Pacific Gas and Electric Company®

Dennis B. Petersen
Station Director

Diablo Canyon Power Plant
Mail code 104/5/502
P.O. Box 56
Avila Beach, CA 93424

805.545.4022
Dennis.Petersen@pge.com

PG&E Letter DCL-23-077 10 CFR 50.90

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

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
License Amendment Request 23-02
Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

Dear Commissioners and Staff:

Pursuant to Title 10 Code of Federal Regulations (10 CFR) 50.90, Pacific Gas and Electric Company (PG&E) hereby requests approval of the enclosed License Amendment Request (LAR) for a proposed amendment to modify the Diablo Canyon Power Plant (DCPP) Units 1 and 2 licensing bases, 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 either 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 contains the basis for the proposed change to the DCPP Units 1 and 2 Operating Licenses. The categorization process being implemented through this change is consistent with Nuclear Energy Institute (NEI) Report 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 system will only occur after these prerequisites are met.

The changes requested in this LAR will prevent unnecessary unit shutdowns for low safety significance equipment, and is consistent with safely maintaining DCPP generation and thereby supporting electrical grid reliability in California.

Approval of the proposed amendment is requested by September 30, 2024. Once approved, the amendment shall be implemented within 365 days.

PG&E makes no regulatory commitments (as defined by NEI 99-04) in this letter. This letter makes no revisions to existing regulatory commitments.

In accordance with site administrative procedures and the DCPP Quality Assurance Program, the proposed amendment has been reviewed by the Plant Staff Review Committee.

Pursuant to 10 CFR 50.91, PG&E is notifying the State of California of this LAR by transmitting a copy of this letter and enclosure to the California Department of Public Health.

If you have any questions or require additional information, please contact James Morris, Manager, Nuclear Regulatory Services, at 805-545-4609.

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

Sincerely,

Dennis B. Petersen

Dennis B. Petersen
Station Director

Executed on: 9/27/2023
Date

kjse/51195555
Enclosure
cc: Diablo Distribution
cc/enc: Anthony Chu, Branch Chief, California Dept of Public Health
Mahdi O. Hayes, NRC Senior Resident Inspector
Samson S. Lee, NRR Project Manager
John D. Monninger, NRC Region IV Deputy Administrator

Evaluation of the Proposed Change

Subject: License Amendment Request 23-02, Application to Adopt 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

ATTACHMENTS:

EVALUATION

1. SUMMARY DESCRIPTION

The proposed amendment modifies the Diablo Canyon Power Plant (DCPP) Units 1 and 2 licensing bases, by the addition of a License Condition, 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) using a risk-informed systematic approach. 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 either will not be changed or will be enhanced. This allows improved focus on equipment that has HSS resulting in improved plant safety.

2. DETAILED DESCRIPTION

2.1 Current Regulatory Requirements

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

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

2.2 Reason for the Proposed Change

A probabilistic approach to regulation enhances and extends the traditional deterministic approach by allowing consideration of a broader set of potential challenges to safety, providing a logical means for prioritizing these challenges based on safety significance, and allowing consideration of a broader set of resources to defend against these challenges. In contrast to the deterministic approach, probabilistic risk assessments (PRAs) address credible initiating events by assessing the event frequency. Mitigating system reliability is then assessed, including the potential for common cause failures. The probabilistic approach to regulation is an extension and enhancement of traditional regulation by considering risk in a comprehensive manner, consistent with changes in the NRC regulatory framework to a more risk-informed approach.

To take advantage of the safety enhancements available using PRA, in 2004 the NRC published a new regulation, 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.". The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of LSS, alternative treatment requirements can be implemented in accordance with the regulation. For equipment determined to be of HSS, requirements either will not be changed or will be enhanced. This allows improved focus on equipment that has HSS 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) Report NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), which uses both risk insights and traditional engineering insights. The safety functions include the design basis functions, as well as functions credited for severe accidents (including external events). Special or alternative treatment for the SSCs is applied as necessary to maintain functionality and reliability, and is a function of the SSC categorization results and associated categorization bases. Finally, periodic assessment activities are conducted to adjust the categorization and/or treatment processes as needed so that SSCs continue to meet all applicable requirements.

The rule does not allow for the elimination of SSC functional requirements or allow equipment that is required by the deterministic design basis to be removed from the facility. Instead, the rule enables licensees to focus their resources on SSCs that make a significant contribution to plant safety while continuing to provide the same overall level of safety and reliability. 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 for treatment that provides reasonable, though reduced, confidence that these SSCs will satisfy functional requirements.

Implementation of 10 CFR 50.69 will allow Pacific Gas and Electric Company (PG&E) to improve focus on DCPP equipment that has HSS resulting in improved plant safety.

2.3 <u>Description of the Proposed Change</u>

PG&E proposes the addition of the following condition to the operating licenses of DCPP Units 1 and 2 to document the NRC's approval of the use 10 CFR 50.69:

> The Pacific Gas and Electric Company is approved to implement 10 CFR 50.69 using the processes for categorization of Risk-Informed Safety Class (RISC)-1, RISC-2, RISC-3, and RISC-4 structures, systems, and components (SSCs) using: Probabilistic Risk Assessment (PRA) models to evaluate risk associated with internal events, including internal flooding, internal fire, and seismic hazards; the shutdown safety assessment process to assess shutdown risk; the Arkansas Nuclear One, Unit 2 (ANO-2) passive categorization method to assess passive component risk for Class 2 and Class 3 and non-Class SSCs and their associated supports; the results of the non-PRA evaluations that are based on the Individual Plant Examination of External Events (IPEEE) Screening Assessment for External Hazards updated using the external hazard screening significance process identified in ASME/ANS PRA Standard RA-Sa-2009 for other external hazards.
>
> Prior NRC approval, under 10 CFR 50.90, is required for a change to the categorization process specified above.

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 Sections 3.1 through 3.4 below.

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

3.1.1 Overall Categorization Process

PG&E will implement a risk categorization process at DCPP in accordance with NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline" (Reference 1), as endorsed by Regulatory Guide (RG) 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance" Revision 1, May 2006 (Reference 3). NEI 00-04 Section 1.5 states "Due to the varying levels of uncertainty and degrees of conservatism in the spectrum of risk contributors, the risk significance of SSCs is assessed separately from each of five risk perspectives and used to identify SSCs that are potentially safety significant." A separate evaluation is appropriate to avoid reliance on a combined result that may mask the results of individual risk contributors.

The process to categorize each system will be consistent with the guidance in NEI 00-04, “10 CFR 50.69 SSC Categorization Guideline,” as endorsed by RG 1.201. RG 1.201 states that "the implementation of all processes described in NEI 00-04 (i.e., Sections 2 through 12) is integral to providing reasonable confidence" and that "all aspects of NEI 00-04 must be followed to achieve reasonable confidence in the evaluations required by §50.69(c)(1)(iv)." However, neither RG 1.201 nor NEI 00-04 prescribe a particular sequence or order for each of the elements to be completed. Therefore, consistent with NEI 00-04, the order in which each of the elements of the categorization process (in Table 3-1) are completed is flexible provided they are all completed or performed in parallel as determined by the evaluator.

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

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

Table 3-1. Categorization Evaluation Summary

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

1 The assessments of the qualitative considerations are agreed upon by the IDP in accordance with NEI 00-04, Section 9.2. In some cases, a 10 CFR 50.69 categorization team may provide preliminary assessments of the seven considerations for the IDP's consideration, however the final assessments of the seven considerations are the direct responsibility of the IDP.

