Pacific Gas and Electric Company®

Dennis B. Petersen
Station Director

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

805.545.4022
Dennis.Petersen@pge.com

PG&E Letter DCL-24-004 10 CFR 50.90

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

Diablo Canyon Units 1 and 2
Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Supplement to License Amendment Request 23-01
Revision to Technical Specifications to Adopt Risk-Informed Completion Times
TSTF-505, Revision 2, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b"

Reference 1: PG&E Letter DCL-23-054, "License Amendment Request 23-01 Revision to Technical Specifications to Adopt Risk-Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" dated July 13, 2023 (ADAMS Accession No. ML23194A228)

2: NRC Letter Diablo Canyon Nuclear Power Plant, Units 1 AND 2 – Regulatory Audit Plan in Support of License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times (EPID L-2023-LLA-0100), dated September 21, 2023

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) submitted Reference 1 that requested approval of a proposed amendment to the Technical Specifications for Diablo Canyon Power Plant Units 1 and 2 to implement risk-informed Completion Times. In Reference 2, the staff informed PG&E of a virtual regulatory audit to support staff review of the Reference 1 request. In support of the audit, the staff provided questions and PG&E provided responses that were discussed on December 11-12, 2023. The staff requested PG&E submit the enclosed response to the staff audit questions to support the staff licensing decision.

The response in the Enclosure does not impact the significant hazards evaluation or environment evaluation contained in Reference 1.

PG&E makes no regulatory commitment (as defined by NEI 99-04) in this letter.

Pursuant to 10 CFR 50.91, PG&E is sending a copy of this letter to the California Department of Public Health.

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

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

Sincerely,

Dennis B. Petersen
Station Director

Executed on: 15 Jan 2024
Date

kjse/ SAPN 51178920-02
Enclosure
cc: Diablo Distribution
cc/enc: Anthony Chu, Branch Chief, California Dept of Public Health
Mahdi O. Hayes, NRC Senior Resident Inspector
Samson S. Lee, NRR Project Manager
John D. Monninger, NRC Region IV Deputy Administrator

Supplement to License Amendment Request 23-01
License Amendment Request 23-01 Revision to Technical Specifications to Adopt Risk-Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b'

Response to Regulatory Audit Questions in Support of License Amendment Request to Revise Technical Specifications to Adopt Risk-Informed Completion Times
Pacific Gas and Electric Company

Diablo Canyon Nuclear Power Plant, Units 1 and 2

Docket Nos. 50-275 and 50-323

Pursuant to 10 CFR 50.90, Pacific Gas and Electric Company (PG&E) submitted PG&E Letter DCL-23-054, "License Amendment Request 23-01 Revision to Technical Specifications to Adopt Risk-Informed Completion Times TSTF-505, Revision 2, 'Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b,'" dated July 13, 2023 (ADAMS Accession No. ML23194A228) that requested approval of a proposed amendment to the Technical Specifications for Diablo Canyon Power Plant Units 1 and 2 (DCPP) to implement risk-informed Completion Times. In support of the staff regulatory audit, the staff provided questions and PG&E provided responses that were discussed on December 11-12, 2023. The staff requested PG&E submit the response to the staff regulatory audit questions to support the staff licensing decision. The response to the audit questions is contained in the next page in this Enclosure.

<u>DRA, Probabilistic Risk Assessment (PRA) Licensing Branch A (APLA)</u>

Audit Question APLA-01 *(Success criteria)*

The NRC staff's final safety evaluation (ML071200238) to Nuclear Energy Institute (NEI) 06-09 (ML122860402) specifies that the LAR should identify the technical specification limiting conditions for operation (LCOs) and action statements for which risk-informed completion times (RICTs) are proposed. The LAR should compare the functions of structures, systems, and components (SSCs) subject to those technical specifications with functions of those SSCs modeled in the probabilistic risk assessment (PRA). For functions that are modeled, the LAR should justify that the scope of the PRA model is consistent with the licensing basis assumptions. The LAR should address any differences and explain how they will be handled, for example, by programmatic restrictions.

The safety evaluation for NEI 06-09 also states that when the licensee determines that risk sources may be excluded from PRA models because they are not significant to the calculation of risk, the LAR should discuss conservative or bounding analysis to be applied to the calculation of RICT when those sources are not addressed in the PRA models.

Table E1-1 in Enclosure 1 of the LAR identifies each LCO proposed for inclusion in the RICT program. For each LCO, the table identifies whether the associated SSCs are modeled in the PRA. For certain LCOs, the table explains that the associated SSCs are not modeled in the PRAs but will be conservatively represented using a surrogate event. The description in the LAR did not allow the NRC staff to conclude that the modeling of surrogate events bounds the risk or conservatively represents the identified SSCs.

a. *For TS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," the LAR states that the PRA success criteria are in some cases "more restrictive when the PORVs are credited to mitigate some beyond-design-basis scenarios." Clarify and justify the PRA success criteria used, including the scenarios where success criteria differ from the design basis.*

PG&E Response:

Note 9 has been added to Table E1-1 stating:

The PORV success criteria for beyond design basis scenarios corresponds to loss of steam generator cooling events where Bleed and Feed cooling is initiated. Bleed and Feed cooling through the PORVs is successful if 2 out of 3 PORVs open. PORV PCV-474 is not safety related and does not have a backup air accumulator, thus it is only credited for those initiating events where instrument air is credited.

b. *For TS 3.5.2, "ECCS [emergency core cooling system] – Operating," the LAR states that the PRA does not credit mitigation for main steamline break (MSLB) events. It also states that the PRA success criteria are based on plant-specific analyses. Justify the proposed modeling does not have an impact on RICT estimates.*

PG&E Response:

The Table E1-1 has been updated to include the following information:

Added to the "PRA success criteria" column:

> "(c) 1 of 4 ECCS CH pumps or SI pumps for SGTR [Steam Generator Tube Rupture] and MSLB (Boration injection when reactor trip function fails)
>
> Added to the "Disposition" column:
>
> (3) only requiring boration mitigation for MSLB events when the reactor trip function fails;"

The design function of the centrifugal charging subsystem of the ECCS is to supply borated water to the reactor core following a main steam line break. The limiting design conditions occur when the negative moderator temperature coefficient is highly negative, such as at the end of each cycle. In the PRA model ECCS injection from the charging subsystem is modeled for main steamline breaks when the reactor fails to trip. There will be an ECCS RICT contribution from main steam line break scenarios where the reactor trip fails.

c. *For TS 3.6.6, "Containment Spray [CS] and Cooling Systems," Note 4 to LAR Table E1-1 states that neither the CS system nor the containment fan cooling units (CFCUs) are credited in the fire PRA. The NRC staff observes that choosing not to model a system in the PRA may produce a nonconservative calculation of RICT. Justify the proposed modeling does not have an impact on the RICT estimates.*

PG&E Response:

Delta Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) used to assess the RICT requires a baseline CDF and LERF number to be subtracted from the application Configuration Risk Management Program (CRMP) tool CDF and LERF. Baseline CDF and LERF are calculated separately in the Riskman PRA software by crediting and guaranteeing success for those systems guaranteed failed in the CRMP application model. This baseline CDF and LERF is then input in the CRMP software as a single CDF and single LERF value that is the same for all RICT calculations. These success impacts used to calculate baseline CDF and LERF are not included in the CRMP tool model.

The baseline CDF and LERF is subtracted from the CRMP tool CDF and LERF to calculate the RICT. A baseline CDF and LERF when the containment spray system and

CFCUs are successful is conservative and results in a larger delta CDF/LERF for all configurations and thus a shorter RICT.

d. *For TS 3.8.1, "AC [Alternating Current] Sources – Operating," Note 8 to LAR Table E1-1 states that the 500 kV offsite circuits are only credited for the mitigation of internal events. Discuss the role of the 500kV system and justify why crediting the system only in the mitigation of internal events results in an acceptable RICT estimate.*

PG&E Response:

The offsite power 500-kV [kilo Volt] system requires a manual action outside the control room to transfer power, as opposed to the 230-kV system which automatically transfers on a loss of power signal. Because the manual action has a higher failure probability than that of an automatic action, removing the 500-kV system from service has less of a risk impact than the 230-kV system.

Delta CDF and LERF used to assess the RICT requires a baseline CDF and LERF number to be subtracted from the application CRMP tool CDF and LERF. Baseline CDF and LERF are calculated separately in the Riskman PRA software by crediting and guaranteeing success for those systems guaranteed failed in the CRMP application model. This baseline CDF and LERF is then input in the CRMP software as a single CDF and single LERF value that is the same for all RICT calculations. These success impacts used to calculate baseline CDF and LERF are not included in the CRMP tool model.

Initiating events (including those from the fire PRA model) that do not credit the 500-kV offsite power system have the 500-kV offsite power system guaranteed successful in the single baseline CDF and LERF used to calculate all RICTs. The baseline CDF and LERF is subtracted from the CRMP tool CDF and LERF to calculate the RICT. A baseline CDF and LERF with the 500-kV system successful is conservative and results in a larger delta CDF/LERF for all configurations and thus a shorter RICT.

Audit Question APLA-02 *(Process for reviewing key assumptions and sources of uncertainty)*

Regulatory Guide (RG) 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession No. ML17317A256), describes an approach that is acceptable to the NRC staff for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights. It provides *general guidance concerning analysis of the risk associated with the proposed changes in plant design and operation. Section C.4. “Documentation to Support a Regulatory Submittal,” of RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (ADAMS Accession No. ML090410014), provides guidance regarding documentation of the acceptability of the PRA to support a regulatory submittal.*

Further, Section 2.5 of RG 1.174 states that the impact of PRA uncertainties should be considered, including uncertainties that are explicitly accounted for in the results and those that are not, and cites NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making” (ADAMS Accession No. ML17062A466), provides acceptable guidance for the treatment of uncertainties in risk-informed decision-making.

NUREG-1855 describes how the impact of PRA uncertainties should be assessed and documented. It states, "Additional qualitative screening criteria may be identified as applicable for specific applications. The bases for any criteria used to qualitatively eliminate missing scope and level-of-detail items from a PRA must be documented," as well as, "At a minimum, assumptions made in lieu of data, operational experience or design detail should be well documented with the basis for the assumptions clearly explained."

Enclosure 9 of LAR describes the process used for reviewing the PRA assumptions and sources of uncertainty. The LAR explained that the list of assumptions and sources of uncertainty were reviewed to identify those that would be significant for the evaluation of configuration-specific changes in risk in the RICT Program.

The NRC staff reviewed the Diablo Canyon documents provided on the audit portal and was unable to identify a document demonstrating this review of PRA assumptions and sources of uncertainty for impact on the RICT Program.

Confirm that the review of plant-specific PRA assumptions and sources of uncertainties was documented for use in the RICT program.

PG&E Response:

Disposition of Key Assumptions and Sources of Uncertainty

Formal documentation of the disposition of each assumption and source of uncertainty has not been completed. As part of implementation of TSTF-505 at Diablo Canyon, PG&E will update its PRA documentation to include the justification for each baseline PRA model assumption and source of uncertainty relevant to the TSTF-505 application to ensure that future updates consider possible impacts that could affect the RICT Program.

The identification of PRA model assumptions and sources of uncertainty is described in the LAR Enclosure 9, and conforms to the applicable requirements of the relevant PRA standard for each PRA model. This process and results were reviewed and found acceptable in the peer reviews and applicable F&O closure reviews.

