UNITED STATES
NUCLEAR REGULATORY COMMISSION
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January 22, 2019

Mr. James M. Welsch
Vice President, Nuclear Generation
and Chief Nuclear Officer
Pacific Gas and Electric Company
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SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2018-JLD-0006)

Dear Mr. Welsch:

The purpose of this letter is to document the staff's evaluation of the Diablo Canyon Power Plant, Unit Nos. 1 and 2 (Diablo Canyon) seismic probabilistic risk assessment (SPRA) which was submitted in response to Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic." The U.S. Nuclear Regulatory Commission (NRC) has concluded that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required for Diablo Canyon.

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the NRC issued a request for information under Title 10 of the *Code of Federal Regulations* Section 50.54(f) (hereafter referred to as the 50.54(f) letter). The request was issued as part of implementing lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant. Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate seismic hazards at their sites using present-day methodologies and guidance. Enclosure 1, Item (8), of the 50.54(f) letter requested that certain licensees complete an SPRA to determine if plant enhancements are warranted due to the change in the reevaluated seismic hazard compared to the site's design-basis seismic hazard.

By letter dated April 24, 2018 (ADAMS Accession No. ML18120A201), Pacific Gas and Electric Company (PG&E, the licensee), provided its SPRA report in response to Enclosure 1, Item (8) of the 50.54(f) letter, for Diablo Canyon. The NRC staff assessed the licensee's implementation of the Electric Power Research Institute's Report 1025287, "Seismic Evaluation Guidance - Screening, Prioritization, and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML12333A170), as endorsed by NRC letter dated February 15, 2013 (ADAMS Accession No. ML12319A074), through the completion of the reviewer checklist in Enclosure 1 to this letter. As described below, the NRC has concluded that Diablo Canyon's SPRA report meets the intent of the SPID guidance and that the results and risk insights provided by the SPRA

support the NRC's determination that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

BACKGROUND

The 50.54(f) letter requested, in part, that licensees reevaluate the seismic hazards at their sites using updated hazard information and current regulatory guidance and methodologies. The request for information and the subsequent NRC evaluations have been divided into two phases:

> **Phase 1**: Issue 50.54(f) letters to all operating power reactor licensees to request that they reevaluate the seismic and flooding hazards at their sites using updated seismic and flood hazard information and present-day regulatory guidance and methodologies and, if necessary, to request they perform a risk evaluation.
>
> **Phase 2**: Based upon the results of Phase 1, the NRC staff will determine whether additional regulatory actions are necessary (e.g., updating the design basis and structures, systems, and components important to safety) to provide additional protection against the updated hazards.

By letter dated March 11, 2015 (ADAMS Accession No. ML15070A607), PG&E submitted the reevaluated seismic hazard information for the Diablo Canyon site. The NRC performed a staff assessment of the submittal and issued a response letter on December 21, 2016 (ADAMS Accession No. ML16341C057). The NRC's assessment concluded that the licensee conducted the hazard reevaluation using present-day regulatory guidance and methodologies, appropriately characterized the site, and met the intent of the guidance for determining the reevaluated seismic hazard.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC documented a determination of which licensees were to perform: (1) an SPRA; (2) limited scope evaluations; or (3) no further actions based on, among other factors, a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. As documented in that letter, Diablo Canyon was expected to complete an SPRA, which would also assess high frequency ground motion effects, and a limited-scope evaluation for the spent fuel pool. These seismic evaluations were expected to be submitted to the NRC by September 30, 2017, and December 31, 2017, respectively. Subsequently in a letter dated September 6, 2017 (ADAMS Accession No. ML17249A431), the licensee requested an extension of the submittal date for the SPRA until April 30, 2018. In a letter dated October 23, 2017 (ADAMS Accession No. ML17269A177), the staff deferred the SPRA submittal required response date until April 30, 2018.

The completion of the December 21, 2016, NRC staff assessment for the reevaluated seismic hazard and the scheduling of Diablo Canyon's SPRA report submittal described in the NRC's October 27, 2015, letter marked the fulfillment of the Phase 1 process for Diablo Canyon.

In its April 24, 2018, letter, PG&E provided the SPRA report that initiated the NRC's Phase 2 decisionmaking process for Diablo Canyon. The NRC described this Phase 2 decisionmaking process in a guidance memorandum from the Director of the Japan Lessons-Learned Division to the Director of the Office of Nuclear Reactor Regulation (NRR) on September 21, 2016 (ADAMS Accession No. ML16237A103). This memorandum details a Senior Management Review Panel (SMRP) consisting of three NRR Division Directors that are expected to reach a screening

decision for each plant submitting an SPRA. The SMRP is supported by appropriate technical staff who are responsible for consolidating relevant information and developing the recommendation for the screening decisions for consideration by the panel. In presenting recommendations to the SMRP, the supporting technical staff is expected to recommend placement of each SPRA plant into one of three groups:

1) **Group 1** includes plants for which available information indicates that further regulatory action is not warranted. For seismic hazards, Group 1 includes plants for which the mean seismic core damage frequency (SCDF) and mean seismic large early release frequency (SLERF) clearly demonstrate that a plant-specific backfit would not be warranted.

2) **Group 2** includes plants for which further regulatory action should be considered under the NRC's backfit provisions. This group may include plants with relatively large SCDF or SLERF, such that the event frequency in combination with other factors results in a risk to public health and safety for which a regulatory action is expected to provide a substantial safety enhancement.

3) **Group 3** includes plants for which further regulatory action may be needed, but for which more thorough consideration of both qualitative and quantitative risk insights is needed before determining whether a formal backfit analysis is warranted.

The evaluation process that was performed to provide the basis for the staff's grouping recommendation to the SMRP for Diablo Canyon is described below. As described below, the staff recommended to the SMRP that Diablo Canyon be classified as a Group 1 plant and therefore, no further regulatory action was warranted.

EVALUATION

Upon receipt of the licensee's April 24, 2018, SPRA report, a technical team of staff performed a completeness review to determine if the necessary information to support Phase 2 decisionmaking had been included in the licensee's submittal. The technical team performing the review consisted of staff experts in the fields of seismic hazards, fragilities evaluations, and plant response/risk analysis. On June 29, 2018, the technical team determined that sufficient information was available to perform the detailed technical review in support of the Phase 2 decision.

As described in the 50.54(f) letter, the staff's detailed review focused on verifying the technical adequacy of the licensee's SPRA such that an appropriate level of confidence could be placed in the results and risk insights of the SPRA to support regulatory decisionmaking associated with the 50.54(f) letter. As stated in its April 24, 2018, submittal, the licensee developed and documented the SPRA in accordance with the SPID guidance, including performing a full-scope peer review against Part 5 of Addendum B to the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (RA-Sb-2013). In addition, the licensee also performed a close-out independent assessment of the resolution of the finding level facts and observations (F&Os) from the full-scope peer review following the process accepted by the NRC (ADAMS Accession No. ML17079A427). The close-out independent assessment also included a concurrent focused-scope peer review. The close-out independent assessment resulted in the closure of all finding level F&Os for the Diablo Canyon

SPRA. Appendix A of the licensee's submittal provided a summary of the full-scope and independent assessment peer reviews including, excerpts from the corresponding peer review reports. It should be noted that NRC staff were present as observers for both the full-scope as well as the close-out independent assessment.

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the 50.54(f) letter. The list of applicable licensees in Enclosure 1 of the July 6, 2017, letter included PG&E as the licensee for Diablo Canyon. The staff exercised the audit by reviewing licensee documents via an electronic reading room (eportal) as documented in Enclosure 2 to this letter.

The staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA report. The staff's clarification questions dated September 9, 2018 (ADAMS Accession No. ML18263A150), were sent to the licensee to support the audit. The licensee subsequently provided answers to the questions in the eportal, which the staff reviewed. The close-out independent assessment report was available to the staff on the eportal. The staff sampled the resolutions to the findings as well as the close-out independent assessment team's conclusions from that report. The findings from the licensee's internal events PRA were not provided in the submittal, therefore the NRC staff reviewed PG&E's F&Os from the internal events PRA peer review, and associated dispositions, provided in the Diablo Canyon license amendment request to transition to NFPA 805 dated June 26, 2013 (ML13196A139), and associated supplements as described in the NRC staff safety evaluation dated April 14, 2016 (ADAMS Accession No. ML16035A441). As part of the audit, the licensee informed the staff that its SPRA model is based on the current internal events PRA model, which includes the disposition to all of the internal events PRA peer review F&Os.

The staff determined that the answers to the questions provided in the eportal served to verify statements that the licensee made in its April 24, 2018, SPRA report submittal. Based on the staff's review of the licensee's submittal, including the resolution of the peer review findings as described above, the NRC staff concluded that the technical adequacy of the licensee's SPRA submittal was sufficient to support regulatory decisionmaking associated with Phase 2 of the 50.54(f) letter.

The staff's review process included the completion of the SPRA Submittal Technical Review Checklist (SPRA Checklist) contained in Enclosure 1 to this letter. As described in Enclosure 1, the SPRA Checklist is a document used to record the staff's review of licensees' SPRA submittals against the applicable guidance of the SPID in response to the 50.54(f) letter. The SPRA Checklist also focuses on areas where the SPID contains differing guidance from standard industry SPRA guidance. Enclosure 1 contains the staff's application of the SPRA checklist to Diablo Canyon's submittal. As documented in the checklist, the staff concluded that the Diablo Canyon SPRA met the intent of the SPID. The staff further concluded that the peer review findings have been closed-out using an NRC-accepted process.