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

The System Categorization Document, including the justifications provided for the qualitative considerations, is reviewed by the IDP. The IDP is responsible for reviewing the preliminary assessment to the same level of detail as the 10 CFR 50.69 team (i.e., all considerations for all functions are reviewed). The IDP may confirm the preliminary function risk and associated justification or may direct that it be changed based upon their expert knowledge. Because the Qualitative Criteria are the direct responsibility of the IDP, changes may be made from preliminary HSS to LSS or from preliminary LSS to HSS at the discretion of the IDP. If the IDP determines any of the seven considerations cannot be confirmed (false response) for a function, then the final categorization of that function is HSS.

The mapping of components to system functions is used in some categorization process steps to facilitate preliminary categorization of components. Specifically, functions with mapped components that are determined to be HSS by the PRA-based assessment (i.e., Internal events PRA or Integral PRA assessment) or defense-in-depth evaluation will be initially treated as HSS. However, NEI 00-04, Section 10.2 allows detailed categorization which can result in some components mapped to HSS functions being treated as LSS; and Section 4.0 discusses additional functions that may be identified (e.g., fill and drain) to group and consider potentially LSS components that may have been initially associated with a HSS function, but which do not support the critical attributes of that HSS function. Note that certain steps of the categorization process are performed at a component level (e.g., passive, non-PRA-modeled hazards – see Table 3-1). These components from the component level assessments will remain HSS (IDP cannot override) regardless of the significance of the functions to which they are mapped. Therefore, if a HSS component is mapped to an LSS function, that component will remain HSS. If an LSS component is mapped to an HSS function, that component either may be driven to HSS based on Table 3-1, or may remain LSS.

NEI 00-04, Sections 4 and 7.1, will be followed for SSCs that support an interfacing system. Those SSCs will typically remain uncategorized until all interfacing systems are categorized. In some cases, impacts that an interfacing component could have on an interfacing system can be fully determined and the interface component can be categorized (and alternative treatment implemented) without categorizing the entire interfacing system. In this event, an assessment of interface component risk associated with uncategorized systems will be limited to cases where the following two conditions are met: 1) the interface component failure cannot prevent performance of interface system functions, and 2) the risk is limited to passive failures assessed as LSS following the passive categorization process for the applicable pressure boundary segments.

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

- The IDP will be composed of a group of at least five experts who collectively have expertise in plant operation, design engineering (mechanical and electrical), system engineering, safety analysis, and probabilistic risk assessment. At least three members of the IDP will have a minimum of five years of experience at the plant, and there will be at least one member of the IDP who has a minimum of three years of experience in the modeling and updating of the plant-specific PRA.
- The IDP will be trained in the specific technical aspects and requirements related to the categorization process. Training will address, at a minimum, (1) the purpose of the categorization; (2) the present treatment requirements for SSCs including requirements for design basis events; (3) PRA fundamentals; (4) the 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 (5) the defense-in-depth philosophy and requirements to maintain this philosophy.
- The decision criteria for the IDP to categorize SSCs in accordance with § 50.69(f)(1) will be documented in DCPP procedures.
- Decisions of the IDP will be arrived at by consensus. Differing opinions will be documented and resolved, if possible. However, a simple majority of the panel is sufficient for final decisions regarding categorization.
- Passive characterization will be performed using the processes described in Section 3.1.2 of this LAR. Consistent with NEI 00-04, an HSS determination by the passive categorization process cannot be changed by the IDP.
- An unreliability factor of 3 will be used for the sensitivity studies described in Section 8 of NEI 00-04. The factor of 3 was chosen as it is representative of the typical error factor of basic events used in the PRA model.
- NEI 00-04 Section 7 requires assigning the safety significance of functions to be preliminary HSS if it is supported by an SSC determined to be HSS from the PRA-based assessment in Section 5 but does not require this for SSCs determined to be HSS from non-PRA-based, deterministic assessments in Section 5. This requirement is further clarified in the Vogtle Units 1 and 2 Safety Evaluation (Reference 4) which states "…if any SSC is identified as HSS from either the integrated PRA component safety significance assessment (Section 5 of NEI 00-04) or the defense-in-depth assessment (Section 6), the associated system function(s) would be identified as HSS."
- Once a system function is identified as HSS, then all the components that support that function are preliminary HSS. The IDP must intervene to assign any of these HSS function components to LSS.
- Regarding the criteria that considers whether the active function is called out or relied upon in the plant Emergency/Abnormal Operating Procedures, PG&E will not take credit for alternate means unless the alternate means are proceduralized and included in Licensed Operator training.

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

- Internal Event Risks: Internal events including internal flooding PRA model.
- Fire Risks: Fire PRA model.
- Seismic Risks: Seismic PRA model.
- Other External Risks (e.g., tornados, external floods): The other external hazards were determined to be insignificant contributors to plant risk. Under the IPEEE program, a systematic reevaluation of selected external hazards was performed (Reference 5). The external hazards were reexamined in 2016 using the guidance of Part 6 of the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) PRA Standard ASME/ANS RA-Sb-2013, "Addendum B to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (Reference 6) to ensure the IPEEE conclusions remained bounding and to account for updated information. The results of the 2016 re-examination were that the external hazards (other than seismic) can be screened out; therefore, there is no need for further detailed PRA models of these hazards.
- Low Power and Shutdown Risks: Qualitative defense-in-depth (DID) shutdown model for shutdown configuration risk management (CRM) based on the framework for DID provided in NUMARC 91-06, "Guidance for Industry Actions to Assess Shutdown Management" (Reference 7), which provides guidance for assessing and enhancing safety during shutdown operations.

Any change to the risk analysis input to the categorization process that is outside the bounds specified above will not be used without prior NRC approval.

The SSC categorization process documentation will include the following elements:

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

3.1.2 Passive Categorization Process

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

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

The use of this method for a 10 CFR 50.69 application was previously approved in the final safety evaluation for the Vogtle Units 1 and 2 10 CFR 50.69 license amendment dated December 17, 2014 (Reference 4). The RI-RRA method as approved for use at Vogtle for 10 CFR 50.69 does not have any plant specific aspects and is therefore 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 since this approach will not allow the categorization of SSCs to be affected by any changes in frequency due to changes in treatment. The passive categorization process is intended to apply the same risk-informed process authorized by the NRC for the passive categorization of class 2, 3, and non-class components at ANO2 (Reference 8). This is the same passive SSC scope the NRC has conditionally endorsed in ASME Code Cases N-660 and N-662 as published in RG 1.147, Revision 15 (Reference 9). Both code cases employ a similar risk-informed safety classification of SSCs in order to change the repair/ replacement requirements of the affected LSS components. All ASME Code Class 1 SSCs with a pressure retaining function, as well as supports, will be assigned HSS for passive categorization which will result in HSS for its risk-informed safety classification and cannot be changed by the IDP. Therefore, this methodology and scope for passive categorization is acceptable and appropriate for use at DCPP for 10 CFR 50.69 SSC categorization.

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

The following sections demonstrate that the quality and level of detail of the processes to be used in categorization of SSCs are adequate. The PRA models described below have been peer reviewed, all finding level facts and observations (F&Os) from the peer reviews have been addressed and closed by a formal closure process, and there are no PRA upgrades that have not been peer reviewed.

3.2.1 Internal Events and Internal Flooding

The DCPP categorization process for internal events including internal flooding events will use a peer reviewed plant-specific PRA model. The PG&E risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for DCPP. Attachment 2 of this enclosure provides a description of PRA models to be used in the categorization with the PRA model results for the modeled hazards.

3.2.2 Internal Fire Events

The DCPP categorization process for internal fire events will use a peer reviewed plant-specific PRA model. The internal fire PRA model has been developed consistent with NUREG/CR-6850, "Fire PRA Methodology for Nuclear Power Facilities," (Reference 10) and only uses methods previously accepted by the NRC. The PG&E risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for DCPP.

3.2.3 Seismic Events

The DCPP categorization process for seismic events will use a peer reviewed plant-specific PRA model. The PG&E risk management process ensures that the PRA model used in this application reflects the as-built and as-operated plant for DCPP.