The screening process applied to the base model list of assumptions and sources of uncertainty is also described in the LAR Enclosure 9. Each assumption was assessed qualitatively to identify if the underlying assumption or source of uncertainty would have any potential significant adverse impact on the calculations of configuration risk to support the RICT Program. The calculation of configuration risk differs from the calculation of baseline risk in the following areas:

- Equipment unavailabilities (i.e., maintenance, surveillance testing, etc.) are set to zero or one, based on actual configurations.
- Actual plant alignments for equipment in operation are set, rather than assuming an alignment or a split fraction based on plant operation over time.
- The numeric results are per reactor-critical-year, without reducing the resulting CDF and LERF by the plant availability factor.
- Equipment recovery probabilities used in the baseline PRA models are not applied for unavailable equipment in the CRMP model.

The screening process considered these differences between configuration risk assessments and baseline risk assessments to identify those assumptions and sources of uncertainty that would be "key" for the TSTF-505 application.

In addition, recent approved TSTF-505 LARs for similar Westinghouse 4-loop NSSS design plants were reviewed, including RAI responses and the NRC safety evaluations, to identify any generic assumptions or sources of uncertainty found to be "key" for TSTF-505, that might therefore also be applicable to Diablo Canyon. No additional items were identified. Therefore, the Diablo Canyon identification of key assumptions and sources of uncertainty are judged to be complete and adequate to support implementation of TSTF-505.

Audit Question APLA-03 *(Key assumptions and sources of uncertainty)*

The NRC staff's safety evaluation to NEI 06-09 specifies that the LAR should provide a discussion of how the key assumptions and sources of uncertainty were identified. Table E9-1 in Enclosure 9 of the LAR discusses and presents the disposition for each identified key assumption or source of uncertainty.

a. Dual unit trips are not considered in the single-unit model (except for seismic events). The LAR further identifies that "this approach is nonconservative because *the plant equipment credited may be required by the second unit and be unavailable for crosstie." The LAR disposition to this uncertainty item states that shared systems and equipment between the units will be identified in procedures for RICT Program implementation so that consideration of additional risk management actions will be made.*

Identify the shared systems and equipment.

i. Explain how the RICT program procedures will capture the unavailability of shared SSCs.

PG&E Response:

Shared systems include Auxiliary Saltwater (ASW) pumps, electrical support for ASW (Diesel Generators, 4KV buses, direct current (DC) power), and Diesel Fuel Oil.

The RICT program will capture the actual unavailability of shared components by including their impacts in the CRMP tool. For example, if ASW pump 2-2 is out of service, this will be modeled in the CRMP tool and any risk impacts to Unit 1 will be assessed.

ii. Describe the process that will be used (for shared systems) for identifying and selecting additional risk management actions.

PG&E Response:

The process used to identify risk management actions for shared system components would not be any different for shared components than the process described in Enclosure 12 of the LAR.

- Risk insights from the CRMP tool will be used to identify SSCs that are most important for a CDF and/or a LERF perspective.
- Aside from actions that protect important OPERABLE components, the RMA process will also consider actions that:
  - Increase awareness and control of activities,
  - Reduce duration of maintenance activities, and
  - Reschedule other maintenance activities.

b. Charging and safety injection (SI) pumps are credited for inventory makeup for a medium loss of coolant accident, and it is assumed that two of the four high-pressure injection pumps are required for success. The LAR states that "this was conservatively modeled as 1 out of 2 charging pumps and 2 out of 2 SI pumps." The LAR further states that this is modeled conservatively, and the model is further *adjusted by an assumed recovery factor to offset this conservatism when all support for the function is available.*

However, the LAR states that this assumption is not conservative "whenever a charging pump is unavailable and the safety injection system fails.... Accordingly, the emergency core cooling system charging pump recovery factor will not be credited in the RICT Program whenever an emergency core cooling system charging pump is made unavailable."

Explain how the proposed model adjustment will be handled in the configuration risk management program (CRMP) tool when pumps become inoperable and justify this treatment for the RICT estimates.

PG&E Response:

Whenever an ECCS charging pump is made inoperable, the CRMP tool will also include a model impact that eliminates recovery credit for additional pumps. This model adjustment removes the small non-conservatism present in the model.

c. *A 6-hour mission time was assumed for the emergency diesel generators (EDGs) and the fuel oil transfer pumps. The LAR states that this assumption "does not have a significant impact on the baseline PRA model." It further states, "Whenever the 230 kV offsite power system is unavailable and cannot reasonably be recovered within 6 hours, the 6-hour mission time is nonconservative."*

   The LAR therefore proposed "the 24-hour mission time will be applied to the EDGs and fuel oil transfer pumps in the RICT Program whenever the offsite power 230 kV system is made unavailable."
   Explain how the proposed adjustment will be addressed in the RICT program.

PG&E Response:

If 230-kV offsite power is unavailable during a RICT, the assumption will be that offsite power recovery is not possible and the convolution power recovery model that is normally used will be removed. This change will is automatically made by modifying the diesel generator fail to run basic events to use a 24-hour mission time when the operator selects the 230-kV offsite power system to be taken out of service in the CRMP software tool.

d. *Vacuum breakers cannot fail in a manner to impact the Auxiliary Salt Water (ASW) function within the 24-hour mission time. The LAR explains that "There are two vacuum relief valves per ASW header." The LAR further states that the "RICT Program will assume inoperability of the ASW train if one or more vacuum breakers are nonfunctional."*

   Explain how this proposed adjustment will be handled in the CRMP tool.

PG&E Response:

One train of ASW will be declared inoperable when one of its two vacuum breakers is unavailable. The CRMP software tool will map the activity code for an ASW vacuum breaker to the applicable ASW train.

e. *RICT guidance will require ASW to be declared inoperable when vacuum breaker is unavailable. The LAR states that certain systems and components are always assumed failed in the fire PRA and the seismic PRA models (and that they are assumed always successful in the baseline PRA model). The LAR concludes that the "resulting RICT is conservatively bounded."*

   Describe how this is modeled and identify the systems treated in this way.

   i. *Describe the systems affected.*

PG&E Response:

Systems and components assumed failed in the Seismic and Fire PRA models include:

- 500-kV offsite power system
- Non-vital power systems
- Unit bus crosstie breaker
- Opposite unit startup power crosstie
- Balance of plant systems including main feedwater, condensate, instrument air, circulating water, service cooling water
- ATWS Mitigating System Actuation Circuitry (AMSAC) System
- Containment Spray System
- Containment Fan Cooler Units
- Makeup from the Spent Fuel Pool System

   ii. *Describe what is meant by stating that these systems are "assumed always successful in the baseline PRA model." Clarify how this treatment applies to the fire and seismic PRAs.*

PG&E Response:

Delta CDF and LERF used to assess the RICT requires a baseline CDF and LERF number to be subtracted from the application CRMP tool CDF and LERF. Baseline CDF and LERF are calculated separately in the Riskman PRA software by crediting and guaranteeing success for those systems guaranteed failed in the CRMP application model. This baseline CDF and LERF is then input in the CRMP software as a single CDF and single LERF value that is the same for all RICT calculations. These success impacts used to calculate baseline CDF and LERF are not included in the CRMP tool model.

iii. *Provide further justification to demonstrate that the RICT estimates are conservative.*

PG&E Response:

Initiating events (including those from the seismic and fire PRA model) that do not credit systems have these systems guaranteed successful in the single baseline CDF and LERF used to calculate all RICTs. The baseline CDF and LERF is subtracted from the CRMP tool CDF and LERF to calculate the RICT. A baseline CDF and LERF with these systems successful is conservative and results in a larger delta CDF/LERF for all configurations and thus a shorter RICT, because the baseline CDF and LERF has no contribution from the system failing.

f. *The LAR describes a model simplification for the auxiliary feedwater (AFW) system that applies to sequences involving depressurization of multiple steam generators. The LAR states that "Pump runout protection is only modeled for Auxiliary Feedwater Pump 1-2 and is always successful for pump 1-3."*

   Further justify (1) the effect of this simplification on other sequences (e.g., those involving AFW pumps or main steam isolation valves), (2) the expression of risk (expressed here as a fraction of core damage frequency (CDF), and (3) impact on the RICT estimate.

PG&E Response:

For more than one steam generator to depressurize, multiple main steam isolation valves (MSIVs) must fail concurrently or initiating events that could cause the depressurization of more than one steam generator (i.e., main steam line break down stream of the MSIVs) need to occur. The runout protection failure probability is approximately 1.60E-03, which is dominated by the miscalibration of the pressure transmitter. The main steam line break initiating event frequency is 7.76E-04/yr while the failure probability of two main steam lines failing to close is 2.41E-06. The contribution of runout failing a motor driven AFW pump to run is approximately 1.60E-03 * (7.76E-04+2.41E-06) = 1.25E-06. The fail to run probability of an AFW pump in the PRA model is 4.24E-04, which more than 2 orders of magnitude higher than the AFW fail to run probability due to runout protection failure.

For the case where one MSIV is unavailable then only one other MSIV needs to fail to isolate for the motor driven AFW pump runout protection system to be required. Failure probability of one main steam line failing to close is 2.31E-04. The contribution of runout failing a motor driven AFW pump to run with one MSIV unavailable is approximately

1.60E-03 * (7.76E-04+3*2.31E-04) = 2.35E-06. The fail to run probability of an AFW pump in the PRA model is 4.24E-04, which is still 2 orders of magnitude higher than the AFW fail to run due to runout failure for the case where one MSIV is unavailable.

Accordingly, the impact on other sequences, expression of risk and the RICT is not significant. The associated AFW pump will be made unavailable in the RICT model if runout protection is unavailable.

Audit Question APLA-04 *(Procedures)*

The NRC staff's safety evaluation to NEI 06-09 specifies that the LAR should include discussion of the licensee's programs and procedures which assure the PRA models that support the risk managed technical specifications (RMTS) are maintained consistent with the as-built, as-operated plant.

a. *The LAR states, "Plant changes that meet the criteria defined in the PRA Configuration Control Program (including consideration of the cumulative impact of other pending changes) will be incorporated into the applicable PRA model(s) as an interim update, consistent with the NEI 06-09-A guidance."*

    i. *Identify the criteria that will be used to assess impact on the RICT Program.*

PG&E Response:

This criteria is still in development as part of TSTF-505 implementation. The current draft process requires a periodic cumulative assessment of pending changes to the PRA model. If the increase (total from all hazards) is greater than 1E-05/yr for CDF or 1E-06/yr for LERF, the program will require an evaluation of the potential impact on future RICT calculations. If necessary, an interim model update will be required.

    ii. *Confirm that an appropriate reference will be included in the configuration control program procedure to cite RMTS guidance.*

PG&E Response:

As part of implementation, the guidance document that includes this impact assessment process will be included in the configuration control program procedure.

b. *The industry guidance NEI 06-09 also states that the purpose of this tracking is to demonstrate the risk accumulated as a result of SSC inoperability beyond the front-stop completion time is appropriately managed.*

An example of tracking is presented in the industry guidance. The accumulated risk is monitored on entering such plant configurations, that is, when the front-stop completion time is exceeded. An alternative presented is to maintain a 52-week rolling average CDF, updated weekly. In contrast, Enclosure 11 to the LAR states that the calculation of cumulative impact will be required every refueling cycle.