Following the staff's conclusion on the SPRA's technical adequacy, the staff reviewed the risk and safety insights contained in the Diablo Canyon SPRA report. The staff also used the screening criteria described in the August 29, 2017, staff memorandum titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic"

(ADAMS Accession No. ML17146A200) to assist in determining the group in which the technical team would recommend placing Diablo Canyon to the SMRP. The criteria in the staff's guidance document includes thresholds to assist in determining whether or not to apply the backfit screening process described in Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," dated October 9, 2013 (ADAMS Accession No. ML12059A460), to the SPRA report review. The Diablo Canyon SPRA report demonstrated that the plant SCDF and SLERF for both Units were not below the initial screening values in the August 29, 2017, staff memorandum. As a result, the NRC staff utilized the Diablo Canyon SPRA report and other available information in conjunction with the guidance in the August 29, 2017, memorandum to complete a detailed screening with respect to SCDF and SLERF for the Diablo Canyon. A discussion of the detailed screening evaluation completed by the NRC staff is provided in Enclosure 3 to this letter.

Based on the detailed screening evaluation and its review of the Diablo Canyon SPRA report, the technical team determined that recommending Diablo Canyon to be classified as a Group 1 site was appropriate and a plant-specific backfit was not warranted.

As a part of the Phase 2 decisionmaking process for SPRAs, the NRC formed the Technical Review Board (TRB), a board of senior-level NRC subject matter experts, to ensure consistency of review across the spectrum of plants that will be submitting SPRA reports. The technical team provided the results of the Diablo Canyon review to the TRB with the Phase 2 recommendation that Diablo Canyon be categorized as a Group 1 plant, meaning that no further response or regulatory actions are required. The TRB members assessed the information presented by the technical team and agreed with the team's recommendation for classification of Diablo Canyon as a Group 1 plant.

Subsequently, the technical team met with the SMRP and presented the results of the review including the recommendation for Diablo Canyon to be categorized as a Group 1 plant. The SMRP members also asked questions and provided input to the technical team. The SMRP approved the staff's recommendation that Diablo Canyon should be classified as a Group 1 plant, meaning that no further response or regulatory action is required.

AUDIT REPORT

The July 6, 2017, generic audit plan describes the NRC staff's intention to issue an audit report that summarizes and documents the NRC's regulatory audit of licensee's submittals associated with reevaluated seismic hazard analyses. The NRC staff's Diablo Canyon audit included a review of licensee documents through an electronic reading room. An audit summary document is included as Enclosure 2 to this letter.

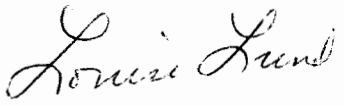
CONCLUSION

Based on the staff's review of the Diablo Canyon submittal against the endorsed SPID guidance, the NRC staff concludes that the licensee responded appropriately to Enclosure 1, Item (8) of the 50.54(f) letter. Additionally, the staff's review concluded that the SPRA is of sufficient technical adequacy to support Phase 2 regulatory decisionmaking in accordance with the intent of the 50.54(f) letter. Based on the results and risk insights of the SPRA report, the NRC staff also concludes that no further response or regulatory actions associated with NTTF Recommendation 2.1 "Seismic" are required.

Application of this review is limited to the review of the 10 CFR 50.54(f) response associated with NTTF Recommendation 2.1 "Seismic" review. The staff notes that assessment of the SPRA for use in other licensing applications would warrant review of the SPRA for its intended application. The NRC may use insights from this SPRA assessment in its regulatory activities as appropriate.

If you have any questions, please contact Joseph Sebrosky at (301) 415-1132 or via e-mail at Joseph.Sebrosky@nrc.gov.

Sincerely,



Louise Lund, Director
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Docket Nos. 50-275 and 50-323

Enclosures:
1. NRC Staff SPRA Submittal Technical Review Checklist
2. NRC Staff Audit Summary
3. NRC Staff SPRA Submittal Detailed Screening Evaluation

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NRC Staff SPRA Submittal Technical Review Checklist

Several nuclear power plant licensees are performing seismic probabilistic risk assessments (SPRAs) as part of their required submittals to satisfy Near-Term Task Force (NTTF) Recommendation 2.1: Seismic. These submittals are prepared according to the guidance in the Electric Power Research Institute – Nuclear Energy Institute (EPRI-NEI) Screening, Prioritization, and Implementation Details (SPID) document (EPRI-SPID, 2012), which was endorsed by the staff for this purpose (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12319A074). The SPRA peer reviews are also expected to follow the guidance in NEI 12-13 (NEI, 2012).

The SPID indicates that an SPRA submitted to satisfy NTTF Recommendation 2.1: Seismic must meet the requirements in the ASME-ANS Probabilistic Risk Assessment (PRA) Methodology Standard (the ASME/ANS Standard). Either the "Addendum A version" (ASME/ANS Addendum A, 2009) or the "Addendum B version" (ASME/ANS Addendum B, 2013) of the ASME/ANS Standard can be used.

Tables 6-4, 6-5, and 6-6 of the SPID also provide a comparison of each of the Supporting Requirements (SRs) of the ASME/ANS Standard to the relevant guidance in the SPID. For most SRs, the SPID guidance does not differ from the requirement in the ASME/ANS Standard. However, because the guidance of the SPID and the criteria of the ASME/ANS Standard differ in some areas, or the SPID does not explicitly address an SR, the staff developed this checklist, in part, to help staff members to address and evaluate the differences.

In general, the SPID allowed departures or differed from the ASME/ANS Standard in the following ways:

(i) In some technical areas, the SPID's requirements tell the SPRA analyst "how to perform" one aspect of the SPRA analysis, whereas the ASME/ANS Standard's requirements generally cover "what to do" rather than "how to do it".

(ii) For some technical areas and issues, the requirements in the SPID differ from those in the ASME/ANS Standard.

(iii) The SPID has some requirements that are not in the ASME/ANS Standard.

All the technical positions in the SPID have been endorsed by the U.S. Nuclear Regulatory Commission (NRC) staff, subject to certain conditions concerning peer review outlined in the staff November 12, 2012, letter to NEI (NRC, 2012).

The following checklist is comprised of the 16 "Topics" that require additional staff guidance because the SPID contains specific guidance that differs from the ASME/ANS Standard or expands on it. Each is covered below under its own heading, "Topic 1," "2," etc. The checklist was discussed during a public meeting held on December 7, 2016 (ADAMS Accession No. ML16350A181).

- Topic 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)
- Topic 2: Site Seismic Response (SPID Section 2.4)
- Topic 3: Definition of the Control Point for the Safe Shutdown Earthquake (SSE) to Ground Motion Response Spectrum (GMRS) Comparison Aspect of the Site Analysis (SPID Section 2.4.2)
- Topic 4: Adequacy of the Structural Model (SPID Section 6.3.1)
- Topic 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)
- Topic 6: Use of Seismic Response Scaling (SPID Section 6.3.2)
- Topic 7: Use of New Response Analysis for Building Response, In-Structure Response Spectra (ISRS), and Fragilities
- Topic 8: Screening by Capacity to Select Structures, Systems, and Components (SSCs) for Seismic Fragility Analysis (SPID Section 6.4.3)
- Topic 9: Use of the Conservative Deterministic Failure Margin (CDFM)/HybridMethodology for Fragility Analysis (SPID Section 6.4.1)
- Topic 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)
- Topic 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)
- Topic 13: Evaluation of Seismic Large Early Release Frequency (SLERF) (SPID Section 6.5.1)
- Topic 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)
- Topic 15: Documentation of the SPRA (SPID Section 6.8)
- Topic 16: Review of Plant Modifications and Licensee Actions

TOPIC 1: Seismic Hazard (SPID Sections 2.1, 2.2, and 2.3)

The site under review has updated/revised its Probabilistic Seismic Hazard Analysis (PSHA) from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	No
The guidance in the SPID was followed for developing the site's probabilistic seismic hazard.	N/A
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the Probabilistic Seismic Hazards Analysis (SHA) requirements in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The guidance in the SPID was followed for developing the probabilistic seismic hazard for the site.	Yes
• An alternate approach was used, and is acceptable on a justified basis.	N/A

TOPIC 2: Site Seismic Response (SPID Section 2.4)

The site under review has updated/revised its site response analysis from what was submitted to NRC in response to the NTTF Recommendation 2.1: Seismic 50.54(f) letter.	No
The guidance in the SPID was followed for developing a site profile for use in the analysis to develop control point seismic hazard curves (site response).	Yes
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs SHA-E1 and E2 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's development of PSHA inputs and base rock hazard curves meets the intent of the SPID guidance or another acceptable approach.	N/A
• The licensee's development of a site profile for use in the analysis adequately meets the intent of the SPID guidance or another acceptable approach.	N/A
• Although the licensee's development of a V_s velocity profile for use in the analysis does not meet the intent of the SPID guidance, it is acceptable on another justified basis.	N/A

TOPIC 3: Definition of the Control Point for the SSE to GMRS Comparison Aspect of the Site Analysis (SPID Section 2.4.2)

The issue is establishing the control point where the SSE is defined. Most sites have only one SSE, but some sites have more than one SSE, for example one at rock and one at the top of the soil layer. This control point is needed because it is used as part of the input information for the development of the seismic site-response analysis, which in turn is an important input for analyzing seismic fragilities in the SPRA. The SPID (Section 2.4.1) recommends one of two criteria for establishing the control point for a logical SSE-to-GMRS comparison:	
A) If the SSE control point(s) is defined in the final safety analysis report (FSAR), it should be used as defined.	Yes
B) If the SSE control point is not defined in the FSAR, one of three criteria in the SPID (Section 2.4.1) should be used.	N/A
C) An alternative method has been used for this site.	N/A
The control point used as input for the SPRA is identical to the control point used to establish the GMRS.	Yes
If yes, the control point can be used in the SPRA and the NRC staff's earlier acceptance governs.	
If no, the NRC staff's previous reviews might not apply. The staff's review of the control point used in the SPRA is acceptable.	N/A
Notes from staff reviewer: None. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's definition of the control point for site response analysis adequately meets the intent of the SPID guidance.	Yes
• The licensee's definition of the control point for site response analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 4: Adequacy of the Structural Model (SPID Section 6.3.1)