3.2.4 Other External Hazards

Under the IPEEE program, a systematic reevaluation of selected external hazards was performed (Reference 5). The external hazards were reexamined in 2016 using the guidance of Part 6 of PRA Standard ASME/ANS RA-Sb-2013 (Reference 6) to ensure the IPEEE conclusions remained bounding and to account for updated information. The results of the 2016 re-examination concluded that the external hazards (other than seismic) can be screened out; therefore, there is no need for further detailed PRA of these external hazards.

Attachment 4 provides a summary of the progressive screening approach applied for external hazards, and a summary of the external hazards screening results.

3.2.5 Low Power and Shutdown

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

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

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

3.2.6 PRA Maintenance and Updates

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

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

3.2.7 PRA Uncertainty Evaluations

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

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

In the overall risk sensitivity studies, PG&E will use a factor of three to increase the unavailability or unreliability of LSS components consistent with that approved for Vogtle Units 1 and 2 (Reference 4). Consistent with the NEI 00-04 guidance, PG&E will perform both an initial sensitivity study and a cumulative sensitivity study. The initial sensitivity study applies to the system that is being categorized. In the cumulative sensitivity study, the failure probabilities (unreliability and unavailability, as appropriate) of all LSS components modeled in all identified PRA models for all systems that have been categorized are increased by a factor of 3. This sensitivity study together with the periodic review process assures that the potential cumulative risk increase from the categorization is maintained acceptably low. The performance monitoring process monitors the component performance to ensure that potential increases in failure rates of categorized components are detected and addressed before reaching the rate assumed in the sensitivity study.

The process of identifying, characterizing, and qualitative screening of model uncertainties is in NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 11), and Section 3.1.1 of EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments" (Reference 12). 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.

Each PRA element notebook was reviewed for assumptions and sources of uncertainties. The characterization of assumptions and sources of uncertainties are based on whether the assumption and/or source of uncertainty is key to the 10 CFR 50.69 application in accordance with RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2 (Reference 13).

Key DCPP PRA model specific assumptions and sources of uncertainty for this application were identified and dispositioned in Attachment 5.

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

The PRA models described in Section 3.2 have been peer reviewed using the endorsed PRA standard of RG 1.200, Revision 2. All finding level F&Os resulting from the peer reviews have been fully addressed by PRA model update or documentation update. Finding level F&Os were independently reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations," (Reference 14) as accepted by NRC in the letter dated May 3, 2017 (Reference 15). The results of these reviews for each PRA model are summarized in Attachment 3; cited references to internal reports are available for NRC audit.

As summarized in Attachment 3 for the internal events including internal flooding PRA model, fire PRA model, and seismic PRA model, each model has been subject to a peer review process against a standard or set of acceptance criteria endorsed by the NRC as required by 10 CFR 50.69(c)(1)(i). The finding level F&Os have been closed by an F&O closure review. There are no remaining open finding level F&Os for any PRA model. Therefore, the DCPP PRA models are of sufficient technical acceptability and level of detail to support the categorization process.

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

The DCPP 10 CFR 50.69 categorization process will implement the guidance in NEI 00-04. The overall risk evaluation process described in the NEI 00-04 guidance addresses both known degradation mechanisms and common cause interactions and meets the requirements of §50.69(b)(2)(iv).

Sensitivity studies described in NEI 00-04 Section 8 will be used to confirm that the categorization process results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF). The failure rates for equipment and initiating event frequencies used in the PRA include the quantifiable impacts from known degradation mechanisms, as well as other mechanisms (e.g., design errors, manufacturing deficiencies, and human errors). Subsequent performance monitoring and PRA updates required by the rule will continue to capture this data and provide timely insights into the need to account for any important new degradation mechanisms.

3.5 Feedback and Adjustment Process

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

Scheduled periodic reviews, at a frequency of at least once every two refueling cycles, will evaluate new insights resulting from available risk information changes (i.e., PRA model or other analysis used in the categorization), design changes, operational changes, and SSC performance. If it is determined that these changes have affected the risk information or other elements of the categorization process such that the categorization results are more than minimally affected, then the risk information and the categorization process will be updated. This review will include:

- A review of plant modifications since the last review that could impact the SSC categorization,
- A review of plant specific operating experience that could impact the SSC categorization,
- A review of the impact of the updated risk information on the categorization process results,
- A review of the importance measures used for screening in the categorization process, and
- An update of the risk sensitivity study performed for the categorization.

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

4. REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

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

- 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors"
- 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 3).
- Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (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 13).

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

4.2 No Significant Hazards Consideration Analysis

Pacific Gas and Electric Company (PG&E) proposes to modify the DCPP Power Plant Units 1 and 2 licensing basis to allow for the voluntary implementation of the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Part 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors.” The provisions of 10 CFR 50.69 allow adjustment of the scope of equipment subject to special treatment controls (e.g., quality assurance, testing, inspection, condition monitoring, assessment, and evaluation). For equipment determined to be of low safety significance, alternative treatment requirements can be implemented in accordance with this regulation. For equipment determined to be of high safety significance, requirements either will not be changed or will be enhanced. This allows improved focus on equipment that has high safety significance resulting in improved plant safety.

PG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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 Systems, Structures, and Components (SSCs) subject to Nuclear Regulatory Commission (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 evaluation, PG&E concludes that the proposed change does not involve a 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

PG&E has evaluated the proposed amendment and 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. Nuclear Energy Institute (NEI) Report NEI 00-04, Revision 0, "10 CFR 50.69 SSC Categorization Guideline," July 2005. (ADAMS Accession No. ML052910035)
2. Not Used.

3. Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance," Revision 1, May 2006. (ADAMS Accession No. ML061090627)
4. Letter from R. Martin (NRC) to C.R. Pierce (Southern Nuclear Operating Company, Inc.), "Vogtle Electric Generating Plant, Units 1 and 2 - Issuance of Amendments Re: Use of 10 CFR 50.69 (TAC NOS. ME9472 AND ME9473)," December 17, 2014. (ADAMS Accession No. ML14237A034)
5. PG&E, “Individual Plant Examination of External Events Report For Diablo Canyon Power Plant Units 1 and 2 in Response to Generic Letter 88-20 Supplement 4,” June 1994.
6. ASME/ANS RA-Sb-2013, “Addendum B to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, September 30, 2013.
7. NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," December 1991. (ADAMS Accession No. ML14365A203)
8. Letter from M.T. Markley (NRC) to VP-Operations (Arkansas Nuclear One), "Arkansas Nuclear One, Unit 2 - Approval of Request for Alternative AN02-R&R-004, Rev. 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. MD52)," April 22, 2009. (ADAMS Accession No. ML090930246)
9. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 15, October 2007. (ADAMS Accession No. ML072070419)
10. NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities Volumes 1 and 2," September 30, 2005. (ADAMS Accession Nos. ML15167A401 and ML15167A411)
11. NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1, March 2017. (ADAMS Accession No. ML17062A466)
12. EPRI TR-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," December 2008.
13. Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, March 2009. (ADAMS Accession No. ML090410014)
14. NEI Letter to NRC, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, 'Close-Out of Facts and Observations (F&Os),'" February 21, 2017. [Note, title contained typographical error, NEI 12-16 should be NEI 12-13] (ADAMS Accession Number No. ML17086A431)