Justify the calculation of cumulative impact only once per refueling cycle and explain how this will be adequate to manage risk in accordance with the risk-informed principles in RG 1.174.

PG&E Response:

Enclosure 11 of the LAR states that the "calculation" of cumulative risk is performed each refueling cycle. The "tracking" of cumulative risk is automatically performed in real time by the CRMP software. The formal assessment of cumulative risk impacts, as described in Enclosure 11 of the LAR, involves determining the impact on the average annual risk model, and includes consideration of various factors as described in Enclosure 11. This formal assessment, performed each refueling cycle, assesses the planned and emergent use of the RICT Program and the risk impact of its use, and if appropriate, may result in administrative limitations on the RICT program application to ensure the program implementation over long periods of time is conforming to the guidance of RG 1.174.

Audit Question APLA-05 *(Risk management actions)*

Enclosure 12 to the LAR presents examples of risk management actions and explains the basis for calculating a risk management action time (RMAT). Clarify the limits proposed for incremental core damage probability, large early release probability, instantaneous CDF, and instantaneous large early release frequency.

PG&E Response:

The RMATs will be calculated using the risk thresholds identified in NEI 06-09-A Table 3-1 for both an ICDP ≥ 1E-6 and ILERP ≥ 1E-7. As noted on Table 3-1, configurations with CDF ≥ 1E-3/year or LERF ≥ 1E-4/year would not be voluntarily entered, and if incurred on an emergent basis, would require risk management actions to be implemented.

Audit Question APLA-06 *(CRMP model)*

Regulatory Position 2.3.3 of RG 1.174, Revision 3, states that the level of detail in the PRA should be sufficient to model the impact of the proposed licensing basis change.

The characterization of the problem should include establishing a cause-effect relationship to identify portions of the PRA affected by the issue being evaluated. Full-scale applications of the PRA should reflect this cause-effect relationship in a quantification of the impact of the proposed licensing basis change on the PRA elements.

Section 4.2 of NEI 06-09, Revision 0-A, describes attributes of the CRMP. A few of these attributes are listed below:

- *Initiating events accurately model external conditions and effects of out-of-service equipment.*
- *Model translation from the PRA to a separate CRMP tool is appropriate; CRMP fault trees are traceable to the PRA. Appropriate benchmarking of the CRMP tool against the PRA model shall be performed to demonstrate consistency.*
- *Each CRMP application tool is verified to adequately reflect the as-built, as-operated plant, including risk contributors which vary by time of year or time in fuel cycle or otherwise demonstrated to be conservative or bounding.*
- *Application specific risk important uncertainties contained in the CRMP model (that are identified via PRA model to CRMP took benchmarking) are identified and evaluated prior to use of the CRMP tool for RMTS applications.*
- *CRMP application tools and software are accepted and maintained by and appropriate quality program.*
- *The CRMP tool shall be maintained and updated in accordance with approved station procedures to ensure it accurately reflects the as-built, as-operated plant.*

Enclosure 8 of the LAR describes the attributes of the CRMP model for use in RICT calculations. The LAR also describes several changes made to the baseline PRA models to support calculation of configuration-specific risk and mentions approaches for ensuring the fidelity of the CRMP tool. Table E8-1 provides a list of CRMP model changes for configuration-specific risk. Clarify the following:

c. (Table E8-1 first item) Clarify how the plant availability factor is addressed in the CRMP tool. The LAR states that the initiating events frequencies are adjusted to per critical year for the CRMP tool, as opposed to per calendar year, however this appears to contradict the remainder of the text.

PG&E Response:

Some initiating event frequencies in the internal events Diablo Canyon PRA are calculated based on a per calendar year basis by applying an availability factor based on historical plant data.

For the CRMP and the RICT Program calculations of configuration-specific risk while in Modes 1 and 2, these initiating event frequencies will be manually adjusted in the model translation from the PRA to the CRMP model to remove the plant availability factor, and thus increase the frequencies numerically from a per calendar year to a per reactor-critical-year basis.

b. *(Table E8-1 fourth item) The LAR states the PRA model includes conservative success criteria for room cooling. Summarize the criteria and explain how they are conservative.*

PG&E Response:

The use of design basis data, calculations and testing data are used to establish equipment temperature damage thresholds. Area or room heatup calculations were performed using conservative criteria to establish room temperatures under accident conditions.

The room heatup analysis used to determine room heatup success criteria for DCPP makes a number of conservative assumptions including the following:

1) Conservative external temperatures are assumed in the heatup calculations.
2) The movement of air by natural convection through ventilation ductwork is not modeled when forced ventilation is lost. This is conservative as the heated air is held within each room when forced ventilation is lost and no operator actions are taken.

Because the room heatup calculations used as input to the PRA success criteria include these conservative assumptions no seasonal impacts are included for room heat up in the CRMP model.

c. *(Table E8-1 seventh item) The LAR states that the baseline PRA model includes credit for a backup portable fuel oil (DFO) pump. It further states this is not credited for the CRMP model when a diesel fuel oil transfer system pump is out of service.*

   Further explain how the credit for backup portable fuel oil pump is removed from the CRMP model.

PG&E Response:

The basic event that represents the portable DFO pump will be set to TRUE whenever a DFO transfer pump is out of service. This will be accomplished by mapping the DFO transfer pump component in the CRMP tool to the DFO transfer pump basic event as well as the portable DFO basic event. The operator will only need to select the DFO pump code in the CRMP software tool since the portable DFO pump impact will be pre-established / pre-programmed? into the DFO CRMP code.

Audit Question APLA-07 *(Open Phase Condition)*

Section C.1.4 of RG 1.200 states the base PRA is to represent the as-built, as-operated plant to the extent needed to support the application. The licensee is to have a process that identifies updated plant information that necessitate changes to the base PRA model.

In response to the January 30, 2012, event at the Byron Generating Station initiated by an open-phase condition (OPC), the NRC issued Bulletin 2012-01. As part of the initial voluntary industry initiative for mitigation of the potential for the occurrence of an OPC in electrical switchyards, licensees have modified their designs to add an open-phase isolation system (OPIS). Per the Staff Requirements Memorandum for SRM-SECY-16-0068, the NRC staff was directed to ensure that licensees have appropriately implemented OPIS and that licensing bases have been updated accordingly. From the revised voluntary initiative and resulting industry guidance in NEI 19-02, "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights" (ADAMS Accession No. ML19122A321), on estimating OPC and OPIS risk, it is understood that the risk impact of an OPC is highly dependent on electrical switchyard configuration and design.

a) For Diablo Canyon, discuss the evaluation of the risk impact associated with OPC events including the likelihood of OPC initiating plant trips and the impact of those trips on PRA modeled SSCs. Report whether an OPIS has been installed. If such a system has been installed, discuss its function and operation. Include any operator actions needed to activate the system or to respond if it annunciates or actuates automatically.

PG&E Response:

During the virtual audit meeting on December 12, 2023, the staff informed PG&E the information provided during the audit for this question did not need to be submitted to the NRC.

b) Clarify whether any installed OPIS equipment and associated operator actions are credited in the PRA models that support this application. If OPIS equipment and associated operator actions are credited, then provide the following information:

i. Describe the OPIS equipment and associated actions that are credited in the PRA models.
ii. Describe the impact, if any, that this treatment has on key assumptions and sources of uncertainty for the RICT program.
iii. Discuss human reliability analysis (HRA) methods and assumptions used for crediting OPIS alarm manual response.

iv. Discuss how OPC-related scenarios are modeled for non-internal event scenarios such as internal floods, fire, and seismic.

v. Regarding inadvertent OPIS actuation:

- *Explain whether scenarios regarding inadvertent actuation of the OPIS, if applicable, are included in the PRA models that support the RICT calculations.*
- *If inadvertent OPIS actuation scenarios are not included in the PRA models, then provide justification that the exclusion of this inadvertent actuation does not impact the RICT calculations.*

PG&E Response:

During the virtual audit meeting on December 12, 2023, the staff informed PG&E the information provided during the audit for this question did not need to be submitted to the NRC.

c) If OPC and OPIS are not included in the application PRA models (whether OPIS equipment is installed or not), then provide justification that the exclusion of this failure mode and mitigating system does not impact the RICT calculations.

PG&E Response:

During the virtual audit meeting on December 12, 2023, the staff informed PG&E the information provided during the audit for this question did not need to be submitted to the NRC.

d) As an alternative to Part (c), propose a mechanism to ensure that OPC-related scenarios are incorporated into the application PRA models prior to implementing the RICT program.

PG&E Response:

During the virtual audit meeting on December 12, 2023, the staff informed PG&E the information provided during the audit for this question did not need to be submitted to the NRC.

<u>*DRA, PRA Licensing Branch C (APLC)*</u>

Audit Question APLC-01

Section 4.0, Item 5 of the final safety evaluation for NEI 06-09 states that the LAR will provide a justification for excluding any risk sources determined to be insignificant to the calculation of configuration-specific risk and will provide a discussion of any conservative or bounding analyses to be applied to the calculation of RICTs for sources of risk not addressed by the PRA models.

Enclosure 4 of the LAR discusses the generic methodology used to identify and disposition such risk sources and provides the plant-specific results of the application of the generic methodology for impacts to the RICT program. One of the screening criteria (screening criterion B) used to disposition the risk sources is if the CDF, calculated using a bounding or demonstrably conservative analysis, has a mean frequency of less than 1E-6 per year.

Table E4-1 of enclosure 4 of the LAR provides the external hazards evaluated, identifies the applicable screening criteria, summarizes the evaluation, and provides a disposition for the RICT program. This table includes the aircraft impact, extreme wind or tornado, hurricane, and tsunami external hazards, and it screens each of these external hazards using screening criterion B as follows:

- *For the aircraft impact external hazard, the table states that the CDF from an aircraft crash is estimated to be 7.43E-7 per year.*

  For the extreme wind or tornado external hazard, the table states that a conservative strike frequency of a tornado is 7.0E-5 per year, the conditional core damage probability (CCDP) for a loss of offsite power (LOOP) due to severe weather with no recovery is 5.16E-4 per year, which results in a conservatively estimated CDF from a tornado event of 3.92E-8 per year. The table also states that a conservatively estimated CDF from tornado missile events is 2.05E-7 per year.