The NRC staff review of the structural model finds an acceptable demonstration of its adequacy.	Yes
Used an existing structural model	No
Used an enhancement of an existing model	No
Used an entirely new model	Yes
Criteria 1 through 7 (SPID Section 6.3.1) are all met.	Yes

Notes from staff reviewer:

Section 4.3.3 of the Diablo Canyon Power Plant, Unit Nos. 1 and 2 (DCPP, Diablo Canyon) SPRA report explains that new three-dimensional finite element structural models (FEMs) were developed for the Containment Structures, Auxiliary Building, Turbine Building, and Intake Structure. Because FEMs use shell and beam elements to explicitly represent the structural boundary of the modeled structures, the criteria in Section 6.3.1 of the SPID regarding lumped-mass stick models (LPSMs) and in-plane floor flexibility are not directly applicable to the DCPP structural analyses. Table 4-2 of the DCPP SPRA report explains that all three dimensions of the earthquake ground motion are evaluated in the FEM models. Sections 4.3.1 and 4.3.2 of the DCPP SPRA report explains that the FEMs included modes up to 33 Hertz (Hz), which is greater than the SPID criteria of 20 Hz.

All peer review finding-level F&Os associated with SFR-C1 through SFR-C6 have been closed by a process acceptable to the NRC (see Topic #14).

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Fragility Analysis (SFR)-C1 through C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's structural model meets the intent of the SPID guidance.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 5: Use of Fixed-Based Dynamic Seismic Analysis of Structures for Sites Previously Defined as "Rock" (SPID Section 6.3.3)

Fixed-based dynamic seismic analysis of structures was used, for sites previously defined as "rock."	Yes
If no, this issue is moot. If yes, on which structure(s)? Structure #1 name: Structure #2 name:	
Structure #1: If used, is V_S > about 5000 feet (ft.)/second (sec.)?	N/A
If 3500 ft./sec. < V_S < 5000, was peak-broadening or peak shifting used?	N/A
Potential Staff Finding: The demonstration of the appropriateness of using this approach is adequate.	Yes

Notes from staff reviewer:

Section 4.3.2 of the DCPP SPRA report explains that a fixed-based dynamic seismic analysis was performed for the Intake Structure. This analysis was composed of three elements: (1) a three-dimensional finite element model of the structure, (2) use of the structure-specific foundation input response spectra (FIRS) for the horizontal In-structure Response Spectra (ISRS) since the structure is founded on rock and is embedded on three sides in a cliff, and (3) performance of a fixed based probabilistic response analysis to obtain the vertical ISRS. The vertical ISRS was developed from 30 time history response simulations, and median and 84 percent non-exceedance probabilities evaluated at selected locations in the intake structure.

Based on this approach to developing a location-specific ISRS, the NRC staff concludes that peak-broadening or peak shifting was not necessary.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this topic.	N/A

• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" adequately meets the intent of the SPID guidance.	Yes
• The licensee's use of fixed-based dynamic analysis of structures for a site previously defined as "rock" does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 6: Use of Seismic Response Scaling (SPID Section 6.3.2)

Seismic response scaling was used.	No
Potential Staff Findings: If a new uniform hazard spectra or review level earthquake is used, the shape is approximately similar to the spectral shape previously used for ISRS generation. If the shape is not similar, the justification for seismic response scaling is adequate. Consideration of non-linear effects is adequate.	N/A N/A N/A
Notes from staff reviewer: Section 4.3 of the DCPP SPRA report explains that completely new finite element models were developed for the Containment Structures, Auxiliary Building, and Turbine Building and that probabilistic soil-structure-interaction (SSI) analyses were performed for these structures. Therefore, no scaling was necessary to develop the ISRS. The peer review finding-level F&Os associated with SFR-C3 has been closed by a process acceptable to the NRC (see Topic #14).	
Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): None.	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-C3 in the ASME/ANS Standard, as well as to the requirements in the SPID. • Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis. • The licensee's use of seismic response scaling adequately meets the intent of the SPID guidance. • The licensee's use of seismic response scaling does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A N/A N/A N/A

TOPIC 7: Use of New Response Analysis for Building Response, ISRS, and Fragilities

The SPID does not provide specific guidance on performing new response analysis for use in developing ISRS and fragilities. The new response analysis is generally conducted when the criteria for use of existing models are not met or more realistic estimates are deemed necessary. The requirements for new analysis are included in the ASME/ANS Standard. See SRs SFR-C2, C4, C5, and C6. One of the key areas of review is consistency between the hazard and response analyses. Specifically, this means that there must be consistency among the ground motion equations, the SSI analysis (for soil sites), the analysis of how the seismic energy enters the base level of a given building, and the in-structure-response-spectrum analysis. Said another way, an acceptable SPRA must use these analysis pieces together in a consistent way. The following are high-level key elements that should have been considered:	
1. FIRS site response developed with appropriate building specific soil velocity profiles. Structure #1 name: Diablo Canyon Containment Structures Structure #2 name: Diablo Canyon Auxiliary Building Structure #3 name: Diablo Canyon Turbine Building Structure #4 name: Diablo Canyon Intake Building Structure #5 name: Diablo Canyon Outdoor Water Storage Tanks Structure #6 name: 230kV Switchyard Are all structures appropriately considered?	Yes Yes Yes Yes Yes Yes Yes
2. Are models adequate to provide realistic structural loads and response spectra for use in the SPRA? • Is the SSI analysis capable of capturing uncertainties and realistic? • Is the probabilistic response analysis capable of providing the full distribution of the responses?	Yes Yes Yes

Notes from staff reviewer:

Section 3.1.1 of the DCPP SPRA report explains that FIRS (horizontal or vertical or both) were developed for the six structures identified above [note: outdoor water storage tanks (OWSTs) include the Refueling Water Storage Tanks (RWSTs), Condensate Storage Tanks (CSTs), and Fire Water and Transfer Tank (FWTT)]. Section 4.3 explains that new three-dimensional finite element analyses and probabilistic SSI analyses were performed to develop the FIRS for the Containment Structures, Auxiliary Building, and Turbine Building using site-specific soil properties (including both shear wave and compression wave velocities). Uncertainties in velocity profiles and soil properties were included in the analyses and 30 Latin hypercube sampling (LHS) simulations were performed to develop probabilistic response distributions.

The Intake Structure and OWSTs are founded in rock and so SSI analyses were determined to not be necessary for these structures. The 230 kV switchyard FIRS was developed using site-specific site response analysis.

Section 3.1.2 of the DCPP SPRA report states "The DCPP SPRA hazard methodology and analysis associated with the horizontal GMRS were submitted to the NRC as part of [3, 4][1], and found to be technically acceptable by the NRC for application to the DCPP SPRA, as documented in their staff assessment [5][2]. No changes to the hazard were made subsequent to these submittals."

All peer review finding-level F&Os associated with SFR-C2, SFR-C4, SFR-C5, and SFR-C6 have been closed by a process acceptable to the NRC (see Topic #14).

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs SFR-C2, C4, C5, and C6 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's FIRS modeling is consistent with the prior NRC review of the GMRS and soil velocity information.	Yes
• The licensee's structural model meets the intent of the SPID guidance and the ASME/ANS Standard's requirements.	Yes

[1] ADAMS Accession Nos. ML15070A607, ML15355A550, and ML15355A551.
[2] ADAMS Accession No. ML16341C057,

• The response analysis accounts for uncertainties in accordance with the SPID guidance and the ASME/ANS Standard's requirements.	Yes
• The NRC staff concludes that an acceptable consistency has been achieved among the various analysis pieces of the overall analysis of site response and structural response.	Yes
• The licensee's structural model does not meet the intent of the SPID guidance and the ASME/ANS Standard's requirements, but is acceptable on another justified basis.	N/A

TOPIC 8: Screening by Capacity to Select SSCs for Seismic Fragility Analysis (SPID Section 6.4.3)

The selection of SSCs for seismic fragility analysis used a screening approach by capacity following Section 6.4.3 of the SPID. If no, see items D and E. If yes, see items A, B, and C. Potential Staff Findings:	Yes
A) The recommendations in Section 6.4.3 of the SPID were followed for the screening aspect of the analysis, using the screening criteria therein.	Yes
B) The approach for retaining certain SSCs in the model with a screening-level seismic capacity follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
C) The approach for screening out certain SSCs from the model based on their inherent seismic ruggedness follows the recommendations in Section 6.4.3 of the SPID and has been appropriately justified.	Yes
D) The ASME/ANS Standard has been followed.	N/A
E) An alternative method has been used and its use has been appropriately justified.	N/A

Notes from staff reviewer:

Section 4.1 of the Diablo Canyon SPRA report describes the development of the seismic equipment list (SEL) for Unit 1 and explains that this SEL is applicable to Unit 2 because the two Units are essentially identical. In addition Section 4.2 of the Diablo Canyon SPRA states that since the SPRA is applicable to both Units 1 and 2, walk-bys of the corresponding Unit 2 components were also conducted to confirm similarity between the components in Unit 1 and 2. Section 5.3.2 of the Diablo Canyon SPRA report identifies the fire water storage tank (FWST) as being a shared water source for Units 1 and 2. Section 5.3.2 also notes that in the internal events PRA, credit for auxiliary saltwater (ASW) supply from the opposite unit is modeled. Opposite unit ASW support systems (4kV and 125V DC) are also credited. However, for seismic events, full correlation is assumed between units for ASW, 4kV switchgear, EDGs, and 125V DC SSCs). No other credit for opposite unit equipment is assumed.