15. Letter from M. J. Ross-Lee (NRC) to G. Krueger (NEI), "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 7-12, and 12-13, Close Out of Facts and Observations (F&Os)," May 3, 2017. (ADAMS Accession No. ML17079A427)
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 3, January 2018. (ADAMS Accession No. ML17317A256)
17. ASME/ANS RA-Sa-2009, “Standard for Level 1/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, February 2009.
18. NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," August 2019. (ADAMS Accession No. ML19241A615)
19. Westinghouse Letter LTR-RAM-II-13-002, “RG 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements for the Diablo Canyon Nuclear Plant Probabilistic Risk Assessment,” March 20, 2013.
20. PWROG Report PWROG-23015-P, “Diablo Canyon F&O Closure and Focused Scope Peer Review Report,” Revision 0,” July 2023.
21. Westinghouse Letter LTR-RAM-II-08-019, “Pilot Application of the Fire PRA Peer Review Process for the Diablo Canyon Power Plant Fire Probabilistic Risk Assessment,” October 17, 2008.
22. American Nuclear Society (ANS) Standard ANSI/ANS-58.23-2007, “FPRA Methodology”.
23. Westinghouse Letter LTR-RAM-II-11-004, “Fire PRA Peer Review Against the Fire PRA Standard SRs From Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for the Diablo Canyon Plant Fire Probabilistic Risk Assessment,” May 24, 2011.
24. Engineering Planning and Management, Inc., P3118-004-001, “F&O Closeout by Independent Assessment Report for the Diablo Canyon Nuclear Power Plant (DCPP) Fire PRA Model Against the 2009 ASME PRA Standard Requirements and NEI 05-04 Appendix X,” Revision 0, September 2018.
25. PWROG Report PWROG-17022-P, “Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment,” Revision 0, September 2017.
26. PWROG Report PWROG-17078-P, “Independent Assessment of Facts & Observations Closure and Focused Scope Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment,” Revision 0, March 2018.
27. 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, May 2011. (ADAMS Accession No. ML100910006)

28. NUREG/CR-2300, "PRA Procedures Guide, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Volume 2, January 1983. (ADAMS Accession No. ML063560440)

29. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," April 1991. (ADAMS Accession No. ML063550238)

30. NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," December 1987. (ADAMS Accession Nos. ML111950285, ML14196A083)

31. NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” December 1990. (ADAMS Accession Nos. ML040140729, 120960691,16281A233, 16284A005)

List of Categorization Prerequisites

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

- Integrated Decision-Making Panel (IDP) member qualification requirements.
- Qualitative assessment of system functions. System functions are qualitatively categorized as preliminary High Safety Significant (HSS) or Low Safety Significant (LSS) based on the seven considerations in Section 9.2 of NEI 00-04 (see Section 3.1). Any component supporting an HSS function is categorized as preliminary HSS. Components supporting an LSS function are categorized as preliminary LSS.
- Component safety significance assessment. Safety significance of active components is assessed through a combination of Probabilistic Risk Assessment (PRA) and non-PRA methods, covering all hazards. Safety significance of passive components is assessed using a methodology for passive components.
- Assessment of defense-in-depth (DID) and safety margin. Safety-related components that are categorized· as preliminary LSS are evaluated for their role in providing DID and safety margin and, if appropriate, upgraded to HSS.
- Review by the IDP. The categorization results are presented to the IDP for review and approval. The IDP reviews the categorization results and makes the final determination on the safety significance of system functions and components.
- Risk sensitivity study. For PRA-modeled components, an overall risk sensitivity study is used to confirm that the population of preliminary LSS components results in acceptably small increases to core damage frequency (CDF) and large early release frequency (LERF) and meets the acceptance guidelines of Regulatory Guide 1.174.
- Periodic reviews are performed to ensure continued categorization validity and acceptable performance for those SSCs that have been categorized.
- Documentation requirements per subsection 3.1.1 of the enclosure.

Description of PRA Models to be Used in Categorization
(Diablo Canyon Baseline PRA Model Results for Modeled Hazards)

Hazard	Unit 1 CDF (per rx-yr)	Unit 1 LERF (per rx-yr)	Unit 2 CDF (per rx-yr)	Unit 2 LERF (per rx-yr)
Internal Events	4.78E-6	9.89E-7	4.78E-6	9.89E-7
Internal Flooding	7.61E-6	2.45E-7	6.48E-6	2.09E-7
Seismic	2.96E-5	5.22E-6	2.96E-5	5.22E-6
Fire	4.60E-5	1.42E-6	3.99E-5	1.31E-6
Total	**8.80E-5**	**7.88E-6**	**8.08E-5**	**7.73E-6**

NOTE: The ASME/ANS PRA Standard, states the units "per reactor year" (“per rx-yr” in this table) and "per calendar year" are equivalent.

Technical Acceptability of the Diablo Canyon Power Plant PRA Models

The Diablo Canyon Power Plant (DCPP) Units 1 and 2 Probabilistic Risk Assessment (PRA) models are at-power models consisting of four hazard models: internal events, internal flooding, internal fire, and seismic events. Each hazard model applies the internal events model as the base model. The models can evaluate both the core damage frequency (CDF) and large early release frequency (LERF).

Peer Review and Peer Review Findings Closure Process

All of the PRA models discussed in this Attachment have been peer reviewed and assessed using PRA Standard ASME/ANS RA-Sa-2009 (Reference 17) and RG 1.200, Revision 2 (Reference 13). Each peer review identified facts and observations (F&Os) for supporting requirements of the relevant parts of the PRA standard applicable to the scope of the peer review. These included: findings for elements which did not meet at least capability category II of a supporting requirement of the standard, suggestions from the peer review team for elements which met the supporting requirement but could be improved, and best practices.

The review and closure of finding-level F&Os was performed by independent assessment teams using the process documented in Appendix X to NEI 05-04/07-12/12-13, "Close-out of Facts and Observations (F&Os)," (Reference 14). All of the reviews met the requirements of NEI 17-07, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2 (Reference 18).

Each assessment team (internal events including internal flooding, internal fire, and seismic) evaluated whether each F&O was closed through the application of a PRA maintenance or upgrade activity, as defined by the PRA standard. If closure of an F&O was identified as an upgrade, a focused scope peer review was conducted. Further, the assessment team re-evaluated any supporting requirements identified by the peer review to be either not met, or met at capability category I, to determine if closure of the associated F&O(s) resulted in a change in status to either met, or met at least at capability category II.

The PRA scope and technical adequacy is met for this application as the applicable PRA Standard supporting requirements for all models are met at capability category II or higher. There are no remaining open finding level F&Os for any of the models discussed in this application, and all finding level F&Os have been independently assessed and closed using the processes discussed above. The resolved findings and the basis for resolution are documented in the DCPP PRA documentation and the F&O Closure Review reports. The results of the peer reviews and independent assessments have been documented and are available for NRC audit.

Internal Event and Internal Flood PRA

An internal event and internal flood PRA peer review was conducted in December 2012, and is documented in LTR-RAM-II-13-002, “RG 1.200 PRA Peer Review Against the ASME/ANS PRA Standard Requirements for the Diablo Canyon Nuclear Plant Probabilistic Risk Assessment” (Reference 19). The full-scope peer review of these models was performed consistent with RG 1.200, Revision 2, using the current endorsed PRA Standard ASME/ANS RA-Sa-2009.

All F&Os categorized by the peer review team as findings have been resolved by either a PRA model revision or a documentation update.

An independent assessment of the finding level F&Os was conducted in June 2023 and is documented in the Pressurized Water Reactor Owners Group (PWROG) report PWROG-23015-P, “Diablo Canyon F&O Closure and Focused Scope Peer Review Report,” (Reference 20). The review was conducted in accordance with Appendix X to NEI 05-04/07-12/12-13. The scope of the assessment included all finding level F&Os resulting from the peer review.

Five suggestion level F&Os in high level requirement LE (Large Early Release) were identified as upgrades because the supporting requirements were met at capability category I only. Seven internal flooding F&Os were identified by PG&E as upgrades. A focused scope peer review was therefore conducted in conjunction with the closure review for these 12 F&Os. No other F&Os were determined to constitute an upgrade, and the use of any new methods was not identified by the assessment team.