  For the hurricane external hazard, the table states that a conservatively estimated CDF from hurricanes is 5.0E-7 per year based on an assumption that a hurricane with wind speeds of 150 mph leads directly to core damage.
- *For the tsunami external hazard, the table states that the CDF from flooding of the intake structure due to a tsunami is estimated to be 2.2E-8 per year.*

Table E4-1 neither provides the basis for these values nor does it describe the assumptions and methodology used to calculate them. The details of these calculations *are discussed in Calculation X.1, Revision 1, "DCPP Other External Events," which was not provided on the docket as part of the LAR.*

The NRC staff reviewed Calculation X.1 during the audit and identified the following issues:

- *Section X.1.8.1 describes the analysis of tornado-generated missiles using the TORMIS methodology. The calculation identifies the plant targets with a high tornado missile damage probability, which includes targets 100, 126, 99, 109, 31, 6, and 55. The calculation states that it provides a discussion of the risk impact of tornado-generated missiles for each target. The calculation provides a discussion of targets 100, 126, 99, and 31. The NRC staff identified that the calculation does not provide a discussion of targets 109, 6, and 55.*
- *The TORMIS methodology determines the probability of components being struck and disabled by a tornado-generated missile, and it was accepted for use by the NRC in a safety evaluation report (ML080870291). This safety evaluation report contains several items to consider when using the TORMIS methodology. The TORMIS methodology is also discussed in Regulatory Issue Summaries (RISs) RIS 2008-14, "Use of TORMIS Computer Code for Assessment of Tornado Missile Protection" (ADAMS Accession No. ML080230578) and RIS 2015-06, "Tornado Missile Protection" (ADAMS Accession No. ML15020A419).*
- *Table X.1.8.1-1 summarizes the results of the analysis of tornado-generated missiles. This table contains a column labeled "Hit" and another column labeled "Damage (Base)." For many of the plant targets, the values in both columns are the same. However, there are some plant targets with different values in these columns. It is unclear to the NRC staff what information these columns are intended to represent.*
- *Section X.1.9 describes the analysis of hurricanes. The calculation uses data from four recorded tropical storms in the 20th century and a hurricane in the 19th century to fit an extreme value distribution to estimate the probability of hurricane winds exceeding 150 mph. The calculation provides a reliability function (also known as a survival function) of the form*

  $$P(V \geq V_0) = 1 - exp\big(-\alpha(V_0 - \beta)\big)$$

  in one location and

  $$P(V \geq V_0) = 1 - exp(1) - exp\big(-\alpha(V_0 - \beta)\big)$$

in another location. *The calculation provides the following formulas to estimate the parameters from the extreme value distribution.*

$$\alpha = \frac{\sqrt{1.65}}{s}$$

$$\beta = \bar{x} - \frac{0.577}{\alpha}$$

In these formulas, $\bar{x}$ *and* s *represent estimates of the mean and standard deviation of the extreme value distribution, respectively. Although not stated, it appears that the calculation estimates* α *and* β *using the method of moments estimation technique.*

The NRC staff identified two concerns with this calculation. First, the NRC staff identified that the reliability function provided is not correct. The correct reliability function for the extreme value distribution is:

$$P(V \geq V_0) = 1 - exp\big(-exp\big(-\alpha(V_0 - \beta)\big)\big)$$

Second, it appears that calculation used the formula for the sample standard deviation to obtain the estimates of $\alpha = 0.117$ *and* $\beta = 51.9$*. The NRC staff identified that the method of moments technique uses the formula for the population standard deviation, which results in different estimates for* α *and* β*.*

Address the following:

Summarize the evaluation of the aircraft impact, extreme wind or tornado, hurricane, and tsunami external hazards. For this item, describe the data sources used to determine the frequency of the external events, summarize the assumptions and methodology used to calculate the CDFs, and summarize the results.

PG&E Response:

A short summary of the evaluations for aircraft impact, extreme wind or tornado, hurricane and tsunami (assumptions, data sources, methodology and results) are included below.

Aircraft crashes were assessed in accordance with the guidance of U.S. NRC Standard Review Plan (SRP) NUREG-0800, Section 3.5.1.6, Aircraft Hazards. A detailed evaluation of aircraft hazards was performed; e.g., quantitative analysis using the formula provided in Section 3.5.1.6 of NUREG-800. Nearby airport information was collected from the FlightAware website, FltPlan.com and SkyVectorAeronautical Charts which includes types of aviation transient landing and departure operations and airways near DCPP. Flight data was generated by the Air Traffic Organization, System Operations Services, Data Management office using the FAA Traffic Flow Management System (TFMS) repository. The core damage frequency is calculated by risk impact of building failures using the conditional core damage probability of the buildings (which is taken from the PRA model) and the aircraft impact frequency. The total CDF induced by aircraft crash at Diablo Canyon Unit 1 is less than 1E-06 per year. Unit 2 is expected to have a similar risk from aircrafts due to the shared building structures and near identical non-shared building structures. Since the total CDF resulting from aircraft crash is below Screening Criterion B of Supporting Requirement [EXT-C1] of ASME/ANS RA-Sb-2013, the aircraft crash hazard from airways can be screened out.

Tornado wind hazard is assessed using frequencies from ARA-002233, “Tornado Missile TORMIS 2014 Analysis of DCPP,” and using the conditional core damage probability for a loss of offsite power due to severe weather (with no recovery) from the PRA model. It is noted that the TORMIS methodology has only been used to support PRA model screening of the tornado external hazard and it is not included in the DCPP design and licensing basis. The result for lower wind speed tornados is a core damage frequency less than 1E-06, which screens out. It is assumed that DCPP can withstand at least a 200 mph wind without major damage (such as collapse of a wall or overturning of a structure) and a maximum of 300 mph tornado wind speeds without causing a LOCA or structural damage impairing containment integrity. (Section 3.3 of “Units 1 and 2, Diablo Canyon Power Plant Final Safety Analysis Report Update” (UFSAR), Pacific Gas and Electric Company). For these high wind speed tornados, the frequency of occurrence is much less than 1E-09 per year. With such low initiator frequencies, it is judged that tornado wind-initiated scenarios are insignificant contributors to the overall core damage frequency and, therefore screened out.

Tornados missiles are assessed using the frequency of a tornado striking the site estimated from the DCPP Tornado Hazard Curves from ARA-002233, “Tornado Missile TORMIS 2014 Analysis of DCPP.” The tornado data used in this study was downloaded from the NOAA SPC website for the years 1950 – 2013. This data has the same source of information as the NCDC Storm events database and is the source of almost all publications and research into tornado hazards in the U.S. The impact of this external event is a loss of offsite power and the conditional core damage probability for loss of offsite power is taken from the PRA model. The core damage frequency from tornado winds is less than 1E-06 and therefore not significant. Tornado missiles are evaluated using the TORMIS process. The damage frequencies are calculated from TORMIS and the risk impact of the target failures are taken from the PRA model. Scenarios involving tornado missiles have been evaluated with a core damage frequency of less than 1E-06 per year for Unit 1. Unit 2 has a similar impact. This CDF frequency is low enough to be screened out per Screening Criterion B.

Hurricanes are evaluated based on a frequency of exceedance corresponding to a failure of the outdoor storage tanks, conservatively resulting in core damage. An extreme value distribution was created from a dataset of hurricanes in the region of DCPP. The parameters of the distribution were estimated using the method of moments technique. Conservatively assuming a hurricane with a wind speed greater than 150 mph that leads to core damage yields a CDF less than 1E-06 per year. This CDF frequency is low enough to be screened out per screening Criterion B.

Tsunamis are evaluated by calculating the frequency of exceedances of a wave that would impact the intake structure ASW pumps via the ASW pump room ventilation snorkels. Tsunami frequencies were taken from "Methodology for Probabilistic Hazard Analysis: Trial Application for the Diablo Canyon Power Plant Site”. The tsunami risk takes into account the failure of the ASW pump room water-tight door failure, tsunami warning alert, and operator error to close the water-tight doors. The conditional core damage probability of a loss of the Auxiliary Saltwater (ASW) system is used to calculate the overall core damage frequency (CDF) from a large wave. The calculated CDF is less than 1E-06 per year and is low enough to be screened out per screening Criterion B.

For the evaluation of extreme wind or tornado external hazard:

i. Discuss if the TORMIS methodology is included in the plant’s licensing basis. If the TORMIS methodology is included in the plant’s licensing basis, identify and justify any deviations from the methodology in the licensing basis and that used for this application. If TORMIS is not included in the plant’s licensing basis, justify that the TORMIS methodology was used consistent with the items of consideration in the safety evaluation report that approved its use, RIS 2008-14, and RIS 2015-06.

PG&E Response:

The TORMIS methodology is not included in the DCPP design and licensing basis. PG&E had planned to submit an LAR for TORMIS to address tornado missile protection for some plant components, however due to complications for other licensees in obtaining NRC approval of use of TORMIS methodology, PG&E instead utilized the NEI 17-02 Revision 1B, “Tornado Missile Risk Evaluator (TMRE) Industry Guidance Document, Nuclear Energy Institute”, September 2018, method through 10 CFR

50.59. As described in UFSAR Section 3.3.2.5.1 "Major Findings of the Tornado Analysis," the following components have been concluded to not require additional physical protection from tornado missiles based on the low safety significance associated with their exposures as determined by the TMRE method (described in UFSAR Section 3.3.2.6.3):

- Diesel generator engine exhaust lines
- Diesel generator ventilation system exhaust plenums
- Large Diameter 4.16 kV raceways routed through the 12 kV switchgear rooms
- Class 1E raceways containing Diesel Generator 2-3 circuits routed along the 140' elevation of the Unit 2 Turbine Building and through the 12 kV switchgear room

Since PG&E has not previously used the TORMIS methodology as a basis for the DCPP plant design and licensing basis, a comparison of TORMIS methodology for consistency with the items of consideration in the safety evaluation report that approved its use (as described in RIS 2008-14, and RIS 2015-06) was not required and was not previously performed.

PG&E has used site specific data developed using the TORMIS methodology as part of the data used for the PRA model screening of the tornado external hazard. Data from NUREG/CR-4461, Revision 2, "Tornado Climatology of the Contiguous United States", February 2007 was also used.

In the PRA Calculation File X.1 for PRA Other External Event screening, the TORMIS method was not used for a licensing basis to preclude installation of tornado missile protection for SSCs as described in TORMIS Safety Evaluation and RIS 2008-14 and RIS 2015-06. Data from the TORMIS calculation performed in ARA-002233, "Tornado Missile TORMIS 2014 Analysis of DCPP," is used to estimate probabilistic risk associated with tornado missiles.

The TORMIS methodology is not included in the DCPP design and licensing basis as addressed in the SER for missile protection. However, to answer the question above, the response to the TORMIS SER list of items is as follows:

TORMIS SER:

(1) Licensees should employ data on tornado characteristics for both broad regions and small areas around the site, with the most conservative values used in the risk analysis, or justify the values selected.

PG&E Response:

The DCPP tornado frequency value conservatively considers regions around the plant and corrects for reporting trend and tornado classification error and random encounter errors. The tornado data used in this study was downloaded from the NOAA SPC website (www.spc.noaa.gov) for the years 1950 – 2013. This data has the same source of information as the NCDC Storm events database and is the source of almost all publications and research into tornado hazards in the U.S. The tornado hazard model used for DCPP meets the TORMIS SER item 1 requirement: “Data on tornado characteristics should be employed for both broad regions and small areas around the site. The most conservative values should be used in the risk analysis or justification provided for those values selected.”

Reference: ARA-002233, “Tornado Missile TORMIS 2014 Analysis of DCPP,” December 2014.

(2) Licensees should use the F-scale tornado classification rather than the modified tornado classification, F’-scale, employed in the EPRI studies.

PG&E Response:

Item 2 of the TORMIS SER states: “The EPRI study proposes a modified tornado classification, F'-Scale, for which the velocity ranges are lower by as much as 25% than the velocity ranges originally proposed in the Fujita, F-Scale. Insufficient documentation was provided in the studies in support of the reduced F'-Scale. The F-Scale tornado classification should therefore be used in order to obtain conservative results.”

