismic* (cont.)		Recommendation: The detailed fragility evaluation for dominant relay contributors should demonstrate that the seismic demand is an appropriate median based response, and that the important uncertainties are included in obtaining log standard deviations. We recommend that the following be incorporated into the Westinghouse Document LTR-RAM-II-12-074 R002 for completeness: 1. During the closure review, the following information was communicated to the reviewers regarding the unaddressed relays via an e-mail: "The initial fragilities for relays were estimated based on the IPEEE analysis for 0.5g RLE. The 69 relays that were not addressed in Rev. 1 of the Relay Fragility Assessment were subsequently incorporated into the Relay Fragility Assessment Rev. 2. Twenty-nine of the 69 relays were screened out from modeling. However, 40 relays were modeled. They were assigned simplified fragilities based on walkdowns that were already performed on the parent cabinets." The above information in the e-mail communication should be documented in WEC Document LTR-RAM-II-12-074 Rev. 2." 2. In Table 3-4: Detailed Fragility Candidates, document the source of the factor SFbldg. We also recommend that the following be incorporated into the S&A Calculation 11c4043-cal-028 Rev. 0 for completeness of documentation: 1. Justify the use of Best Estimate ISRS as the median. In our opinion, the SSI analysis using BE soil properties, best estimate structure stiffness and a conservative estimate of best estimate structure damping results in a 84th percentile response. 2. The βu associated with SSI is obtained using the BE, UB and LB envelop as the 84th and the BE alone as the median. Please explain the rationale that this results for the same building (Control Building) a wide range of SSI βc from 0.09 to 0.22. 3. Explain why the uncertainties associated with structure stiffness and damping, time history simulation and earthquake component combination are ignored in the SOV calculations.	

Attachment 4 External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note 1)**	**Comment**
Aircraft Impact	Y	PS2 PS4	Airport hazard meets 1975 Standard Review Plan (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 Essential Cooling Water System (ECWS) and Component Cooling Water System (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.
Hail	Y	C1 C4	Limited occurrence and bounded by other events for which the plant is designed. Flooding impacts covered under Intense Precipitation.

Attachment 4 External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note 1)**	**Comment**
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 PVNGS 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 PVNGS Internal Fire PRA.
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.

Attachment 4 External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note 1)**	**Comment**
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.
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 PVNGS Seismic PRA.
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.

Attachment 4 External Hazards Screening			
External Hazard	**Screening Result**		
	Screened? (Y/N)	**Screening Criterion (Note 1)**	**Comment**
Storm Surge	Y	C3	Not applicable to the site because of location.
Toxic Gas	Y	C4	Toxic gas covered under release of chemicals in onsite storage, industrial or military facility accident, and transportation accident.
Transportation Accident	Y	PS2 PS4 C3 C4	Potential accidents meet the 1975 SRP requirements. Bounding analyses used for offsite rail shipment of chlorine gas and onsite truck shipment of ammonium hydroxide. Marine accident not applicable to the site because of location. Aviation and pipeline accidents covered under those specific categories.
Tsunami	Y	C3	Not applicable to the site because of location.
Turbine-Generated Missiles	Y	PS2	Potential accidents meet the 1975 SRP requirements.
Volcanic Activity	Y	C3	Not applicable to the site because of 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 1 – See Attachment 5 for descriptions of the screening criteria.			

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