At the conclusion of the independent assessment and focused scope peer reviews, all applicable supporting requirements of the PRA standard are met, and supporting requirements which distinguish different capability categories satisfy at least capability category II. There are no remaining open peer review finding level F&Os.

Therefore, the DCPP internal events and internal flooding PRA model is acceptable for use in the 10 CFR 50.69 Program.

Fire PRA

The internal fire PRA model peer review consisted of two reviews. The first review was in January 2008 as part of the pilot application of the fire PRA peer review process of NEI 07-12, and is documented in LTR-RAM-II-08-019, “Pilot Application of the Fire PRA Peer Review Process for the Diablo Canyon Power Plant Fire Probabilistic Risk Assessment” (Reference 21). The 2008 peer review was conducted against the requirements of the ANS Standard ANSI/ANS-58.23-2007 “FPRA Methodology” (Reference 22). At the time of this first peer review, certain technical elements of the fire PRA had not been completed.

The second peer review was completed in December 2010 and is documented in LTR-RAM-II-11-004, "Fire PRA Peer Review Against the Fire PRA Standard SRs From Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications for the Diablo Canyon Plant Fire Probabilistic Risk Assessment" (Reference 23). The 2010 peer review was conducted against the requirements of ASME/ANS RA-Sa-2009. The scope of the 2010 review included re-review of elements from the 2008 review which did not meet at least capability category II of the PRA standard.

After the final peer review in 2010, there were 17 identified finding level F&Os. All F&Os categorized as findings have been resolved by either a PRA model revision or a documentation update.

An independent assessment of the F&Os was conducted in August - September 2018, and is documented in report P3118-004-001, "F&O Closeout by Independent Assessment Report for the Diablo Canyon Nuclear Power Plant (DCPP) Fire PRA Model Against the 2009 ASME PRA Standard Requirements and NEI 05-04 Appendix X" (Reference 24). The review was conducted in accordance with Appendix X to NEI 05-04/07-12/12-13. The scope of the assessment included all 17 F&Os resulting from the two previous peer reviews. Two F&Os were identified by PG&E as upgrades, and a focused-scope peer review was therefore conducted in conjunction with the closure review. No other F&Os were determined to constitute an upgrade, and the use of any new methods was not identified by the assessment team.

At the conclusion of the independent assessment and focused scope peer review, all applicable supporting requirements of the PRA standard are met, and supporting requirements which distinguish different capability categories satisfy at least capability category II. There are no remaining open peer review finding level F&Os.

Therefore, the DCPP fire PRA model is acceptable for use in the 10 CFR 50.69 Program.

It is noted that the DCPP fire PRA model was reviewed by the NRC as part of the DCPP NFPA-805 license amendment request dated June 26, 2013. The NRC review was concluded on April 14, 2016. Based on the staff's review, the NRC staff concluded that the DCPP fire PRA is of sufficient technical adequacy and that its quantitative results, considered together with the sensitivity studies, can be used to demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines in RG 1.174, Revision 2 (Reference 27). (It is noted that the current RG 1.174 Revision 3 (Reference 16) did not modify the acceptance guidelines found in RG 1.174 Revision 2.)

Seismic PRA

A seismic PRA peer review was conducted in June 2017, and is documented in the report PWROG-17022-P, "Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment" (Reference 25). The full-scope peer review, that also included a review of the seismic hazard and fragility analyses, was performed consistent with RG 1.200, Revision 2, using the current endorsed PRA Standard ASME/ANS RA-Sb-2013 (Reference 6). All F&Os categorized as findings have been resolved by either a PRA model revision or a documentation update.

An independent assessment of the finding level F&Os was conducted in October - December 2017, and is documented in report PWROG-17078-P, "Independent Assessment of Facts & Observations Closure and Focused Scope Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment," (Reference 26). The scope of the assessment included all finding level F&Os resulting from the peer review. Three F&Os were identified by PG&E as upgrades, and two additional F&Os were identified by the assessment team as upgrades; therefore, a focused scope peer review was conducted in conjunction with the closure review. The use of any new methods was not identified by the assessment team.

At the conclusion of the independent assessment and focused scope peer review, all applicable supporting requirements of the PRA standard are met, and supporting requirements which distinguish different capability categories satisfy at least capability category II. There are no remaining open peer review finding level F&Os.

Therefore, the DCPP seismic PRA model is acceptable for use in the 10 CFR 50.69 Program.

It is noted that the DCPP seismic PRA model was submitted to the NRC for review in response to a 10 CFR 50.54(f) letter regarding lessons learned from the accident at the Fukushima Daiichi nuclear power plant. The NRC review was concluded on January 22, 2019, which concluded that the seismic PRA is of sufficient technical adequacy to support phase 2 regulatory decision making in accordance with the intent of the 10 CFR 50.54(f) letter.

Additional Information on the Use of FLEX Equipment

The current DCPP PRA models do not credit risk reduction through use of FLEX equipment.

However, the PRA model does model operator actions for two FLEX strategies contained in procedures to shed vital direct current (DC) loads and to manually control the turbine driven Auxiliary Feedwater pump. The operator actions for these two FLEX strategies are included in the seismic PRA model for a seismically induced Station

Black Out (SBO) or SBO with loss of all DC power. These actions include credit for FLEX strategies to monitor steam generator level at the hot shutdown panel without instrument AC power available.

External Hazards Screening

This Attachment discusses the generic methodology used to identify and disposition other external hazards risk sources, and provides the Diablo Canyon Power Plant (DCPP) Units 1 and 2 specific results of the application of the generic methodology for impacts on the categorization process.

In order to identify a comprehensive listing of other external hazards for consideration, the supporting requirement EXT-A1 from Reference 17 was used, which includes a review of the following sources:

- NUREG/CR-2300, "PRA Procedures Guide, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Volume 2 (Reference 28)
- NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" (Reference 29)
- NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States" (Reference 30)
- NUREG-1150 “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants” (Reference 31)
- Appendix 6-A of the ASME/ANS PRA Standard (Reference 17)

In addition to the above generic sources, the DCPP Final Safety Analysis Report was reviewed to identify any site-specific hazards consistent with the ASME/ANS PRA Standard supporting requirement EXT-A2. Based on this review, no additional external hazards were identified that are not already listed in the generic references.

Technical Approach

The other external hazards are evaluated using a preliminary screening and a quantitative screening using the criteria of the ASME/ANS PRA Standard.

The ASME/ANS PRA Standard identifies external hazard screening criteria (supporting requirements EXT-B1, EXT-B2, and EXT-C2), identified below:

Supporting Requirement EXT-B1:

(1) The hazard would result in equal or lesser damage than the events for which the plant has been designed. This requires an evaluation of plant design bases to estimate the resistance of plant structures and systems to a particular external hazard.

(2) The hazard has a significantly lower mean frequency of occurrence than another event (taking into account the uncertainties in the estimates of both frequencies), and the hazard could not result in worse consequences than the other event.

(3) The hazard cannot occur close enough to the plant to affect it. Application of this criterion needs to take into account the range of magnitudes of the hazard for the recurrence frequencies of interest.

(4) The hazard is included in the definition of another event.

(5) The hazard is slow in developing, and it can be demonstrated that sufficient time exists to eliminate the source of the threat or to provide an adequate response.

Supporting Requirement EXT-B2

SRP For screening out an external hazard other than seismic events, the design basis for the hazard meets the criteria in the NRC Standard Review Plan (SRP) with justification of the screening if based solely on conformance to SRP.

The ASME/ANS PRA Standard also requires the above qualitative screening to be supported by a review of information on the plant's design hazard and licensing basis relevant to the event screened (supporting requirement EXT-B3) as well as a review of any significant changes since the issuance of the original plant operating license for selected events (supporting requirement EXT-B4).