The methodology for screening high seismic capacity SSCs from the SPRA model is described in Section 4.4.1 of the DCPP SPRA report. This screening method is consistent with the methodology of Section 6.4.3 of the SPID in that SSCs were screened based on two criteria: (1) SSCs determined to be inherently seismically rugged based on industry experience and engineering judgement and (2) SSCs calculated to have an SCDF contribution less than 5×10^{-7} per year based on a screening level High Confidence of Low Probability of Failure (HCLPF) determined from convolving the fragility of the SSC with the seismic hazard curves.

Section A2.3.3 in Appendix A of the submittal makes the following statement: "Findings were established to address fragility cutoff at the HCLPF capacity...." During its audit of the submittal, the NRC staff observed that the use of the HCLPF cutoff in the SPRA quantification appears to be an important assumption in the development of the SPRA and therefore, as an audit question, the staff requested the licensee provide the impact of the removal of the "fragility cutoff" on (i) the risk metrics, (ii) the top seismic CDF and seismic LERF contributors, and (iii) the contribution from each 'bin' to the seismic CDF and seismic LERF. In its response to the audit question, the licensee explained that the finding related to the use of a HCLPF cutoff was resolved by providing additional supporting arguments in favor of a fragility truncation level and by lowering the truncation from a HCLPF cutoff to a cutoff at the 95 percent confidence of 1 percent of failure. This resolution was reviewed by an Appendix X closure review team and the F&O was closed.

The licensee also provided the results of a sensitivity analysis that assumed a "very low truncation level." In response to a follow-up audit question, the licensee clarified that the sensitivity used a fragility truncation level of $1x10^{-13}$ which is sufficiently low to determine the impact of the HCLPF cutoff. Tables of sensitivity analysis results were provided as follows: (1) the Fussel-Vesely (F-V) importance value for each of the 15 SSCs having the most significant seismic failure contribution to SCDF and SLERF, (2) the SCDF and SLERF contribution for each of the top 30 accident sequences, and (3) the SCDF and SLERF contribution for each of the 16 seismic hazard intervals. The licensee summarized the changes between the sensitivity analysis result and the submittal results as (1) the seismic fragility importance of instrumentation and control components increased, (2) several new accident sequences showed up in the top 30 list because of the increase in importance of instrumentation and control components and increase in importance of seismically-induced station blackout, and (3) the mean SCDF and SLERF increased by about $4E^{-06}$/reactor-year and $7x10^{-7}$/reactor-year, respectively, with the maximum SCDF and SLERF increase for an individual seismic hazard interval being less than $1E^{-06}$/reactor-year and $1.8E^{-07}$/reactor-year, respectively.

The NRC staff compared the sensitivity analysis results to those provided in the submittal and observed that (1) the top 10 SSCs having the most significant seismic failure contributions were common to both cases, (2) the F-V values for the top SSCs generally did not change appreciably, and the seismic contributions for those that did increase substantially (e.g., TDAFW motor-operated valves for SCDF and 120 Volt AC instrument breaker panel for SLERF) did not alter the staff's decisionmaking, and (3) the mean SCDF and SLERF did not increase substantially (by 13.5 percent and 16.9 percent, respectively). Based on these results, the NRC staff concludes that the use of a fragility cutoff does not change the conclusion that DCPP should be considered a Group 1 plant.

In its response to an audit question on the approach for screening high-capacity SSCs, the licensee explained that the screening level median capacity corresponding to an SCDF of 5×10^{-7} per year was calculated to be 9.4g and that this was increased to 10.2g to ensure that the screening criterion was adequate. The licensee also explained that the sensitivity analysis assumed 10.2g, which resulted in a lower SCDF contribution for each robust SSC. The NRC staff finds the screening level chosen by the licensee for capacity-based screening of SSCs is acceptable because it results in less than 1% contribution to the total SCDF.

Also, all peer review finding-level F&Os associated with SFR-B1 have been closed by a process acceptable to the NRC (see Topic #14).

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-B1 in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of a screening approach for selecting SSCs for fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of a screening approach for selecting SSCs for fragility analysis does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 9: Use of the CDFM/Hybrid Methodology for Fragility Analysis (SPID Section 6.4.1)

The Conservative Deterministic Failure Margin (CDFM)/Hybrid method was used for seismic fragility analysis. If no, See item C) below and next issue. If yes: Potential Staff Findings:	Yes
A) The recommendations in Section 6.4.1 of the SPID were followed appropriately for developing the CDFM High Confidence Low Probability of Failure capacities.	Yes
B) The Hybrid methodology in Section 6.4.1 and Table 6-2 of the SPID was used appropriately for developing the full seismic fragility curves.	Yes
C) An alternative method has been used appropriately for developing full seismic fragility curves.	N/A

Notes from staff reviewer:

The DCPP SPRA report states that the fragility of SSCs included in the SPRA analysis was calculated primarily using the SOV method. The CDFM methodology was used to obtain fragility parameters for two components (which are not identified). The fragility parameters in Table 6-2 of the SPID were used in the screening of high seismic capacity SSCs (See Topic #8).

Fragility evaluations for the offsite power system, firewater piping, and various non-vital electrical panels were determined using earthquake experience data and industry consensus data. The peer review team judged that the fragility values used in the SPRA are indeed realistic. This is described in Sections 4.4.2 and A2.3.2 of the DCPP SPRA report.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	Yes
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis meets the intent of the SPID guidance.	Yes
• The licensee's use of the CDFM/Hybrid method for seismic fragility analysis does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 10: Capacities of SSCs Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain SSCs that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility using a methodology described in Section 6.4.2 of the SPID.	
Potential Staff Findings:	
The NRC staff review of the SPRA's fragility analysis of SSCs sensitive to high frequency seismic motion finds that the analysis is acceptable.	N/A
The flow chart in Figure 6-7 of the SPID was followed.	N/A
The flow chart was not followed but the analysis is acceptable on another justified basis.	N/A

Notes from staff reviewer:

The DCPP SPRA submittal fragility analysis performed for all structures, systems, and components (SSCs) are based on full range frequencies. Section 4.4.2 of the submittal states:

> *Note that the fragility analyses for SSCs are based on a full range of frequencies and the DCPP control point GMRS, unlike Central and Eastern United States plants, does not include peaks in the high frequency range (frequencies greater than 10 Hz). Therefore, separate "high-frequency capacity" evaluations of SSCs, including relays, as described in Section 6.4.2 of the SPID [2] are not applicable to DCPP.*

This is confirmed in Figure 3-1 of the DCPP SPRA report, which shows the peak ground acceleration, which corresponds to 100 Hz oscillator, to be significantly less than 10 Hz. Based on this, the NRC staff concludes that an analysis of high frequency responses of SSCs, as described in Section 6.4.2 of the SPID, is not applicable to DCPP.

Deviation(s) or deficiency(ies) and Resolution: None.

Consequence(s): None.

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SR SFR-F3 in the ASME/ANS Standard, as well as to the requirements in the SPID.	N/A
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's fragility analysis of SSCs sensitive to high frequency seismic motion meets the intent of the SPID guidance.	N/A
• The licensee's fragility analysis of SSCs sensitive to high-frequency motion does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

TOPIC 11: Capacities of Relays Sensitive to High-Frequencies (SPID Section 6.4.2)

The SPID requires that certain relays and related devices (generically, "relays") that are sensitive to high-frequency seismic motion must be analyzed in the SPRA for their seismic fragility. Although following the ASME/ANS Standard is generally acceptable for the fragility analysis of these components, the SPID (Section 6.4.2) contains additional guidance when either circuit analysis or operator-action analysis is used as part of the SPRA to understand a given relay's role in plant safety. When one or both of these are used, the NRC reviewer should use the following elements of the checklist.	
i) Circuit analysis: The seismic relay-chatter analysis of some relays relies on circuit analysis to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to circuit analysis for maintaining safety after seismic relay chatter is acceptable.	Yes Yes
ii) Operator actions: The relay-chatter analysis of some relays relies on operator actions to assure that safety is maintained. (A) If no, then (B) is moot. (B) If yes: Potential Staff Finding: The approach to analyzing operator actions for maintaining safety after seismic relay chatter is acceptable.	No N/A
Notes from staff reviewer: The DCPP SPRA report Section 4.1.2 states that an extensive contact chatter evaluation was performed in accordance with Section 5-2.2 of the ASME/ANS PRA Standard, which included a circuit analysis of all modeled components that could be impacted by contact chatter and identification of contacts that could have an impact on modeled PRA functions. All of these contacts were dispositioned by including the impact of chatter in	

the SPRA model. Based on this, the NRC staff concludes that operator actions were not credited in the SPRA model to mitigate the impact of contact chatter. Also, all peer review finding-level F&Os associated with SPR-B4, SPR-B4b, and SPR-B6 have been closed by a process acceptable to the NRC (see Topic #14). Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to the SRs Seismic Plant Response Analysis (SPR)-B6 (Addendum A) or SPR-B4 (Addendum B) in the ASME/ANS Standard, as well as to the requirements in the SPID. • Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis. • The licensee's analysis of seismic relay-chatter effects meets the intent of the SPID guidance. • The licensee's analysis of seismic relay-chatter effects does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	Yes N/A Yes N/A

TOPIC 12: Selection of Dominant Risk Contributors that Require Fragility Analysis Using the Separation of Variables Methodology (SPID Section 6.4.1)