The Enhanced Fujita (EF-Scale) of tornado winds has been used in the wind hazard curve for DCPP. The NRC has adopted NUREG CR-4461 (Ramsdell and Rishel, 2007) and the hazard data in that document is based on the EF wind scale. In the FERMI Safety Evaluation for TORMIS (FERMI 2- ISSUANCE OF AMENDMENT RE: REVISE THE FERMI 2 LICENSING BASIS CONCERNING PROTECTION FROM TORNADO-GENERATED MISSILES (TAC NO. MF0497), March 10, 2014, ML14016A487), the NRC concluded that the EF scale is acceptable.

Reference: ARA-002233, “Tornado Missile TORMIS 2014 Analysis of DCPP,” December 2014.

(3) Licensees should calculate the effect of assuming velocity profiles with ratios of speed at ground level (V0) to speed at the 33-foot elevation (V33) higher than that in the EPRI study. Licensees should discuss the sensitivity of the results to changes in the modeling of the tornado windspeed profile near the ground.

PG&E Response:

Because TORMIS is not part of the plant licensing basis, this sensitivity was not performed. Hazard wind speeds were calculated only for use in other external event screening evaluation.

(4) Licensees should provide sufficient information to justify the assumed missile density based on site-specific missile sources and dominant paths of travel.

PG&E Response:

Site specific walkdowns were performed at the plant to characterize missile sources and plant configuration. Missile sources surveyed include buildings/structures, rebar, gas cylinders, drum tanks, utility poles, cable reels, pipes of various sizes, storage bins, pavers, concrete fragments, wood beams and planks, metal siding, plywood, flanges, channel sections, small equipment, large equipment, steel grating, vehicles, and trees. More than 180,000 missiles were postulated. This is a reasonable missile density in comparison to some other plants that use 25,000 to 74,000 (FERMI 2- ISSUANCE OF AMENDMENT RE: REVISE THE FERMI 2 LICENSING BASIS CONCERNING PROTECTION FROM TORNADO-GENERATED MISSILES (TAC NO. MF0497), March 10, 2014, ML14016A487)

Reference: Applied Research Associate Inc., "StructureZoneMissileTablesDCPP_150210", December 2014.

(5) Licensees should justify any deviations from the calculation approach.

PG&E Response:

The TORMIS methodology is not included in the DCPP design and plant licensing basis as addressed in the SER for missile protection. The use of site-specific data developed using the TORMIS methodology for the PRA external events screening evaluation has no impact to the tornado generated missile licensing basis.

RIS 2008-14 was issued to address the following:

(1) the NRC staff position on the use of the TORMIS computer code for assessing nuclear power plant tornado missile protection

The use of the TORMIS methodology is not used for tornado missile protection. It is used as part of the data used for the PRA model screening of the tornado external hazard.

(2) issues identified in recent license amendment requests to use the TORMIS computer code

DCPP does not have a license amendment for the use of TORMIS. It is not part of the licensing basis.

(3) information needed in license amendment applications using the TORMIS computer

DCPP does not have a license amendment for the use of TORMIS. It is not part of the licensing basis.

“Licensees. . . not providing adequate justification that the analysis used the most conservative value for tornado frequency”

The DCPP TORMIS analysis fully complies with this RIS comment, which is basically a restatement of SER Item 1, discussed in Section 11.1. As part of the tornado hazard analysis, wind speed vs. probability of exceedance curves were developed for DCPP and compared to the tornado frequency data in NUREG CR-4461.

RIS 2015-06 was issued to remind plants to be in conformance with their tornado-generated missile licensing basis.

The TORMIS methodology is not included in the DCPP design and licensing basis. The use of site-specific data developed using the TORMIS methodology for the PRA external events screening evaluation has no impact to the tornado generated missile licensing basis.

ii. *Describe the risk impact of tornado-generated missiles for targets 109, 6, and 55. Describe the information the columns labeled “Hit” and “Damage (Base)” in table X.1.8.1-1 are intended to represent and the difference between them.*

PG&E Response:

During the virtual audit meeting on December 12, 2023, the staff informed PG&E the information provided during the audit for this question did not need to be submitted to the NRC.

For the evaluation of hurricane external hazard:

iii. *Describe the estimation technique used to obtain the parameter estimates for the extreme value distribution or provide a reference for the basis and formulas used to obtain the parameter estimates.*

PG&E Response:

The extreme value function method was taken from PLG-0637 Section F.5.6.3. The parameter estimates were calculated from the dataset mentioned in Section X.1.9 which lists on the windspeeds of identified hurricanes (52, 46, 57, 54, 75 mph). The calculated mean for the dataset is 56.8 mph and the calculated sample variance is 119.7. The variance was originally calculated based on a sample, not a population. The method of moments estimations was used for distribution parameters. However, the appropriate calculation for variation should be based on a population dataset. The new parameters and results were recalculated in the response below.

iv. *Confirm the reliability function for the extreme value distribution.*

PG&E Response:

The correct form of the reliability function/extreme value distribution is as follows:

$$\mathrm{P(V \geq V_0)} = 1 - exp\left(-exp\left(-\alpha(V_0 - \beta)\right)\right)$$

as shown in the second instance in Section X.1.9 [P(V≥V_0)=1-exp- e^(-α(V0-β))]. A parenthesis is not used which would help clarify. The first instance of the equation is shown incorrectly. The values are calculated using the correct 2nd form of the equation (using rounded values for alpha and beta).

v. *If any errors are identified in the parameter estimates or reliability function, provide an updated evaluation of the risk from hurricanes.*

PG&E Response:

Section X.1.9 of Calculation file was corrected to use the population dataset in calculating the parameter values. The new calculated value is less than previously calculated with the sample dataset. The conclusions remain unchanged.

These data are used to fit an extreme value distribution of the form

$$P(V \geq V_0) = 1 - exp\left(-exp^{-\alpha(V_0 - \beta)}\right)$$

Where the method of moments estimation technique is used calculate the equation parameters given below:

$$\alpha = (1.65/S^2)^{1/2}$$

$$\beta = V - \frac{0.577}{\alpha}$$

where V and S^2 are the mean and variance of the data, respectively. The variance is calculated based on a population dataset of 52, 46, 57, 54, and 75 mph (as opposed to based on a sample dataset). The mean value of the dataset is 56.8 mph and the variance is 95.76. From the data, α is 0.131 and β is 52.4. This gives the probability of exceeding V_0 = 150 mph of

$$P(V \geq V_0) = 1 - exp\left(-exp^{-\alpha(V_0 - \beta)}\right)$$

$$P(V \geq V_0) = 2.73E - 06$$

This gives $\varphi_{V|h}$ = 2.73E-06. Other parametric distributions such as lognormal and normal give smaller numerical values. The annual frequency of hurricane wind speeds in excess of 150 mph is therefore

$$F_{hurricane} = (0.05)(2.73e - 06) = 1.37E - 07\; per\; year$$

Conservatively assuming a hurricane with this wind speed leads to core damage yields a CDF of 1.37E-07 per year.

Audit Question APLC-02

Section 4.0, item 3 of the final safety evaluation for NEI 06-09 states that the LAR will provide a discussion of the results of peer reviews and self-assessments conducted for the plant-specific PRA models that support the RMTS, including the resolution or disposition of any deficiencies identified in peer reviews (i.e., facts and observations (F&Os)). This item states that the discussion will include a comparison of the requirements of RG 1.200 using the elements of ASME RA-Sb–2005, "Addenda to ASME RA-S–2002: Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," for capability Category II for internal events PRA models and for other models for which standards exist that have been endorsed in RG 1.200. This item further states that, if additional standards have been endorsed by revision to RG 1.200, the LAR will also provide similar information for those PRA models used to support the RMTS program.

Enclosure 2 of the LAR addresses this requirement by providing information on the technical acceptability of the internal events, internal flood, fire, and seismic PRA models that support the RICT program. The LAR states that this information is consistent with the requirements of section 4.0, item 3 of the final safety evaluation for NEI 06-09 and addresses each PRA model for which a PRA standard endorsed by RG 1.200, Revision 2 exists.

The LAR states that the PRA models were peer reviewed and assessed using ASME/ANS RA-Sa–2009, "Standard for Level 1/Large Early Release Frequency

Probabilistic Risk Assessment for Nuclear Power Plant Applications," and RG 1.200, Revision 2. For the seismic PRA, however, the LAR states that a full-scope seismic PRA peer review, which also included a review of the seismic hazard and fragility analyses, was conducted in June 2017, and it was performed consistent with RG 1.200, Revision 2, using ASME/ANS RA-Sb–2013, "Probabilistic Risk Assessment for Advanced Non-Light Water Reactor Nuclear Power Plants." The LAR states that an independent assessment of the finding-level F&Os was conducted from October–December 2017 and the scope of the assessment included all finding-level F&Os resulting from the peer review. The LAR also states that a focused-scope peer review was conducted in conjunction with the closure review and that all F&Os categorized as findings have been resolved by either a PRA model revision or a documentation update.

The NRC staff notes that RG 1.200, Revision 2, endorses ASME/ANS RA-Sa–2009, but it does not endorse ASME/ANS RA-Sb–2013. Similarly, the NRC staff notes that RG 1.200, Revision 3, does not endorse ASME/ANS RA-Sb–2013. As discussed in RG 1.200, Revision 2, a risk-informed submittal should contain discussions concerning peer review. If the peer review is not performed against the endorsed standards, RG 1.200, Revision 2 states that information needs to be included in the submittal that demonstrates that the different criteria used are consistent with the endorsed standards.

Address the following:

a. *Provide a comparison of the criteria in ASME/ANS RA-Sb–2013, which has not been endorsed by the NRC for licensing applications, with the criteria in the endorsed ASME/ANS RA-Sa–2009, including an explanation that demonstrates that the analogous ASME/ANS RA-Sa–2009 supporting requirements have been met for instances where the criteria between the two standards differ.*

PG&E Response:

Diablo Canyon performed a gap assessment between Addendum A and B consistent with that from Southern Nuclear Operating Company, Inc., letter to NRC, NL-17-1201, "Vogtle Electric Generating Plant Units 1 and 2 Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report," dated July 11, 2017 (ML17192A245). NRC acceptance of the assessment was documented in a letter to Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment Into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248)," dated August 10, 2018 (ML18180A062). In the Vogtle Assessment, all but six of the Addendum B SRs have been shown to either be equal to the corresponding Addendum A SRs or have been shown to envelop the corresponding Addendum A SRs. The remaining six SRs (SHA-B3, SHA-C3, SFR-C3, SFR-C6, SFR-G3, and SPR-B1) were assessed and the DCPP

SPRA was shown to conform to these Addendum A SRs (See table A2-2 in Attachment 2).

b. *Discuss if any changes were made to the seismic PRA since the peer review was completed. For each such change, identify the change and discuss, with justification, if the change is PRA maintenance or a PRA upgrade per the definition in RG 1.200, Revision 2.*

PG&E Response:

See Table A2-1 in Attachment 2 for assessment of SPRA model changes since the last update. All changes were classified as PRA maintenance.

Audit Question APLC-03

APLC staff requests a discussion on quantification of the seismic PRA model for RICT calculations, including any simplifications that were considered or implemented.

PG&E Response:

No Seismic PRA simplifications have been made for RICT calculations. DCPP has considered simplification for high CCDP seismic initiators if quantification time for the seismic PRA becomes an issue but this has not been required as of yet.