For hazards other than internal events, internal flooding, internal fire, and seismic events, the following criteria provide an acceptable basis for a bounding analysis for a demonstrably conservative analysis per the ASME/ANS PRA Standard:

Supporting Requirement EXT-C1

A The current design-basis hazard event has mean frequency less than 10^{-5}/year, and the mean value of the conditional core damage probability (CCDP) is assessed to be less than 10^{-1}.

B The core damage frequency, calculated using a bounding or demonstrably conservative analysis, has a mean frequency of less than 10^{-6}/year.

As allowed in the ASME/ANS PRA Standard supporting requirements EXT-C2 through EXT-C6, the quantitative screening analyses use the mean frequency and other parameters of design-basis hazards. For the remaining other hazards, either realistic or conservative models (i.e., an Internal Event model that meets the system-analysis requirements in Part 2 of the ASME/ANS PRA Standard) identifying those SSCs vulnerable to the hazard and data (i.e., the hazard analysis and any fragility analysis) were used.

The following table provides the external hazards evaluated, the screening criteria, summarizes the evaluation, and provides a disposition for the 10 CFR 50.69 categorization process.

External Hazard	Screening Criteria	Evaluation	Disposition
Aircraft Impact	SRP, B	The total CDF induced by an aircraft crash at DCPP Unit 1 is $7.43x10^{-07}$ per year. Unit 2 is expected to have a similar risk from aircrafts due to the shared building structures and near identical non-shared building structures. Projected air traffic from the small airport and airways does not pose a significant safety impact to DCPP based on the design of the facility and the low frequency of core damage due to such events.	Based on the facility design and conformance to the SRP, the aircraft impact hazard is not significant.
Avalanche	3	Location of the site does not support heavy snowfall and accumulation that may cause an avalanche.	An avalanche is not a credible hazard and is therefore not significant.
Biological Event	4, 5	Excessive fouling by slime is not expected in a 24-hour period. Slime buildup occurs over a period of several weeks and is controlled by chlorination over the long term. Rapidly occurring biological plugging can occur at DCPP, such as an intermittent kelp intrusion or a concentration of salp (a gelatinous marine invertebrate) at the cooling water intake cove that can cause the plant to ramp down in power, with the possibility of a reactor trip or a loss of condenser vacuum. Both of these consequences are currently modeled in the PRA as reactor trip and loss of condenser vacuum initiators. Since the impact of the above biological events is accounted for via the associated internal initiating events, these biological events are screened from further analysis.	Biological events are bounded by the existing internal events PRA modeling; therefore, the consequences of this hazard are adequately addressed in the existing PRA model.
Coastal Erosion	4, SRP	This is a very slow process; there is a long lead time to respond by placing the units into cold shutdown. The bedrock beneath the power plant site occupies	Due to the long lead time available to respond to coastal erosion, coastal erosion is not a significant hazard.

External Hazard	Screening Criteria	Evaluation	Disposition
		the southerly flank of a major syncline that trends west to northwest. No evidence of a major fault has been recognized within or near the coastal area, and the bedrock relationships in the exploratory trenches positively indicate that no such fault is present within the area of the plant site.	
Drought	3	The ultimate heat sink is the Pacific Ocean; the plant is not adversely impacted by drought conditions.	Since the event has no adverse impact, it is not a significant hazard.
External Flooding	SRP, A, B	It is unlikely that the reservoirs can fail in such a way to pose a threat to the plant; however, a worst case scenario is still evaluated to conservatively estimate the hazard. Reservoirs 1-A and 1-B are holding reservoirs (Reservoir 1-B is behind 1-A, and is located nearly 500 feet from the edge of the hillside). Assuming that both reservoirs lose all their water and that the entire volume of water flows toward the plant. The area covered by the flood is taken to be the triangle formed by the closest point of Reservoir 1-A to the plant (800 feet), and the north and south sides of the plant (800 feet). This area is approximately 320,000 square feet. If the entire reservoir inventory is applied to this area, the depth of flooding will be approximately 2 feet at the back of the plant. The flood will only be temporary and not sustained. Results of the hydrologic and hydraulic analysis indicate that no safety related SSCs are inundated by a probable maximum flood (PMF). The 230kV switchyard (non-safety related) would be inundated during the PMF event, however, this type of event is already included in the definition of a severe weather-related loss of offsite power (LOOP). All other DCPP	External flooding scenarios do not pose a significant safety impact based on the design of the facility and conformance to the SRP. It is therefore concluded external flooding is not a significant hazard.

External Hazard	Screening Criteria	Evaluation	Disposition
		facilities and site features remain above the calculated PMF water surface levels, including the intake structure and the entire power block, which consists of the fuel handling building, the auxiliary building, the turbine building, and the two containment buildings.	
Extreme Wind or Tornado	B	The conservative strike frequency of a tornado is 7.0×10^{-5} per year. The CCDP for LOOP due to severe weather with no recovery is estimated to be 5.16×10^{-4}. The conservatively estimated CDF for a tornado event is then 3.92×10^{-8} per year. Tornado missile scenarios have been conservatively evaluated with a CDF of 2.05×10^{-7} per year (Unit 2 has a similar impact).	A conservative evaluation of an extreme wind or tornado event demonstrated an insignificant contribution to CDF. It is therefore concluded the extreme wind or tornado hazard is not significant.
Fog	4	No direct impact to CDF and LERF due to fog, however indirect impact of fog, such as impact on aircraft crash frequency, accident data include the effect of fog.	Since the event has no direct adverse impact not already addressed by another event, the fog hazard is not significant.
Forest or Range Fire	1, 4	The area immediately around the plant site boundary is not heavily wooded, and is adjacent to the Pacific Ocean. The hazard from external fires to the plant is remote and the impact of external fires on the offsite grid have been accounted for in the LOOP initiating events.	External fire impacts are bounded by the existing internal events PRA modeling; therefore, forest or range fire is not a significant hazard.
Frost	4	Frost may impact the switchyard and grid. The frequency of a LOOP initiator includes the impact of frost, and the contribution of frost is judged to be negligible.	Frost impacts are bounded by the existing internal events PRA modeling; therefore, this hazard is adequately addressed in the PRA model for categorization.
Hail	4	The impact of hail on offsite power is included in the	Hail impacts are bounded by the existing

External Hazard	Screening Criteria	Evaluation	Disposition
		frequency of LOOP analysis. The contribution to the overall risk is judged to be negligible.	internal events PRA modeling; therefore, this hazard is adequately addressed in the PRA model for categorization.
Heavy Load Drop	SRP	Maintenance activities are the cause of heavy load drops, and as such, are controlled and evaluated under the 10 CFR 50.65 (a)(4) risk assessment process on a case-by-case basis. The DCPP design basis for heavy load drops, single-failure-proof heavy load handling systems and the control of heavy load program at the plant satisfies the SRP Screening Criteria.	A heavy load drop is not judged to have any significant impact on categorization.
High Summer Temperature	4	The impact of a high temperature environment on equipment performance is included in equipment failure data.	Since there is no unique impact on plant operation not already considered in the PRA models, this hazard is adequately addressed in the PRA model for categorization.
High Tide, Lake Level, or River Stage	4	The impact is already considered for External Flooding.	Since the event has no adverse impact not already addressed by another event, this hazard is adequately addressed in the PRA model for categorization.
Hurricane	B	Conservatively assuming a hurricane with a wind speed of 150 mph leads to core damage, this yields a CDF of 5.0×10^{-7} per year, which is below the screening criterion B. Therefore, it is judged that hurricane-initiated scenarios are insignificant contributors to the overall CDF.	The frequency of a hurricane leading to core damage is well below 1×10^{-6} per year, it is therefore concluded that the hurricane hazard is not significant.
Ice Cover	4	May impact the switchyard and grid. The frequency of a LOOP initiator includes the impact of ice cover.	Ice impacts are bounded by the existing internal events PRA modeling; therefore, this hazard is adequately addressed in the