The CDFM methodology has been used in the SPRA for analysis of the bulk of the SSCs requiring seismic fragility analysis. If no, the staff review will concentrate on how the fragility analysis was performed, to support one or the other of the "potential staff findings" noted just below. If yes, significant risk contributors for which use of SOV fragility calculations would make a significant difference in the SPRA results have been selected for SOV calculations.	No
Potential Staff Findings: A) The recommendations in Section 6.4.1 of the SPID were followed concerning the selection of the "dominant risk contributors" that require additional seismic fragility analysis using the SOV methodology.	Yes
B) The recommendations in Section 6.4.1 were not followed, but the analysis is acceptable on another justified basis.	N/A
Notes from staff reviewer: The DCPP SPRA report states that the fragility of SSCs included in the SPRA analysis was calculated primarily using the SOV method. The CDFM methodology was used to obtain fragility parameters for two components that were not significant contributors to seismic risk. Fragility evaluations for the offsite power system, firewater piping, and various non-vital electrical panels were determined using earthquake experience data and industry consensus data (see Topic #9). The peer review team judged that the fragility values used in the SPRA are indeed realistic. This is described in Section 4.4.2 of the DCPP SPRA report. Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred	Yes

to relate to the requirements in the SPID. No requirements in the ASME/ANS Standard specifically address this Topic.	
• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology meets the intent of the SPID guidance.	Yes
• The licensee's method for selecting the "dominant risk contributors" for further seismic fragilities analysis using the SOV methodology does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 13: Evaluation of SLERF (SPID Section 6.5.1)

The NRC staff review of the SPRA's analysis of SLERF finds an acceptable demonstration of its adequacy.	Yes
Potential Staff Findings:	
A) The analysis follows each of the elements of guidance for SLERF analysis in Section 6.5.1 of the SPID, including in Table 6-3.	Yes
B) The SLERF analysis does not follow the guidance in Table 6-3 but the analysis is acceptable on another justified basis.	NA

Notes from staff reviewer: None.

In accordance with the SPID, the internal events PRA (IEPRA) was adapted to include seismic-related basic events and the HEPs developed for the IEPRA HRA were re-evaluated to include the impact of seismic events. The containment isolation failure modes evaluated in the SPRA model are specifically described in Section 5.1.3 of the DCPP SPRA report. The LERF contributors listed in Table 6-3 of the SPID either had no significant seismic-induced impact (per Table 6-3); were determined not to apply to a PWR; or were judged to be addressed at a general level in the report.

The DCPP SPRA report does not discuss the impact of a seismic event on emergency plans, which is acceptable per the SPID for NTTF Recommendation 2.1.

See Topic 14 for the NRC staff review of the LERF-related IEPRA peer review finding-level F&Os (i.e., F&Os associated with PRA standard supporting requirements from technical element "LE").

Deviation(s) or deficiency(ies) and Resolution: None

Consequence(s): N/A

The NRC staff concludes that:	
• The peer review findings have been addressed and the analysis approach has been accepted by the staff for the purposes of this evaluation. The peer review findings referred to relate to SRs SFR-F4, SPR-E1, SPR-E2, and SPR-E6 (Addendum B only) in the ASME/ANS Standard, as well as to the requirements in the SPID.	Yes

• Although some peer review findings have not been resolved, the analysis is acceptable on another justified basis.	N/A
• The licensee's analysis of SLERF meets the intent of the SPID guidance.	Yes
• The licensee's analysis of SLERF does not meet the intent of the SPID guidance, but is acceptable on another justified basis.	N/A

TOPIC 14: Peer Review of the SPRA, Accounting for NEI 12-13 (SPID Section 6.7)

The NRC staff review of the SPRA's peer review findings, observations, and their resolution finds an acceptable demonstration of the peer review's adequacy.	Yes
Potential Staff Findings:	
A) The analysis follows each of the elements of the peer review guidance in Section 6.7 of the SPID.	Yes
B) The composition of the peer review team meets the SPID guidance.	Yes
C) The peer reviewers focusing on seismic response and fragility analysis have successfully completed the Seismic Qualifications Utility Group training course or equivalent (see SPID Section 6.7).	Yes
In what follows, a distinction is made between an "in-process" peer review and an "end-of-process" peer review of the completed SPRA report. If an in-process peer review is used, go to (D) and then skip (E). If an end-of-process peer review is used, skip (D) and go to (E).	
D) The "in process" peer-review process followed the guidance in the SPID (Section 6.7), including the three "bullets" and the guidance related to NRC's additional input in the paragraph immediately following those three bullets. These three bullets are: • The SPRA findings should be based on a consensus process, and not based on a single peer review team member • A final review by the entire peer review team must occur after the completion of the SPRA project • An "in-process" peer review must assure that peer reviewers remain independent throughout the SPRA development activity.	N/A
If no, go to (F).	

If yes, the "in process" peer review approach is acceptable. Go to (G).	
E) The "end-of-process" peer review process followed the peer review guidance in the SPID (Section 6.7). If no, go to (F). If yes, the "end-of-process" peer review approach is acceptable. Go to (G).	Yes
F) The peer-review process does not follow the guidance in the SPID but is acceptable on another justified basis.	N/A
G) The licensee peer-review findings were satisfactorily resolved or were determined not to be significant to the SPRA conclusions for this evaluation.	Yes

Notes from staff reviewer: None.

The DCPP SPRA report appears to follow the recommendations of Section 6.7 of the SPID. Section 5.2 and Appendix A describe the peer review process used to establish the technical adequacy of the SPRA. All elements of the SPRA were peer reviewed.

The SPRA was peer reviewed in June 2017, in accordance with: 1) NEI 12-13, "External Hazard PRA Peer Review Process Guidelines," Revision 0, dated August 2012 (ADAMS Accession No. ML122400044); 2) Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009 (ADAMS Accession No. ML090410014); and 3) Capability Category II requirements of PRA Standard ASME/ANS RA Sb-2013, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated September 30, 2013. The qualifications of each of the nine peer review members is described; one member was designated as the team leader, three members conducted the review of the SHA elements, three members conducted the review of the SFR elements, and two members conducted the review of the SPR elements. The DCPP SPRA report states that three of the peer review members had appropriate SQUG training, but does not specifically identify which of the peer review members had completed the SQUG training course. From the SPRA Peer Review Report (PWROG-17022-P), of the five members doing the review of the seismic response and fragility analysis, only one is identified as having completed the SQUG training course (the team leader was also identified as completing the SQUG training course). However, the resumes for each of the other four other SFR and SPR peer reviewers were shown to demonstrate significant seismic structural and SPRA experience, which are judged by the NRC staff to be "equivalent" to the SQUG training.

According to the DCPP SPRA report, the full-scope peer review team identified 47 finding-level F&Os. Subsequently, the SPRA model and documentation was updated to resolve the 47 F&Os. In December 2017, an Independent Assessment Team (IAT) performed an independent assessment to close-out the SPRA F&Os in accordance with the Appendix X process described in an NEI letter dated February 21, 2017, "Final Revision of Appendix X to NEI 05-04/07-12/12-16, Close-Out of Facts and Observations (F&Os)," (ADAMS Accession No. ML17086A450). The NRC's acceptance of NEI's approach outlined in the February 21, 2017, letter is documented in a letter dated May 3, 2017 (ADAMS Accession No. ML17079A427). The IAT, consisting of six members and a dedicated team lead, determined that all the F&Os had been closed. A process to ensure team member qualifications and independence in alignment with Appendix X guidance was established. Concurrence on the resolution of each finding was based on a consensus process involving all members of the review team.

Table A-2 of the DCPP SPRA report identified that the dispositions to two F&Os resulted in PRA model changes that are considered PRA upgrades per the ASME/ANS PRA Standard. The SRs associated with both those F&Os were reassessed by a focused scope peer review team (concurrent with the IAT) and determined to be met with no additional F&Os.

The DCPP SPRA report explains that the DCPP IEPRA model was the starting point for the development of the SPRA model. Hence, the NRC staff considered the technical adequacy of certain elements of the IEPRA model when evaluating the technical adequacy of the SPRA model for this application. The DCPP SPRA report provided no information on the technical adequacy of the IEPRA model, therefore the NRC staff reviewed PG&E's F&Os from the IEPRA peer review, and associated dispositions, provided in the Diablo Canyon LAR to transition to NFPA 805 (ADAMS Accession No. ML13196A139) and associated supplements as described in the NRC staff safety evaluation (ADAMS Accession No. ML16035A441).

The IEPRA was peer reviewed in December 2012 in accordance with NEI 05-04 Revision 2, the ASME/ANS PRA Standard (ASME/ANS RA Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" dated February 2009), and RG 1.200 Revision 2. All elements of the standard were reviewed, resulting in 77 Finding-level F&Os. These 77 F&Os and their resolutions were provided in the NFPA 805 LAR.

The NRC staff reviewed these F&Os and associated resolutions to determine if there was any potential impact on the SPRA model and the 10 CFR 50.54(f) submittal conclusions. The review was therefore limited to those F&Os that are related to accident sequence analysis (AS), success criteria (SC), systems analysis (SY), data analysis (DA), and LERF analysis (LE) (42 of the 77 F&Os). The review results of these 36 F&Os fall into one of three categories: (1) no potential impact to the SPRA model, (2) resolution of the disposition required a change for the NFPA 805 application and it is unclear if this change was incorporated in the IEPRA model used for the SPRA model, and (3) determined to be of minimal impact to the FPRA model but could impact the SPRA model and 10 CFR 50.54(f) results. Several of the findings were determined by the NRC staff to have a potential impact on the SPRA if not included in the IEPRA model that formed the base for the SPRA. In response to an audit question, the licensee explained that SPRA model is based on the current IEPRA model, which includes the disposition to all of the IEPRA peer review F&Os.