<u>DSS, Technical Specifications Branch (STSB)</u>

Audit Question STSB-01

The LAR Table E1-1, "In-Scope TS/LCO Conditions to the Corresponding PRA Functions," lists the technical specification number, but not the specific condition. More than one condition may be associated with a given number. Typically, these conditions are distinguished from one another by a letter, and different RICTs are calculated for each condition.

The NRC staff considers that each row of Table E1-1 should identify a condition with a proposed risk-informed completion time (RICT). Information provided at the LCO level as currently shown is not sufficient for NRC staff review. Revise Table E1-1 to provide TS LCO conditions and RICT applicable to each row. Note: The design success criteria (DSC) are the minimum set of remaining credited equipment that can achieve the TS safety function while in the specified TS Condition. For example, a condition with two of two offsite circuits inoperable should not have a DSC of one offsite circuit.

PG&E Response:

Table E1-1 identifies each LCO for which a RICT is proposed for one or more Conditions and associated Required Actions. An LCO identifies the function(s) that are controlled, and Table E1-1 identifies those functions, the design success criteria, and the PRA success criteria (if different). The functions and success criteria are not unique to the individual Conditions, and are not altered by the Conditions. Including each Condition separately in Table E1-1 would simply duplicate the information already provided for each LCO. An updated table E1-1 is contained in Attachment 1 to the Enclosure of this letter that provides additional information requested by the staff during audit discussions. This Table supersedes the table E1-1 contained in the LAR.

Individual Conditions are identified in Table E1-2 for each unique Condition within the LCO where a RICT is proposed, and the example calculated RICT is identified for each Condition.

Audit Question STSB-02

Attachment 1 to the LAR references the Risk-Informed Completion Time Program by stating "In accordance with the RICT Program" instead of spelling the program name out. This is inconsistent with the Diablo Canyon TS and TSTF-505, Revision 2 where each program reference spells the name out fully except when the acronym for the program is defined on the same page where the acronym is used. Provide justification for this variation.

PG&E Response:

The LAR defined the Risk Informed Completion Time Program as the "RICT Program" in the first time it is used in the proposed TS. TS 1.3 Example 1.3-8 states "However, the licensee may elect to apply the Risk Informed Completion Time (RICT) Program which permits calculation of a RICT that may be used" and also in the new TS Section 5.5.20 title "Risk Informed Completion Time (RICT) Program." This a variation to the TSTF-505 term used which is "Risk Informed Completion Time Program."

PG&E personnel have been using the acronym RICT and the phrase "RICT Program" since 2016, consistent with use in the industry, when a Risk-informed Completion Time or the "Risk Informed Completion Time Program" are being referred to. TSTF-GG-05-01, "Writer's Guide for Plant Specific," June 2005, Section 3.3.2.b states that "Some acronyms and initialisms are commonly used and understood. In many instances they are more commonly understood than the complete spelled out phrase. These may be used without initially defining them with the full phrase." Therefore, use of the acronym RICT within the proposed TS meets the TSTF-GG-05-01 guidance.

Audit Question STSB-03

In Attachment 1 to the LAR, the licensee proposes a variation for technical specification (TS) Condition 3.4.11.F, which provides restoration requirements for PORV Block Valves (i.e., more than one inoperable). The licensee's Required Action (RA) F.2 appears to be similar to Standard Technical Specification (STS) RA F.1, which is a loss of function condition. TSTF-505 does not provide a RICT for STS RA F.1. The licensee did not provide a technical justification for applying a RICT to this condition and referred to it as an editorial change. Describe how the proposed TS changes preclude the application of the RICT program for loss of function conditions or modify the proposed TS changes to do so.

PG&E Response:

Based on further discussion with the staff during the audit, PG&E withdraws the request to apply a RICT to TS 3.4.11 Condition F Required Action F.2. A RICT is still appropriate to be applied to TS 3.4.11 Condition F Required Action F.3 to support application of a RICT for Condition C (one block valve inoperable) Required Action C.2 (if the block valve is associated with a Class I PORV: restore block valve to OPERABLE status). A revised TS page 3.4-21 is contained in Attachment 3 to the Enclosure of this letter.

Audit Question STSB-04

What is the reason for Condition D in LCO 3.6.6? Typically, a licensee must enter all applicable conditions at the same time. Condition A is for one inoperable Containment Spray System. Condition C is for one inoperable Containment Fan Cooler Unit (CFCU) system. To help staff, ensure that the RICT program can be appropriately applied, discuss the reasons for a separate Condition D for one inoperable CS and one inoperable CFCU.

PG&E Response:

During the virtual audit meeting on December 12, 2023, the staff informed PG&E the information provided during the audit for this question did not need to be submitted to the NRC.

<u>*DEX, Electrical Engineering Branch (EEEB)*</u>

Audit Question EEEB-01

In reference to Table E1-2: "Unit 1/Unit 2 In-Scope TS/LCO Conditions RICT Estimate," clarify the term "subsystem" for the following considering that Updated Final Safety Analysis Report, Chapter 8 refers to "load groups," "groups," "channels," and "trains."

d. TS 3.8.9.A—One AC electrical power distribution subsystem inoperable

e. TS 3.8.9.B—One AC vital bus electrical power distribution subsystem inoperable

f. TS 3.8.4.C—DC electrical power subsystem inoperable.

PG&E Response:

The Diablo Canyon TS are based on NUREG-1431, Revision 1, and the associated Standard Technical Specification terms used for electrical system components. The NUREG-1431, Revision 1, terms used for electrical system components can be different than the terms used in the Updated Final Safety Analysis Report.

The Diablo Canyon TS 3.8.9 Bases describe the terms associated with the AC electrical power distribution subsystems (associated with Condition A) as follows:

"There are three AC electrical power subsystems, each comprised of a primary ESF 4.16-kV bus and secondary 480-V and 120-Vac buses, distribution panels, motor control centers and load centers. Each 4.16-kV ESF bus has two separate and independent offsite sources of power as well as a dedicated onsite diesel generator (DG) source. Each 4.16-kV ESF bus is normally connected to the 500-kV offsite source. After a loss of this normal 500-kV offsite power source to a 4.16-kV ESF bus, a transfer to the alternate 230-kV offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16-kV ESF bus. Control power for the 4.16-kV breakers is supplied from the Class 1E batteries."

The Diablo Canyon TS 3.8.9 Bases describe the terms associated with the 120 VAC vital bus subsystem (associated with Condition B) as follows:

"The 120-Vac Class 1E buses are arranged in four buses and are normally powered from the inverters. The alternate power supply for the 120-Vac Class 1E buses are Class 1E constant voltage source transformers powered from the same bus as the associated inverter, and its use is governed by LCO 3.8.7, 'Inverters - Operating.' Each constant voltage source transformer is powered from a Class 1E AC bus. In addition, each inverter can be powered from a bus other than its associated bus."

The Diablo Canyon TS 3.8.4 Bases describe the terms associated with the DC electrical power subsystems (associated with Condition C) as follows:

“The 125-Vdc- electrical power system consists of three independent Class 1E DC electrical power subsystems. Each subsystem consists of one 60-cell 125-Vdc battery (Batteries 11(21), 12 (22), and 13 (23)), the dedicated battery charger and backup charger for each battery, and all the associated switchgear, control equipment, and interconnecting cabling.”

TS 3.8.9 also addresses the DC electrical power distribution subsystems (Condition C) as follows:

“There are three independent 125-Vdc electrical power distribution subsystems (one for each bus).”

The Diablo Canyon TS 3.8.9 Bases include Table B 3.8.9-1 that specifies equipment associated with each of Conditions A, B, and C:

“Table B 3.8.9-1 (page 1 of 1)
AC and DC Electrical Power Distribution Systems

LCO 3.8.9 CONDITION A
4.16-kV and 480-V”

VOLTAGE	BUS F MAJOR ESF LOADS (TRAIN A)	BUS G MAJOR ESF LOADS (TRAIN B)	BUS H MAJOR ESF LOADS (TRAIN A&B)
4.16-kV	ASW PP 1 AFW PP 3 CC PP 1 CCW PP 1 SI PP 1 480-V BUS F	ASW PP 2 CS PP 1 RHR PP 1 CC PP 2 CCW PP 2 480-V BUS G	AFW PP 2 (B) CS PP 2 (A) RHR PP 2 (A) SI PP 2 (B) CCW PP 3 (A&B) 480-V BUS H
480-V *	CFCU 1 CFCU 2	CFCU 3 CFCU 5	CFCU 4 (A&B)

* Partial listing of loads

“LCO 3.8.9 CONDITION B
120-Vac”

BUS 1 PY11 (21)** PY11A (21A)**	BUS 2 PY12 (22)**	BUS 3 PY13 (23)** PY13A (23A)**	BUS 4 PY14 (24)**
IY Powered by: 480-V BUS F/DC BUS 1 or TRY1 Powered by: 480-V BUS F or Backup 480-V BUS G	IY1 Powered by: 480-V BUS G/DC BUS 2 or TRY2 Powered by: 480-V BUS G or Backup 480-V BUS F	IY Powered by: 480-V BUS H/DC BUS 3 or TRY3 Powered by: 480-V BUS H or Backup 480-V BUS G	IY Powered by: 480-V BUS H/DC BUS 2 or TRY4 Powered by: 480-V BUS H or Backup 480-V BUS F

** Unit 2 in parentheses

"LCO 3.8.9 CONDITION C
125-Vdc"

DC BUS 1 - Powered From:	DC BUS 2 - Powered From:	DC BUS 3 - Powered From:
Battery 1 and Battery Charger 11 (21)** or Battery Charger 121 (221)**	Battery 2 and Battery Charger 12 (22)** or Battery Charger 121 (221)**	Battery 3 and Battery Charger 131 (231)** or Battery Charger 132 (232)**

** Unit 2 in Parentheses

Audit Question EEEB-02

The following statement is from Diablo Canyon FSAR section 8.3.1.1.2.3.2, General Design Criterion 4, 1967—Sharing of Systems:

The 12-kV system is designed with crosstie capability to align a single 230 kV / 12 kV standby startup transformer (11 or 21) to provide power to both units via the crosstie breaker. Operation in this configuration is restricted by Technical Specification. The shared portion of the 12-kV system *[emphasis added] is designed with sufficient capacity and capability to operate the ESFs for a design* *basis accident (or unit trip) on one unit, and those systems required for a concurrent safe shutdown of the second unit consistent with the requirements of IEEE 308–1971, Section 8.*

g. Explain which TS section restricts the 12 kV crosstie capability to align a single 230 kV / 12 kV standby startup transformer (11 or 21).

h. Under what conditions is this sharing permitted by TSs?

PG&E Response:

TS Section 3.8.1 "AC Sources – Operating," which requires two qualified circuits between the offsite transmission network and the onsite Class IE AC Electrical Power Distribution System to be OPERABLE, provides controls for the 12 kV crosstie capability to align a single 230 kV / 12 kV standby startup transformer (11 or 21). DCPP operating procedures contain guidance on offsite power circuit system conditions that represent an unanalyzed condition to ensure that entry into a TS 3.8.1 Condition A entry is performed when the one offsite power circuit system is in a condition that is not bounded by the electrical system design and accident analyses assumptions. The operating procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," identifies that if the 12 kV crosstie breaker is closed ("IF 52VU11 is CLOSED"), then the startup offsite power circuit is inoperable. This condition requires entry into TS 3.8.1 Condition A for one required offsite circuit inoperable.