External Hazard	Screening Criteria	Evaluation	Disposition
			PRA model for categorization.
Industrial or Military Facility Accident	3	Nearby industrial and military facilities with the potential to store or use hazardous materials are all located at distances greater than five miles from the site. Chemicals stored, used, or situated at distances greater than five miles from the plant do not need to be considered because, if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that either: the toxic limits will never be reached; or, there would be sufficient time for the control room operators to take appropriate action. In addition, the probability of a plume remaining within a given sector for a long period of time is small. Due to very limited industry within San Luis Obispo County and the distances involved, any hazardous products or materials manufactured, stored, or processed in the areas beyond five miles from the site are not considered to be a significant hazard to the plant and, as such, the explosion, fire, and toxic gas hazards can be screened.	Nearby facility accidents do not pose a significant safety impact to DCPP. It is therefore concluded that industrial or military facility accidents hazard is not significant.
Intense Precipitation	4, A	The water depth above the door thresholds and areas to the west of the turbine and buttress buildings varied between 0.05 ft. and 0.68 ft., with six of the doors/areas showing no inundation. The total force due to hydrostatic and hydrodynamic loading due to a local intense precipitation event was generally small for all the doors and safety-related structures, varying from one to 35 lb./ft. for doors and areas experiencing inundation. Forces due to the associated local intense precipitation flood event	Since there are no adverse impacts from intense precipitation events, this hazard is not significant.

External Hazard	Screening Criteria	Evaluation	Disposition
		effects will not adversely impact the doors or power block and surrounding structures. The safety-related fuel oil transfer equipment is elevated six inches above grade, and therefore would not experience any flooding.	
Landslide	3	Earthquake loading, as a result of an earthquake on the Hosgri fault zone, following periods of prolonged precipitation, will not produce any significant slope failure that can impact the class I structures and equipment. In addition, potential slope failures under such conditions will not adversely impact other important facilities, including the raw water reservoirs, the 230 kV and 500 kV switchyards, and the intake and discharge structures. Potential landslides may temporarily block the normal paved south access road at several locations. However, there is considerable room adjacent to and north of the paved road to reroute emergency traffic. There is also an unpaved north access road that may be used. Therefore, landslides can be screened.	Since there are no adverse impacts from landslide events, this hazard is not significant.
Lightning	1, 4	The plant contains lightning protection in the plant design. The impact on offsite power included in the loss of offsite power frequency evaluation. The contribution to the overall risk judged to be negligible.	Since the event has no adverse impact not already addressed by another event, this hazard is adequately addressed in the PRA model for categorization.
Low Lake Level or River Stage	3	Not applicable to DCPP.	Since the event is not applicable, there is no impact on categorization.
Low Winter Temperature	1, 4	The impact on equipment has been included in the component (independent and common cause) failure rates. Thermal stresses and embrittlement are	Since there is no unique impact on plant operation not already considered in the PRA models, this hazard is adequately

External Hazard	Screening Criteria	Evaluation	Disposition
		usually insignificant and covered by design codes and standards for the plant design.	addressed in the PRA model for categorization.
Meteorite or Satellite Impact	2	The probability of the event is less than 1 x 10^{-9} per year per Reference 6.	Based on the low frequency of the event, a meteorite or satellite impact is not a significant hazard.
Pipeline Accident	3, 4	No natural gas or other pipelines pass within five miles of the plant site. The onsite hazardous buried piping at the DCPP plant site are those owned by PG&E which may carry diesel fuel oil, hydrogen, etc. However, fire and explosion resulting from fuel oil and hydrogen are evaluated separately in the DCPP Fire PRA. As such, rupture of the hydrogen line and the potential explosion that may result are not re-evaluated. The other buried pipes containing hydrocarbons are mainly the diesel fuel oil and waste oil pipes which are not in the form of toxic gas and have an extremely low likelihood of explosion.	There are no pipelines in sufficient proximity to the plant site to cause a significant hazard. Since there is no unique impact on plant operation not already considered in the PRA models, this hazard is adequately addressed in the PRA model for categorization.
Release of Chemicals in On Site Storage	1	Hazards due to explosion, toxicity, or asphyxiation were evaluated and it was concluded that they pose no hazard to the control room personnel and PRA equipment. Any toxic gas that may be generated from the accidental release of onsite chemicals would not impact the control room habitability. Therefore, release of chemicals in onsite storage can be screened.	There are no chemicals on site which can cause a significant safety hazard. Therefore, this is not a significant hazard.
River Diversion	3	Not applicable to the DCPP site.	Since the event is not applicable, there is no impact on categorization.
Sandstorm	3, 4	Sandstorms are included in the extreme winds and tornados. They are judged to be insignificant in	Since the event has no adverse impact not already addressed by another event, this

External Hazard	Screening Criteria	Evaluation	Disposition
		occurrence, frequency, and risk.	hazard is adequately addressed in the PRA model for categorization.
Seiche	1, SRP	Seiche effects on intake cove wave heights are less than 3.2 ft of the maximum crest wave level inside the breakwaters of 12.8 ft, and are therefore, not a concern. The maximum expected water volume loss from each of the raw water storage reservoirs is 14,684 gallons. The raw water storage reservoirs are able to perform their design function with only one million gallons per reservoir. As such, loss of 14,684 gallons is not significant.	A seiche has no adverse impact and therefore is not a significant hazard.
Sink Holes	1, 5	The site suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.	Sink hole impacts are not a credible event, therefore, it is not a significant hazard.
Snow	3	The location of the site is such that it does not experience heavy snowfall which could impact the switchyard and grid.	Snow impacts are not a credible event, therefore, it is not a significant hazard.
Soil Shrink-Swell Consolidation	1, 5	This event is a slow process. Contribution to the overall risk is judged to be negligible. The site suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.	A slow developing soil shrink-swell consolidation hazard is negligible.
Storm Surge	4	The maximum estimated wave height outside the breakwaters was 44.6 ft. The maximum crest wave level inside the breakwaters was 12.8 ft. While seiche effects were noted in the intake cove, the wave heights were found to be less than 3.2 ft. of the maximum estimated wave height, and are therefore, not a concern.	A storm surge has no adverse impact and therefore is not a significant hazard.

External Hazard	Screening Criteria	Evaluation	Disposition
Toxic Gas	4	See the table entries for “Release of Chemical in Onsite Storage” and “Industrial Accidents.”	See the table entries for “Release of Chemical in Onsite Storage” and “Industrial Accidents.”
Transportation Accident	3, B	Various scenarios involving shipping hazards to the plant were analyzed. Scenarios involving ship breakthrough over the breakwater in its normal state (not degraded by heavy wave action) were shown to be not possible due to the speed required to generate the kinetic energy needed to physically force a passage through the breakwater. Scenarios involving oil spills and other floating debris were also shown to not have a significant consequence. Analysis scenarios involving a degraded breakwater, and therefore greatly increasing the possibility of a ship arriving in the intake cove, result in an estimated CDF of 2.90 x 10^{-8} per year. Scenarios involving a ship blocking the flow of water into the intake cove result in a conservatively estimated core damage frequency of 2.91 x 10^{-8} per year. Both of these CDF frequencies are low enough to be screened out.	Transportation accidents involving ships cannot cause damage to the plant under normal conditions, and are shown to have a bounding CDF of significantly less than 1 x 10^{-6} per year for degraded conditions. It is therefore concluded that transportation accidents are not significant.
Tsunami	4, B	Flooding of the intake structure due to a tsunami has an estimated CDF of 2.2 x 10^{-8} per year which is low enough to be screened out.	A tsunami is shown to have a bounding CDF of significantly less than 1 x 10^{-6} per year. It is therefore concluded that the tsunami hazard is not significant.
Turbine-Generated Missiles	SRP	Factory test procedures, redundancy in the control system, and routine testing of the main steam valves and the mechanical emergency over speed protection system while the unit is carrying load make the generation of missiles by a turbine runaway that might penetrate the turbine casing highly improbable.	The turbine missile hazard is judged to be not significant.