Deviation(s) or deficiency(ies) and Resolution: None Consequence(s): N/A	
The NRC staff concludes that: • The licensee's peer-review process meets the intent of the SPID guidance. • The licensee's peer-review process does not meet the intent of the SPID guidance but is acceptable on another justified basis.	Yes N/A

TOPIC 15: Documentation of the SPRA (SPID Section 6.8)

The NRC staff review of the SPRA's documentation as submitted finds an acceptable demonstration of its adequacy.	Yes
The documentation should include all of the items of specific information contained in the 50.54(f) letter as described in Section 6.8 of the SPID.	Yes
Notes from staff reviewer: The submittal appears to follow the recommendations of Section 6.8 of the SPID. Tables 2-1 and 2-2 of the submittal provide a cross-reference of information required by the 50.54(f) letter and specified in Section 6.8 of the SPID to the sections of the submittal where the information can be found. The level-of-detail of the information provided appears to be generally consistent with that specified in Section 6.8 of the SPID. It is noted, however, that not all the information identified in Section 6.8 of the SPID with regard to what was submitted for the Individual Plant Examination of External Events (IPEEE) program is included in the submittal (e.g., all functional/systemic event trees). However, the SPID only identifies this IPEEE information as guidance for consideration in the 50.54(f) response. All F&Os related to SPRA documentation (e.g., HLR_SHA-J, HLR-SPR-G, and HLR-SPR-F) have been closed by an NRC-accepted process (see Topic #14). Deviation(s) or deficiency(ies) and Resolution: None. Consequence(s): N/A	
The NRC staff concludes that: • The licensee's documentation meets the intent of the SPID guidance. The documentation requirements in the ASME/ANS Standard can be found in HLR-SHA-J, HLR-SFR-G, and HLR-SPR-F.	Yes
• The licensee's documentation does not meet the intent of the SPID guidance but is acceptable on another justified basis.	N/A

Topic 16: Review of Plant Modifications and Licensee Actions, If Any

The licensee: • identified modifications necessary to achieve seismic risk improvements • provided a schedule to implement such modifications (if any), consistent with the intent of the guidance • provided Regulatory Commitment to complete modifications • provided Regulatory Commitment to report completion of modifications.	Yes Yes Yes No
Plant will: • complete modifications by: Unit 1 – end of Refueling Outage No. 21 (March 2019); Unit 2 – end of Refueling Outage No. 21 (December 2019) • report completion of modifications by: Not Necessary (see "Notes from the Reviewer")	

Notes from the Reviewer:

Section 6.0 of the DCPP SPRA report states that "no seismic hazard vulnerabilities were identified." However, a deficiency in the installation of the 480V switchgear room ventilation ducts in both Units was identified during plant walk-downs. To address this deficiency, the licensee identified a modification in each Unit to modify the ducts and duct supports to accommodate differential movements between the turbine and auxiliary buildings. The licensee made a commitment to complete these modifications, with a completion date for Unit 1 of the end of Refueling Outage No. 21 (March 2019) and for Unit 2 of the end of Refueling Outage No. 21 (December 2019). The results of a sensitivity analysis provided in Section 5.7.13 of the submittal shows that not crediting this modification yields an increase in SCDF and SLERF of 4.2 percent and 1.8 percent, respectively. During its audit of the submittal, the NRC staff requested the licensee to discuss the seismic-specific considerations for selecting these modifications given the minimal reduction in the seismic risk from implementing these modifications. The licensee explained that during initial SPRA quantifications the ventilation room ducting was determined to be a "notable" risk contributor, but its contribution decreased following model refinements. Nevertheless, the licensee decided to maintain the corrected duct configuration in the updated SPRA model, and the modifications are planned to be implemented in the 2019 refueling outages. Because the SPRA model represents the modified ventilation duct configuration, and the impact of the modifications on seismic risk is small, the NRC staff concludes that the licensee need not provide a report to the NRC that the modifications have been completed for the staff's decisionmaking on the 50.54(f) submittal.

Refer to Enclosure 2 for detailed evaluation.

Deviation(s) or Deficiency(ies), and Resolution:

Refer to Enclosure 2 for detailed evaluation. Consequences: N/A	
The NRC staff concludes that: • The licensee identified plant modifications necessary to achieve the appropriate risk profile. • The licensee provided a schedule to implement the modifications (if any) with appropriate consideration of plant risk and outage scheduling.	Yes Yes

REFERENCES

ASME/ANS Addendum A, 2009: Standard ASME/ANS RA-Sa-2009, Addenda A to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2009

ASME/ANS Addendum B, 2013: Standard ASME/ANS RA-Sb-2013, Addenda B to ASME/ANS RA-S-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2013

EPRI-SPID, 2012: "Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," Electric Power Research Institute, EPRI report 1025287, November 2012, ADAMS Accession No. ML12333A170

NEI, 2012: NEI 12-13 "External Hazards PRA Peer Review Process Guidelines," Nuclear Energy Institute, August 2012, ADAMS Accession No. ML12240A027

NRC, 2012: "U.S. Nuclear Regulatory Commission Comments on NEI 12-13, 'External Hazards PRA Peer Review Process Guidelines' Dated August 2012," NRC letter to Nuclear Energy Institute, November 16, 2012, ADAMS Accession No. ML12321A280

AUDIT SUMMARY BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO DIABLO CANYON POWER PLANT, UNITS NOS. 1 AND 2 SUBMITTAL OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2018-JLD-0006)

BACKGROUND AND AUDIT BASIS

By letter dated March 12, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12053A340), the U.S. Nuclear Regulatory Commission (NRC) issued a request for information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.54(f) (hereafter referred to as the 50.54(f) letter). Enclosure 1 to the 50.54(f) letter requested that licensees reevaluate the seismic hazards for their sites using present-day methods and regulatory guidance used by the NRC staff when reviewing applications for early site permits and combined licenses.

By letter dated October 27, 2015 (ADAMS Accession No. ML15194A015), the NRC made a determination of which licensees were to perform: (1) a Seismic Probabilistic Risk Assessment (SPRA), (2) limited scope evaluations, or (3) no further actions based on a comparison of the reevaluated seismic hazard and the site's design-basis earthquake. (Note: Some plant-specific changes regarding whether an SPRA was needed or limited scope evaluations were needed at certain sites have occurred since the issuance of the October 27, 2015, letter.)

By letter dated July 6, 2017 (ADAMS Accession No. ML17177A446), the NRC issued a generic audit plan and entered into the audit process described in Office Instruction LIC-111, "Regulatory Audits," dated December 29, 2008 (ADAMS Accession No. ML082900195), to assist in the timely and efficient closure of activities associated with the letter issued pursuant to Title 10 CFR Part 50, Section 50.54(f). The list of applicable licensees in Enclosure 1 to the July 6, 2017, letter included Pacific Gas and Electric (PG&E, the licensee) as the licensee for Diablo Canyon Power Plant, Unit Nos. 1 and 2 (Diablo Canyon).

REGULATORY AUDIT SCOPE AND METHODOLOGY

The areas of focus for the regulatory audit are the information contained in the SPRA submittal and all associated and relevant supporting documentation used in the development of the SPRA submittal including, but not limited to, methodology, process information, calculations, computer models, etc.

AUDIT ACTIVITIES

The NRC staff developed questions to verify information in the licensee's submittal and to gain understanding of non-docketed information that supports the docketed SPRA report. The staff's clarification questions (ADAMS Accession No. ML18263A150) were sent to the licensee to support the audit.

The licensee provided clarifying information in the following areas:

- Discussion of enhancements or improvements that were considered to decrease human error probabilities
- Discussion of the seismic-specific consideration behind the selection of the 480 volt switchgear room ventilation duct modifications given the minimal impact on seismic risk.
- Discussion of how the same basic events, which were discretized by binning during the development of the licensee's SPRA, were then combined to develop representative importance measures.
- Discussion of the impact of the removal of the fragility cutoff at the high confidence of low probability of failure capacity on the SPRA results
- Discussion of the difference between sequence 6 which involves an earthquake between 1.75 and 2 g, and sequence 9 which involves an earthquake between 2 g and 2.25 g.
- Discussion of the approach used to perform the sensitivity study for "robust" components described in Section 4.4.1, "Structure, System, and Component Screening Approach," of Enclosure 1 of the licensee's SPRA report.
- The status of implementation of resolutions to peer review findings on the internal events PRA model.

The licensee's response to the questions aided in the staff's understanding of the Diablo Canyon SPRA docketed submittal. Following the review of the licensee's response and the supporting documents provided by the licensee on the eportal, the staff determined that no additional documentation or information was needed to supplement Diablo Canyon's docketed SPRA report.

DOCUMENTS AUDITED

- PWROG-17022-P, "Peer Review of the Diablo Canyon Units 1 & 2 Seismic Probabilistic Risk Assessment," September 2017
- 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," March 2018
- Document No. 128027-CD-01, "Seismic Fragility Evaluation of the Diablo Canyon Power Plant," 13 March 2018
- DCA 9000041965 Part No. 001, "Seismic Fragility Evaluation of the Diablo Canyon Power Plant," 13 March 2018

- DCA 9000042207 Part 002, "Seismic Walkdown of the Diablo Canyon Power Plant," 20 November 2017

- Calculation File No. F.6.1 Revision 4, "Subject: DCPP Seismic Equipment List Development," March 26, 2018

- Calculation File No. F.6.3 Revision 2, "Subject: DCPP Seismic PRA Human Reliability Analysis," March 26, 2018

- Calculation File No. F.6.2 Revision 3, "Subject: DCPP Seismic PRA Plant Logic Model," March 22, 2018

- Calculation File No. F.6.5 Revision 3, "Subject: DCPP Seismic PRA Quantification," March 26, 2018

OPEN ITEMS AND REQUEST FOR INFORMATION

There were no open items identified by the NRC staff that required proposed closure paths and there were no requests for information discussed or planned to be issued based on the audit.