The time that alignment of a single 230 kV / 12 kV standby startup transformer (11 or 21) to provide power to both units is limited by the current TS 3.8.1 Condition A completion time to 72 hours.

Audit Question EEEB-03

Provide clarification for design success criteria (DSC) in Table E1-1 corresponding to the following TS conditions like DSC for TS LCO 3.8.1 (2):

a. TS LCO 3.8.1, Condition A – One required offsite circuit inoperable (e.g., 2 of 3 engineered safety feature (ESF) buses)

b. TS LCO 3.8.1, Condition B (One diesel generator (DG) inoperable)

c. TS LCO 3.8.1, Condition C (Two required offsite circuits inoperable)

d. TS LCO 3.8.1, Condition D (One required offsite circuit inoperable AND One DG inoperable)

PG&E Response to a-d:

TS 3.8.1 is for onsite and offsite AC power sources. As stated in Table E1-1 for TS 3.8.1, the Diablo Canyon design has two offsite circuits, three diesel generators, and two diesel fuel oil supply trains.

The Diablo Canyon TS 3.8.1 Bases states:

“The unit Class 1E AC Electrical Power Distribution System AC sources consist of offsite power sources (normal and alternate), and the onsite standby power sources (three diesel generators (DGs) for each unit). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The onsite Class 1E AC Distribution System for each unit is divided into three load groups so that the loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two offsite power sources and a single DG. …

Fuel oil is transferred from the storage tanks via the diesel fuel oil storage and transfer system to replenish the day tanks as required. The design incorporates sufficient redundancy so that a malfunction of either an active or a passive component will not impair the ability of the system to supply fuel oil. Two redundant fuel oil transfer pumps supply fuel oil to each DG day tank from either storage tank. One pump is adequate to supply the six DGs operating at full load.”

The offsite power sources are redundant, so the loss of either source does not cause a loss of offsite power to any of the three load groups. Similarly, the loss of any single DG assuming a loss of offsite power results in the loss of a single load group, which does not result in a loss of any safety function. The fuel oil supply system consists of two redundant trains, and a single train and associated pump is adequate to support all six diesel generators for both units.

The functions covered by TS 3.8.1, identified in Table E1-1, are support functions to supply power to ESF systems and provide fuel oil to the onsite DGs. For the fuel oil function, the design success criterion is one of two trains. For the ESF system power function, the offsite and onsite sources must be capable of automatically powering their associated ESF bus. One load group, associated with one of the three ESF busses, can be de-energized without a loss of any required ESF function. Therefore, the function of automatically powering two of three ESF busses can be achieved by either of the two offsite circuits, or by two of three DGs.

TS LCO 3.8.1 Condition A applies when one required offsite circuit is inoperable. In this Condition, the second offsite circuit is Operable to supply all three ESF busses, and each ESF bus has its associated EDG Operable. Therefore, no loss of function exists.

TS LCO 3.8.1 Condition B applies when one DG is inoperable. In this Condition, all three ESF busses have both offsite circuits Operable, and two of the three ESF busses have their associated DG Operable. Therefore, no loss of function exists.

TS LCO 3.8.1 Condition C applies when two required offsite circuits are inoperable. In this Condition, all three ESF busses have their associated EDG Operable. Therefore, no loss of function exists.

TS LCO 3.8.1 Condition D applies when one required offsite circuit and one of three DGs are inoperable. In this Condition, the second offsite circuit is Operable to supply all three ESF busses, and two of three ESF busses have their associated EDG Operable. Therefore, no loss of function exists.

e. TS LCO 3.8.4, Condition A (One battery charger inoperable)

f. TS LCO 3.8.4, Condition B (One battery inoperable)

g. TS LCO 3.8.4, Condition C (One DC electrical power subsystem inoperable for reasons other than Condition A or B)

PG&E Response to e-g:

TS 3.8.4 is for DC sources. As stated in Table E1-1 for TS 3.8.4, the Diablo Canyon design has three Class 1E DC subsystems.

The Diablo Canyon TS 3.8.4 Bases states:

“As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. …

The 125-Vdc- electrical power system consists of three independent Class 1E DC electrical power subsystems. Each subsystem consists of one 60-cell 125-Vdc battery (Batteries 11(21), 12 (22), and 13 (23)), the dedicated battery charger and backup charger for each battery, and all the associated switchgear, control equipment, and interconnecting cabling.”

The design success criterion is therefore two of three DC electrical power subsystems.

TS LCO 3.8.4 Condition A applies when one battery charger is inoperable. In this Condition, only one of the three DC electrical power subsystems is impacted; two of the three DC electrical power subsystems have their dedicated battery charger Operable. Therefore, no loss of function exists.

TS LCO 3.8.4 Condition B applies when one battery is inoperable. In this Condition, only one of the three DC electrical power subsystems is impacted; two of the three DC electrical power subsystems have their dedicated battery Operable. Therefore, no loss of function exists.

TS LCO 3.8.4 Condition C applies when one DC electrical power subsystem is inoperable. In this Condition, only one of the three DC electrical power subsystems is impacted; two of the three DC electrical power subsystems remain Operable with their dedicated battery and charger Operable. Therefore, no loss of function exists.

h. *TS LCO 3.8.9, Condition A (One AC electrical power distribution subsystem inoperable)*
i. *TS LCO 3.8.9, Condition B (One 120 VAC vital bus subsystem inoperable)*
j. *TS LCO 3.8.9, Condition C (One DC electrical power distribution subsystem inoperable)*

PG&E Response to h-j:

TS 3.8.9 is for Distribution Systems. As stated in Table E1-1 for TS 3.8.9, this TS covers all Class 1E AC, DC, and 120 VAC vital bus subsystems.

TS LCO 3.8.9 Condition A applies when one (of three) AC electrical power distribution subsystem is inoperable. The Diablo Canyon TS 3.8.9 Bases for Required Action A.1 states:

“With one required Class 1E AC electrical power subsystem inoperable, the remaining portions of the AC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure.”

Therefore, Condition A does not represent a loss of function.

TS LCO 3.8.9 Condition B applies when one (of four) 120 VAC vital bus subsystem is inoperable. The Diablo Canyon TS 3.8.9 Bases for Required Action B.1 states:

"With one 120-Vac Class 1E bus subsystem inoperable, the remaining OPERABLE 120-Vac Class 1E buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition."

Therefore, Condition B does not represent a loss of function.

TS LCO 3.8.9 Condition C applies when one (of three) DC electrical power distribution subsystem is inoperable. The Diablo Canyon TS 3.8.9 Bases for Required Action C.1 states:

"With one DC electrical power distribution subsystem inoperable, the remaining portions of the DC electrical power distribution subsystem are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure."

Therefore, Condition C does not represent a loss of function.

Audit Question EEEB-04

In reference to Table E1-2, provide the reasons why the "One AC electrical power distribution subsystems inoperable (TS LCO 3.8.9.A)" has a significantly lower RICT estimate (4.8 days) compared to the RICT estimate (20.6 days) for "One DC electrical power distribution subsystem inoperable" (TS LCO 3.8.9.C).

PG&E Response:

The one AC electrical power distribution RICT should not be significantly different from the one DC electrical power distribution RICT. There were additional impacts that were intended to be used for the DC electrical power distribution RICT that were left out of the sample calculation. These impacts involve the operator-action to recover breaker failures, which should be failed in the RICT calculation. Accounting for these additional impacts results in a RICT of approximately 4.2 days for one DC electrical power distribution. The RICT model will include these additional impacts.

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.4.9 Pressurizer Heaters	2 groups of heaters	No	(1) Maintain RCS subcooling margin	1 of 2 groups		Note 1
3.4.9.B	One required group of pressurizer heaters inoperable			Remaining one operable of two total required group of pressurizer heaters provides the function		
3.4.11 Pressurizer power operated relief valves (PORVs)	3 PORVs (2 Class I, 1 non-Class I) 3 PORV block valves	Yes	(1) Depressurize the RCS (2) Mitigate spurious operation of the safety injection system at power (3) No excessive seat leakage	(1) 1 of 2 Class I PORVs open (2) 1 of 2 Class I PORVs open (3) Associated block valve manually closed	(1) SAME or more restrictive (2) SAME (3) SAME	SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria, and in some cases are more restrictive when the PORVs are credited to mitigate some beyond design basis scenarios. The PRA also credits the non-Class I PORV if it is not failed; this is consistent with the TS bases which identifies that the non-Class I PORV can be used if it is available. See Note 9

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.4.11.B	One PORV inoperable			Remaining one or two operable of two total Class I PORVs provides the pressure relief functions		
3.4.11.C	One block valve inoperable			With inoperable block valve open, the associated PORV provides the isolation function and is capable of opening to provide the pressure relief function. With the inoperable block valve closed, the isolation function is satisfied, and the remaining one or two Class I PORVs with operable block valves provide the pressure relief function		
3.4.11.F	More than one inoperable PORV block valve			See Disposition		The RICT is only applied to Required Action F.3 after at least one block valve for a Class I PORV has been restored to operable status by Required Action F.2; change is required only to be consistent with Condition C for one block valve inoperable, Required Action C.2 to restore block valve to OPERABLE status, for which a RICT is applied