External Hazard	Screening Criteria	Evaluation	Disposition
		Therefore, turbine missiles can be screened based upon conformance with the Standard Review Plan.	
Volcanic Activity	3	Not applicable, no active volcanic mountains are near the plant.	Since the event is not applicable, there is no impact on categorization.
Waves	4	See the table entry for “External Flooding.”	See the table entry for “External Flooding.”

Disposition of Key Assumptions/Sources of Uncertainty

Process for Identification of Key Assumptions and Sources of Uncertainty:

The sources of model uncertainty and related assumptions are defined for the DCPP PRA models consistent with the guidance in Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, (Reference 13) and ASME/ANS RA-Sa-2009, "American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Probabilistic Risk Assessment (PRA) Standard," (Reference 17). The sources of model uncertainty and related assumptions have been identified for the DCPP baseline 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). These were reviewed by the respective hazard PRA peer review teams and closure review teams for internal events including internal flooding, fire, and seismic (see Attachment 3).

The detailed process of identifying, characterizing and qualitative screening of model uncertainties is in 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 approach taken is to review the assumptions and sources of uncertainty for each PRA model to identify the items which may be directly relevant to the categorization process.

Disposition of Key Assumptions and Sources of Uncertainty

The list of assumptions and sources of uncertainty were reviewed to identify those which would be significant for the categorization process of 10 CFR 50.69. If a DCPP model uses a nonconservative treatment identified as having a non-negligible impact, or uses methods which are not commonly accepted, the underlying assumption or source of uncertainty was reviewed to determine the impact on categorization. This evaluation meets the intent of steps C-1 and E-1 of NUREG-1855 (Reference 11). To assess the impact of these sources of uncertainties on the 10 CFR 50.69 application, a review of the base case sources of uncertainty for the PRA models was performed. Each identified uncertainty was evaluated with respect to its potential to significantly impact the categorization. This evaluation meets the intent of the screening portion for steps C-2 and E-2 of NUREG-1855 (Reference 11).

In addition, for the 10 CFR 50.69 Program, the guidance in NEI 00-04 (Reference 1) specifies sensitivity studies to be conducted for each PRA model to address key sources of uncertainty. The sensitivity studies are performed to ensure that assumptions and sources of uncertainty (e.g., human error, common cause failure, and maintenance probabilities) do not mask the SSC(s) importance.

Key assumptions and sources of uncertainty for the 10 CFR 50.69 application not screened as discussed above are identified and dispositioned in table below.

Disposition of Key Assumptions/Sources of Uncertainty Impacting 10 CFR 50.69		
Assumption/Uncertainty	Discussion	Disposition for 10 CFR 50.69
Dual unit trips (except for seismic events) are not considered in the single unit model, and crosstie to the other unit's resources may be unavailable.	The effects of dual unit trips and events may not be considered in accident sequences. This approach is nonconservative because the plant equipment credited may be required by the second unit and be unavailable for crosstie.	Sensitivity studies will be performed for impacted SSCs.
Charging and Safety Injection (SI) pumps are credited for inventory make-up for a medium LOCA. It is assumed that 2 out of 4 high pressure injection pumps (charging or SI) are required for success; this was conservatively modeled as 1 out of 2 charging pumps and 1 out of 2 SI pumps. To eliminate this modeling conservatism when all support is available and when 2 out of 2 charging pumps are required, a conservative estimate of the charging system failure fraction is to multiply the split fraction value for 1 of 1 pump train unavailability (CH2) by a factor of 2. Thus, the recovery factor (or conservatism reduction factor) for these conditions is 2*CH2.	The impact should be minimal for the baseline PRA model as the conservatism in the scenario of 1 out of 1 pump available is compensated with a factor of 2 when 2 out of 2 pumps are required.	This is a conservative approach and should not have a significant impact on the baseline PRA model, and therefore no significant impact on categorization results.

Disposition of Key Assumptions/Sources of Uncertainty Impacting 10 CFR 50.69		
Assumption/Uncertainty	Discussion	Disposition for 10 CFR 50.69
A 6-hour mission time for the emergency diesel generators (DGs) and fuel oil transfer pumps is assumed sufficient for non-seismic initiators rather than the standard 24-hour mission time.	A DG mission time of 6 hours is used for initiating events since the probability of non-recovery of offsite power after 6 hours is sufficiently small. The probabilities of non-recovery values are much larger than the fail to run values of the DGs and fuel oil transfer pumps for the remaining 18 hours of the mission time.	The 6-hour mission time of the DGs does not have a significant impact on the baseline PRA model, and therefore no significant impact on categorization results.
Containment penetrations which would require the failure of three or more valves are screened from the containment isolation analysis.	Lines that require the failure of three or more valves to cause the failure of containment isolation have a negligible contribution to the containment isolation failure frequency.	Screening these penetrations from the baseline model is a realistic treatment of a low failure probability function, and would not adversely impact categorization results.
Vacuum breakers cannot fail in a manner to impact the Auxiliary Salt Water (ASW) function on each unit within the 24-hour mission time.	The magnitude of the uncertainty attributable to this nonconservative assumption is not known. Although the magnitude of the non-conservatism is expected to be small, it has not been quantitatively demonstrated.	There are two vacuum relief valves per ASW header. They are mechanical components with a relatively high reliability, thus the random failure of an ASW header due to failing both vacuum relief valves should not be significant. This uncertainty would not significantly impact categorization results.

Disposition of Key Assumptions/Sources of Uncertainty Impacting 10 CFR 50.69		
Assumption/Uncertainty	Discussion	Disposition for 10 CFR 50.69
SI minimum flow valves are not modeled.	Failure of the minimum flow valves could result in flow diversion and impact the success criteria non-conservatively.	SI Recirculation Valves 8974A and 8974B are in series, thus both valves must be impacted, which has a low probability. The operator action to close these valves is also evaluated in the human reliability analysis (HRA) for switchover to cold leg recirculation. For categorization of these valves, the human action may be used as a surrogate.
Designation of systems/components as always failed in the fire PRA model and seismic PRA model.	Assuming certain systems and components are always failed for all fires or seismic events is conservative.	This conservative assumption is an accepted practice in fire and seismic PRA.
Pump runout protection on each unit is only modeled for Auxiliary Feedwater Pump 1-2 and 2-2 and is always successful for pump 1-3 and 2-3.	This is a model simplification and applies to sequences involving multiple steam generator depressurization.	In order for more than one steam generator to depressurize in a steam line break event, multiple main steam isolation valves (MSIV) must fail concurrently. Given the low failure probability of the MSIV function and low risk contribution of a steam line break event (approximately 5×10^{-9} CDF), the resulting risk contribution is expected to be less than 1×10^{-10} CDF. For purpose of categorization, the unmodeled components would be assigned the same preliminary HSS/LSS as the modeled components for the PRA modeled risk evaluation.

Disposition of Key Assumptions/Sources of Uncertainty Impacting 10 CFR 50.69		
Assumption/Uncertainty	Discussion	Disposition for 10 CFR 50.69
The recirculation valves to the refueling water storage tank (RWST) are not modeled.	The recirculation valves are interlocked to prevent pump discharge to the RWST during the recirculation phase. If these valves failed, long term cooling during recirculation could be lost.	Two valves must both fail to close; the combined failure probability would be approximately $5x10^{-7}$, which is insignificant, and therefore has no significant impact on categorization results.