DEVIATIONS FROM AUDIT PLAN

There were no deviations from the July 6, 2017, generic audit plan.

AUDIT CONCLUSION

The issuance of this document, containing the staff's review of the SPRA submittal, concludes the SPRA audit process for Diablo Canyon.

NRC Staff SPRA Submittal Detailed Screening Evaluation

Introduction

The Diablo Canyon Power Plant, Unit Nos. 1 and 2 (DCPP, Diablo Canyon) Seismic Probabilistic Risk Assessment (SPRA) report (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18120A201) indicates that the mean seismic core damage frequency (SCDF) is 2.8×10^{-5} per reactor-year (/rx-yr) and the mean seismic large early release frequency (SLERF) is 5.2×10^{-6} /rx-yr for both Units 1 and 2. The NRC staff compared these values against the guidance in NRC staff memorandum dated August 29, 2017, titled, "Guidance for Determination of Appropriate Regulatory Action Based on Seismic Probabilistic Risk Assessment Submittals in Response to Near Term Task Force Recommendation 2.1: Seismic" (ADAMS Accession No. ML17146A200; hereafter referred to as SPRA Screening Guidance), which establishes a process the NRC staff uses to develop a recommendation on whether the plant should move forward as a Group 1, 2, 3 plant.[1]

The SPRA Screening Guidance is based on NUREG/BR-0058, Revision 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," (ADAMS Accession No. ML042820192), NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," (ADAMS Accession No. ML050190193), and NUREG-1409, "Backfitting Guidelines," (ADAMS Accession No. ML032230247), as informed by Nuclear Energy Institute (NEI) 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document" (ADAMS Accession No. ML060530203). In order to determine the significance of proposed modifications in terms of safety improvement, NUREG/BR-0058 uses screening criteria based on the estimated reduction in core damage frequency, as well as the conditional probability of early containment failure or bypass. Per NUREG/BR-0058, the conditional probability of early containment failure or bypass is a measure of containment performance and the purpose of its inclusion in the screening criteria is to achieve a measure of balance between accident prevention and mitigation. The NUREG/BR-0058 uses a screening criteria of 0.1 or greater for conditional probability of early containment failure or bypass. In the context of the SPRA reviews, the staff guidance uses SCDF and SLERF as the screening criteria where SLERF is directly related to the conditional probability of early containment failure or bypass. Following NUREG/BR-0058, the threshold for the screening criterion in the staff guidance for SLERF is (1.0×10^{-6} /rx-yr), or 0.1 times the threshold for the screening criterion for SCDF (1.0×10^{-5} /rx-yr).

The NRC staff found that because the SCDF and SLERF for DCPP were above the initial screening values of 1.0×10^{-5}/rx-yr and 1.0×10^{-6}/rx-yr, respectively, a detailed screening following the SPRA Screening Guidance was performed. The detailed screening shows that DCPP should be considered a Group 1 plant because:

- Sufficient reductions in SCDF and/or SLERF cannot be achieved by potential modifications considered in this evaluation to constitute substantial safety improvements based upon importance measures, available information, and engineering judgement;

[1] The groups are defined as follows: regulatory action not warranted (termed Group 1), regulatory action should be considered (termed Group 2), and more thorough analysis is needed to determine if regulatory action should be considered (termed Group 3).

- Additional consideration of containment performance, as described in NUREG/BR-0058, does not identify a modification that would result in a substantial safety improvement; and

- The staff did not identify any potential modifications that would be appropriate to consider necessary for adequate protection or compliance with existing requirements.

As such, additional refined screening, or further evaluation, was not required.

Detailed Screening

The licensee, in performing its seismic analysis in response to the Near-Term Task Force Recommendation 2.1, and the NRC in conducting its review, did not identify concerns that would require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. In addition, there were no issues identified as non-compliances with the DCPP licenses, or with the rules and orders of the Commission. For these reasons, the licensee and the staff did not identify a potential modification necessary for adequate protection or compliance with existing regulations.

The detailed screening uses information provided in the DCPP SPRA report, particularly the importance measures, SCDF, and SLERF, as well as other information described below, to establish threshold and target values to identify potential cost-justified substantial safety improvements. The detailed screening process makes several simplifying assumptions, similar to a Phase 1 SAMA analysis (NEI 05-01, ADAMS Accession No. ML060530203) used for license renewal applications. The detailed screening process uses risk importance values as defined in NUREG/CR-3385, "Measures of Risk Importance and Their Applications" (ADAMS Accession No. ML071690031). The NUREG/CR-3385 states that the risk reduction worth (RRW) importance value is useful for prioritizing feature improvements that can most reduce the risk. The DCPP SPRA report provides Fussell-Vesely (F-V) importance measures, which were converted to RRW values by the NRC staff for this screening evaluation using an established mathematical relationship (included in the SPRA Screening Guidance). Data used to develop the maximum averted cost-risk (MACR) for the severe accident mitigation alternative (SAMA) analysis provided in the *Diablo Canyon Power Plant - License Renewal Application, Applicant's Environmental Report, Operating License Renewal Stage*, dated November 23, 2009 (ADAMS Accession No. ML093340086; hereafter DCPP SAMA), was used to calculate the RRW threshold.[2] For this analysis, the NRC staff determined the RRW threshold from the SCDF-based MACR to be 1.149 for both Units. The MACR calculation includes estimation of offsite exposures and offsite property damage, which captures the impact of SLERF. Therefore, separate SLERF-based MACR calculations were not performed. The MACR calculation was based on the remaining operating life of DCPP of 7 years. The licensee requested withdrawal of its license renewal application for DCPP by letter dated March 7, 2018 (ADAMS Accession No. ML18066A937). The NRC granted the licensee's request in letter dated April 23, 2018 (ML18093A117). The staff further notes that based on the RRW values and the detailed screening described below, the staff's decision remains unchanged even if an extended operating life of 20 years is considered.

Section 5 of the DCPP SPRA report included tables listing and describing the structures,

[2] The MACR data from the 2015 updated DCPP SAMA analysis was not utilized because it incorporated the risk for multiple hazards (i.e., internal events, internal flooding, fire, and seismic) and insufficient information was provided to develop a seismic-specific MACR.

systems, and components (SSCs) that are the most significant contributors to SCDF and SLERF. The descriptions of the significant contributors included the corresponding F-V importance measures. Tables 1 and 2 of this Enclosure list the SCDF and SLERF contributors, respectively, from the submittal. These tables provide the following information by column: (1) Description of the component, (2) Failure mode of the component, if applicable, (3) RRW, and (4) maximum SCDF or SLERF reduction (MCR or MLR, respectively) from eliminating that failure. There were no single SCDF contributors that exceeded the RRW threshold for SCDF, while a single SPRA model element or contributor exceeded the RRW threshold for SLERF. This element was seismically-induced failure of the containment exterior shell structure that results in core damage, which has an SLERF RRW of 1.789 and an SLERF contribution of 2.3×10^{-6}/rx-yr. According to Section 5 of the DCPP SPRA report, this containment failure is also the contributor for the top four accident sequences (for four different seismic acceleration intervals) for both SCDF and SLERF. The NRC staff experience from SAMA analyses is that the implementation cost of modifications to the containment building sufficient to eliminate the seismic risk from a seismically-induced containment failure or to substantially reduce the probability of containment failure substantially exceed the calculated MACR for this detailed screening.

Other than seismically-induced failure of the containment exterior shell structure, none of the individual contributors to either SCDF or SLERF exceed the calculated RRW threshold for this detailed screening.

Nevertheless, during its audit of the submittal, the NRC staff requested the licensee to discuss any enhancements or improvements (e.g., frequency of training) that were considered to decrease the human error probabilities (HEPs) associated with the two diverse and flexible coping strategies (FLEX) recovery actions that are credited to reduce seismic risk for seismic-induced extended loss of AC power (ELAP) scenarios, which are shown in Table 1. The licensee explained that the operators are trained on those FLEX procedures once a year, and that no additional improvements to these procedures have been identified. The licensee also explained that further procedural improvements or additional operator training are not expected to significantly improve seismic risk because (1) the development of the HEPs accounted for delays caused by the aftereffects of the earthquake, and (2) a sensitivity analysis assuming a 25 percent reduction in the FLEX HEPs decreases SCDF by 3.5 percent and SLERF by less than one percent.

The NRC staff notes that the FLEX actions cited above utilized installed equipment and were performed in seismic Category I buildings. The licensee's SPRA model does not credit those actions beyond acceleration of 3g. The NRC staff also observes that, based on these results, complete elimination of the failure of these FLEX actions would result in a reduction in SCDF and SLERF of about 14 percent and four percent, respectively. This corresponds to a maximum reduction in the mean SCDF and SLERF of about 3.9×10^{-6}/rx-yr and 2.1×10^{-7}/rx-yr, respectively which are below the initial screening values of 1.0×10^{-5}/rx-yr and 1.0×10^{-6}/rx-yr, respectively. Therefore, the NRC staff concludes that there are no potential modifications, including procedure improvements that would be considered substantial safety improvements.

The NRC staff also considered combinations of basic events in accordance with the SPRA Screening Guidance. It is not the intent of that aspect of the guidance to aggregate several disparate basic events that individually have RRW values close to the threshold. The total SCDF of the SPRA model elements identified in Table 1 is about 1.7×10^{-5}/rx-yr. A review of these model elements reveals that any modification or set of modifications to achieve a SCDF reduction of at least 1.0×10^{-5}/rx-yr will have to mitigate or prevent multiple failure types (e.g.,

seismically-induced failures, random failures[3], and failure of operator actions) and failure modes (e.g., seismically-induced structural failures of multiple SSCs and seismically-induced functional failures of multiple SSCs). One potential such modification is to provide redundant and diverse sources of seismically-qualified auxiliary feedwater and AC power. The NRC staff experience gained from reviewing SAMA analyses for various license renewal applications, including information related to SAMA analysis related to seismic risk as summarized in the SPRA Screening Guidance, is that the cost of eliminating the seismic risk from these types of plant improvements would substantially exceed the calculated MACR for this detailed screening.

A review of the SLERF model elements, which included random failures, and human failure events, reveals that SLERF is dominated by seismically-induced failure of SSCs and therefore, any modification or set of modifications to achieve a SLERF reduction of at least 1.0×10^{-6}/rx-yr will have to mitigate or prevent multiple unrelated seismically-induced SSC failure modes. These seismically-induced SSC failures include those that result in containment bypass, which leads directly to SLERF. One potential modification that impacts such failures is to route the discharges from all containment penetrations through a structure where a water spray would condense the steam and remove most of the fission products. The NRC staff experience from SAMA analyses is that the cost of eliminating the seismic risk from these types of plant improvements would substantially exceed the calculated MACR for this detailed screening.

Based on the analysis described above, the NRC staff concludes that no modifications are warranted in accordance with Title 10 of the Code of Federal Regulations Section 50.109 (10 CFR 50.109) to reduce SCDF and SLERF because a potential cost-justified substantial safety improvement was not identified.

In accordance with Section 3.3.2 of NUREG/BR-0058, Rev. 4, the NRC staff further evaluated DCPP accident sequences impacting the conditional probability of early containment failure or bypass (CPCFB) for seismic events to determine if any substantial safety improvements would reduce the SCDF and related SLERF of those sequences. The dominate LERF sequences are due to seismically-induced failure of the containment exterior shell structure that results in core damage and seismically-induced failure of the steam generators that results in containment bypass. Each of these dominant LERF sequences were already evaluated, as described above. Furthermore, these seismic accident sequences corresponded to peak ground acceleration greater than 3.0g and were modeled by the licensee as guaranteed failure for all human failure events. Given the large uncertainties in human reliability analysis and the conditions of the site during these large beyond design bases earthquakes, it was not feasible to request that the licensee make modifications to reduce the contribution of these human actions to the CPCFB.

Based on the available information and engineering judgement, the NRC staff concluded that there were no further potential improvements to containment performance that would rise to the level of a substantial safety improvement or would warrant further regulatory analysis. Additionally, the NRC staff reviewed the results of the IPEEE and SAMA analyses previously completed for DCPP to identify additional substantial safety improvements that would be cost justified. No other potential improvements were found based on that review.

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[3] The licensee provided information on random failures and operator actions that are not due to the seismic event in its submittal. The staff included this information as an aid to help identify potential modifications that could reduce the overall SCDF and/or SLERF.

Conclusion

Based on the analysis of the submittal and supplemental information, the NRC staff concludes that no modifications are warranted under 10 CFR 50.109 because:

- The staff did not identify a potential modification necessary for adequate protection or compliance with existing regulations;
- no potential cost-justified substantial safety improvement was identified based on the estimated achievable reduction in SCDF and/or SLERF; and
- additional consideration of containment performance, as described in NUREG/BR-0058 and assessed via SLERF, did not identify a modification that would result in a substantial safety improvement.

Table 1. Importance Analysis Results of Top Contributors to SCDF

Description	Failure Mode	RRW	MCR (/rx-yr)
Seismically-failed SSCs			
CST/Refueling water storage tank (RWST)	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	1.045	1.22×10^{-6}
Main control board vertical board	Functional	1.031	8.54×10^{-7}
Firewater storage tank (FWST)	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	1.029	8.04E-07
Auxiliary building (common structure)	Structural - Diagonal Shear Failure	1.029	7.92×10^{-7}
Process control and protection system (Eagle 21)	Functional	1.029	7.90×10^{-7}
EDG room interior NLBW (between EDG 1-1 and 1-2)	Structural - Out-of-plane flexural failure	1.023	6.40×10^{-7}
Turbine building Unit 1 portion (shear wall failure mode)	Structural - Excessive story drifts in EW shear walls	1.022	6.20×10^{-7}
Nuclear instrumentation regulating transformer	Functional	1.017	4.62×10^{-7}
Firewater piping in auxiliary building	Structural - leakage	1.016	4.46×10^{-7}
East-West concrete block wall between 4kV cable spreading room Bus G and H	Structural - Out-of-plane flexural failure	1.013	3.50×10^{-7}
ASW piping (buried)	Structural - pull out from dresser coupling	1.012	3.21×10^{-7}
Containment exterior structure	Structural - Tangential shear failure	1.011	3.05×10^{-7}
EDG engine (includes jacket water and radiator)	Structural - shear failure of skid end seismic stays	1.009	2.57×10^{-7}
EDG Generator	Structural - shear failure of skid end seismic stays	1.009	2.57×10^{-7}
TDAFW MOVs	Structural - Binding of valve stem	1.007	1.88×10^{-7}

Table 1. Importance Analysis Results of Top Contributors to SCDF

Description	Failure Mode	RRW	MCR (/rx-yr)
Randomly-failed SSCs			
Common cause failure of all three 125VDC batteries	NA	1.074	1.93×10^{-6}
Common cause failure of three vital 4KV AC circuit breakers	NA	1.018	4.85×10^{-7}
RCP shutdown seal fails to actuate and initially seal	NA	1.007	1.99×10^{-7}
Diesel generator 1-1 fails to run more than 1 hour	NA	1.007	1.86×10^{-7}
TDAFW Pump 11 fails to start	NA	1.006	1.77×10^{-7}
Diesel generator 1-3 fails to run more than 1 hour	NA	1.006	1.68×10^{-7}
Common cause triple failure of diesel generator fuel oil day tank LCVs	NA	1.006	1.64×10^{-7}
Human Failure Events			
FLEX Action: Operator Fails to Shed Battery Loads on Extended Loss of AC Power-SEISMIC-HIGH	NA	1.125	3.13×10^{-6}
FLEX Action: Operators fail to prevent Steam Generator overfill during a loss of all power (AC & DC)-SEIS-HIGH	NA	1.025	6.88×10^{-7}
Operators fail to trip RCPs from the Control Room on Loss of CCW (13 MIN)	NA	1.018	4.88×10^{-7}
FLEX Action: Operator Fails to Shed Battery Loads on Extended Loss of AC Power-SEISMIC-LOW	NA	1.006	1.56×10^{-7}
Operators fail to trip RCPs locally on Loss of CCW (30 MIN)- SEIS-LOW	NA	1.006	1.55×10^{-7}

Table 2. Importance Analysis Results of Top Contributors to SLERF

Description	Failure Mode	RRW	MLR (/rx-yr)
Seismically-failed SSCs			
Containment exterior structure	Structural - Tangential shear failure	1.789	2.30×10^{-6}
Steam generators	Structural - Upper Support Ring Band in bending	1.084	4.07×10^{-7}
Containment fan cooler (piping containment bypass)	Structural - Housing column footplate weld Failure	1.072	3.52×10^{-7}
120V AC instrument breaker panel	Functional	1.044	2.22×10^{-7}
SSPS input relay panel	Functional	1.042	2.12×10^{-7}
SSPS test cabinet	Functional	1.021	1.05×10^{-7}
Firewater piping in auxiliary building	Structural - leakage	1.020	1.01×10^{-7}
Process control and protection system (Eagle 21)	Functional	1.019	9.81×10^{-7}
Main control Board vertical board	Functional	1.016	8.35×10^{-8}
CST/RWST	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	1.015	7.78×10^{-8}
Containment mechanical penetration	Structural	1.015	7.73×10^{-8}
ASW piping (buried)	Structural - pull out from dresser coupling	1.013	6.79×10^{-8}
Turbine building Unit 1 portion (shear wall failure mode)	Structural - Excessive story drifts in EW shear walls	1.012	6.00×10^{-8}
Nuclear instrumentation regulating transformer	Functional	1.011	5.79×10^{-8}
FWST	Structural - Tangential shear failure of the tank at elevation 125 ft-6 in.	1.009	4.67×10^{-8}

Table 2. Importance Analysis Results of Top Contributors to SLERF

Description	Failure Mode	RRW	MLR (/rx-yr)
Randomly-failed SSCs			
Common cause failure of all three 125VDC batteries	NA	1.019	9.71×10^{-8}
Human Failure Events			
FLEX Action: Operator Fails to Shed Battery Loads on Extended Loss of AC Power-SEISMIC-HIGH	NA	1.026	1.32×10^{-7}
Operators fail to trip RCPs locally on Loss of CCW (30 MIN)- SEIS-LOW	NA	1.007	3.69×10^{-8}
Operators fail to manually actuate ESF equipment with SSPS failures - SEISMIC-HIGH	NA	1.007	3.54×10^{-8}

SUBJECT: DIABLO CANYON POWER PLANT, UNIT NOS. 1 AND 2 – STAFF REVIEW OF SEISMIC PROBABILISTIC RISK ASSESSMENT ASSOCIATED WITH REEVALUATED SEISMIC HAZARD IMPLEMENTATION OF THE NEAR-TERM TASK FORCE RECOMMENDATION 2.1: SEISMIC (EPID NO. L-2018-JLD-0006) DATED JANUARY 22, 2019

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