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.5.2 ECCS – Operating	2 Centrifugal charging (CH) pumps (high pressure) 2 SI pumps (intermediate pressure) 2 RHR pumps (low pressure) Associated piping, valves and heat exchangers	Yes	(1) Injection from the RWST into cold legs (2) Cold leg recirculation from the containment sumps (3) Hot leg recirculation from the containment sumps	(1) (a) 1 of 2 CH pumps for small/medium LOCA until the RCS is depressurized to allow injection from 1 of 2 SI pumps (b) 1 of 2 RHR pumps, SI pumps, and CH pumps for a large LOCA (c) 1 of 2 CH pumps for a steam generator tube rupture (SGTR) or main steam line break (MSLB) (2) 1 of 2 RHR pumps to supply other required ECCS pump suctions and injection into the RCS cold legs (3) 1 of 2 RHR pumps to supply other required ECCS pump suctions and injection into the RCS hot legs	(1) (a) small LOCA: 1 of 4 CH pumps or SI pumps into 3 cold legs; medium LOCA: 2 of 4 CH pumps or SI pumps (b) 1 of 2 RHR pumps (c) 1 of 4 ECCS CH pumps or SI pumps for SGTR and MSLB (Boration injection when reactor trip function fails) (2) SAME except injection into 2 cold legs (3) Not required	SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The PRA success criteria differ from the design basis in: (1) crediting the CCPs or SI pumps where the design basis requires one of each to function; (2) not requiring injection into all RCS loops; (3) only requiring boration mitigation for MSLB events when the reactor trip function fails; (4) not requiring hot leg recirculation. The success criteria in the PRA are based on plant-specific realistic analyses consistent with the PRA standards for capability category II. See Note 2.
3.5.2.A	One or more trains inoperable and 100% flow available			Per the Condition, 100% ECCS flow must be available in one or two trains; the operable equipment in one or two trains of two total provides the functions		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.6.2 Containment Air Locks	2 air locks (personnel and emergency)	No	(1) post-accident containment leakage within limits			SSCs for the containment air locks can be evaluated by a bounding assessment as permitted by NEI 06-09-A. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding for the risk on an inoperable air lock and can be used as a bounding surrogate. See Note 3.
3.6.2.C	One or more containment air locks inoperable			Required Action C.1 ensures containment leakage rate is evaluated; Required Action C.2 ensures one door in the affected containment air lock is closed; therefore, the isolation function is maintained for each inoperable containment air lock.		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.6.3 Containment Isolation Valves	2 active or passive isolation devices on each fluid penetration line	Yes	(1) Each containment penetration isolated within the time limits assumed in the safety analyses	(1) 1 of 2 isolation devices per penetration isolate within the required stroke time.	(1) SAME for all penetrations modeled in the PRA; all other penetrations were evaluated and determined not to be significant sources of fission product leakage and were screened out.	SSCs for those containment isolation valves not in the PRA model can be evaluated by a bounding assessment as permitted by NEI 06-09-A. The PRA model includes an event which involves a large, pre-existing containment leak; this is bounding for the risk on an inoperable isolation valve and can be used as a bounding surrogate. The PRA does not explicitly model the impact of excessive stroke times. This condition can be addressed in the RICT Program by conservatively assuming that the inoperable containment isolation valve cannot be closed if it is open. Otherwise, the success criteria in the PRA are consistent with the design basis criteria.
3.6.3.A	One or more penetrations with one of two isolation valves inoperable			For each affected penetration, remaining one operable of two total isolation valves provides the isolation function		
3.6.3.C	One or more penetrations with one of one isolation valve inoperable			For each affected penetration, the closed system provides the isolation function		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.6.6 Containment Spray (CS) and Cooling Systems	2 CS trains 5 containment fan cooling units (CFCU)	Yes	(1) Containment atmosphere cooling to limit post-accident pressure and temperature (2) Iodine removal to reduce the release of fission product radioactivity from containment to the environment	(1) 1 of 2 CS trains and 2 of 5 CFCUs (2) 1 of 2 CS trains	(1) SAME (2) Not modeled	The SSCs in the TS scope are modeled in the PRA. The iodine removal function of the CS trains is not required for mitigation of severe accidents and is not modeled. See Notes 4 and 5.
3.6.6.A	One CS train inoperable			Remaining one operable of two total CS trains provides the iodine removal function and CS portion of the heat removal function		
3.6.6.C	One CFCU system inoperable and a minimum of two CFCUs remain operable			Remaining two or three operable of five total CFCUs provide the CFCU portion of the heat removal function		
3.6.6.D	One CS train and one CFCU system inoperable with a minimum of two CFCUs remaining operable			Remaining one operable of two total CS trains provides the iodine removal function and CS portion of the heat removal function; remaining two or three operable of five total CFCUs provide the CFCU portion of the heat removal function.		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.2 Main Steam Isolation Valves (MSIVs)	4 MSIVs	Yes	(1) Isolate steam flow from the secondary side of the SGs following a high energy line break (HELB)	(1) MSIV on the affected steam line closes, or the remaining 3 MSIVs on unaffected steam lines close.	(1) SAME or more restrictive	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria for a HELB. See Note 6. The PRA also credits MSIV closure for isolation of a ruptured SG, and on 3 of 4 steam lines to prevent RCS overcooling in the event of a failure of the turbine trip function.
3.7.2.A	One MSIV inoperable (Mode 1)			Remaining three operable of four total MSIVs provide the isolation function		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.7.4 Atmospheric Dump Valves (ADVs)	4 ADV lines (one per steam generator, each with an ADV and associated block valve)	Yes	(1) Cool down the unit to RHR entry conditions, if the preferred heat sink via steam dump to the condenser is not available (2) Cool down the RCS following a SGTR to permit termination of primary to secondary break flow.	(1) 4 of 4 ADVs to cool down the unit at the design rate of 100°F per hour; 1 of 4 ADVs permits a 25°F per hour cooldown for a natural circulation cooldown event. (2) 3 of 4 ADVs on the intact steam generator lines.	(1) 1 of 4 ADVs (2) 1 of 4 ADVs on the intact steam generator lines	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA do not require the maximum rate cooldown capability to mitigate severe accidents, and therefore more realistic criteria are applicable consistent with the PRA standards for capability category II. See Note 7.
3.7.4.A	One required ADV line inoperable			Remaining three operable of four total required ADV lines provide the functions		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.7.5 Auxiliary Feedwater (AFW) System	2 motor-driven pumps and 1 turbine-driven pump	Yes	(1) Supply feedwater to the steam generators to remove decay heat	(1) 2 motor-driven pumps or 1 turbine-driven pump for the most limiting event (loss of main feedwater)	(1) 1 of 3 pumps	SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are based on a "better estimate" evaluation which demonstrates any one AFW pump can provide 100% of the feedwater flow required for removal of decay heat from the reactor. This is discussed in the plant-specific TS Bases. The use of more realistic success criteria is consistent with the PRA standards for capability category II.
3.7.5.A	Turbine driven AFW train inoperable due to one inoperable steam supply			Remaining two operable of two total motor-driven trains provide the function.		Turbine-driven pump may still be available using remaining one operable of two total steam supplies for events not impacting the affected steam generator.
3.7.5.B	One AFW train inoperable			Remaining two operable of three total AFW trains provide the function		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.5.C	Turbine driven AFW train inoperable due to one inoperable steam supply and one motor-driven AFW train inoperable			Remaining one operable of two total motor-driven AFW trains and the turbine-driven pump using remaining one operable of two total steam supplies provide the function		
3.7.7 Component Cooling Water (CCW) System	2 vital loops	Yes	(1) Heat sink for the removal of process and operating heat from safety-related components	(1) 1 of 2 vital loops with 2 of 3 CCW pumps and 1 of 2 heat exchangers	(1) SAME; successful isolation of unnecessary CCW heat loads is also credited and then only 1 of 3 CCW pumps is required	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria, but also include credit for operator action to isolate unnecessary CCW heat loads; in this case, only 1 of 3 CCW pumps is required.
3.7.7.A	One vital CCW loop inoperable			Remaining one operable of two total vital CCW loops provides the function		
3.7.8 Auxiliary Saltwater (ASW) System	2 trains	Yes	(1) Heat sink for the removal of process and operating heat from the CCW system	(1) 1 of 2 trains	(1) SAME; cross tie to the unaffected unit is also credited.	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.7.8.A	One ASW train inoperable			Remaining one operable of two total ASW trains provides the function		
3.8.1 Alternating Current (AC) Sources - Operating	2 offsite circuits 3 diesel generators (DG) 2 supply trains of the diesel fuel oil (DFO) transfer system	Yes	(1) Source of power to the engineered safety features (ESF) systems (2) Source of fuel oil to the DGs	(1) Automatically power the associated ESF busses by either 1 of 2 offsite circuits or any 2 of 3 DGs (2) 1 of 2 trains	(1) SAME (2) SAME	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. See Note 8. The success criteria in the PRA are consistent with the design basis criteria.
3.8.1.A	One offsite circuit inoperable			Remaining one operable of two total offsite circuits, or two of three operable DGs, provide the power source function		
3.8.1.B	One DG inoperable			Either one of two operable offsite circuits, or the remaining two operable of three total DGs, provide the power source function		
3.8.1.C	Two offsite circuits inoperable			Any two of three operable DGs provide the power source function		
3.8.1.D	One offsite circuit and one DG inoperable			Remaining one operable of two total offsite circuits, or remaining two operable of three total DGs, provide the power source function		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.8.1.F	One DFO supply train inoperable			Remaining one operable of two total DFO supply trains provides the fuel oil source function		
3.8.4 Direct Current (DC) Sources - Operating	3 Class 1E DC subsystems	Yes	(1) Provide control power to the AC emergency power system, motive and control power to selected safety related equipment and backup 120 VAC vital bus power	(1) Aligned to provide power to the associated equipment from the battery and associated charger (2 of 3 subsystems required)	(1) SAME	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.4.A	One battery charger inoperable			Remaining two operable of three total Class 1E DC subsystems with operable battery chargers provide the function		
3.8.4.B	One battery inoperable			Remaining two operable of three total Class 1E DC subsystems with operable batteries provide the function		
3.8.4.C	One subsystem inoperable			Remaining two operable of three total Class 1E DC subsystems provide the function		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions

TS LCO/Condition	SSCs Covered by TS LCO/ Condition	SSCs Modeled in the PRA	Function Covered by the TS LCO/ Condition	Design Success Criteria	PRA Success Criteria	Disposition
3.8.7 Inverters - Operating	4 Class 1E inverters	Yes	(1) Provide uninterruptible power to the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS)	(1) Align to the associated 120 VAC vital bus, with input power aligned to its associated battery (3 of 4 inverters required)	(1) SAME	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.7.A	One inverter inoperable			Remaining three operable of four total inverters provide the function		
3.8.9 Distribution Systems - Operating	Class 1E AC, DC, and 120 volt vital alternating current (VAC) vital bus electrical power distribution subsystems	Yes	(1) Provide necessary power to the ESF systems	(1) Align to provide power to the busses (2 of 3 Class 1E AC; 2 of 3 Class 1E DC, and 3 of 4 Class 1E 120 VAC busses)	(1) SAME	The SSCs are modeled consistent with the TS scope and can be directly evaluated using the CRMP. The success criteria in the PRA are consistent with the design basis criteria.
3.8.9.A	One AC electrical power distribution subsystem inoperable			Remaining two operable of three total AC electrical power distribution subsystems provide the function		
3.8.9.B	One 120 VAC vital bus subsystem inoperable			Remaining three operable of four total 120 VAC vital bus subsystems provide the function		

Table E1-1: In Scope TS/LCO Conditions to the Corresponding PRA Functions						
TS LCO/Condition	**SSCs Covered by TS LCO/ Condition**	**SSCs Modeled in the PRA**	**Function Covered by the TS LCO/ Condition**	**Design Success Criteria**	**PRA Success Criteria**	**Disposition**
3.8.9.C	One DC electrical power distribution subsystem inoperable			Remaining two operable of three total DC electrical power distribution subsystems provide the function		

Note 1: The pressurizer heaters will be evaluated for the RICT Program by a bounding assessment as permitted by NEI 06-09-A. Inoperability of the pressurizer heaters will be conservatively bounded by assuming an increase in the frequency of a reactor trip initiating event by a factor of 10; this reflects the adverse impact on pressure control due to inoperable pressurizer heaters. This is conservative since the redundant pressurizer heater group of TS 3.4.9 must be operable, and additional pressurizer heater groups not required by TS 3.4.9 would typically be available. The safe shutdown of the plant after a reactor trip without pressurizer heaters available is addressed by plant procedures. This surrogate is consistent with recently approved TSTF-505 applications for plants similar in design to Diablo Canyon. This note satisfies the requirements of Table 1 of TSTF-505 Revision 2.

Note 2: TS 3.5.2 Condition A explicitly requires 100% of the ECCS flow equivalent to a single operable ECCS train. Therefore, TS 3.5.2 Condition A meets the requirements for inclusion in the RICT Program. This note satisfies the requirements of Table 1 of TSTF-505 Revision 2.

Note 3: The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness is essential for maintaining the containment leakage rate within limit in the event of a design basis accident (DBA). Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the safety analyses. The DBA that results in a release of radioactive material within containment is the loss of coolant accident. In the analysis of this accident, it is assumed that containment is operable such that release of fission products to the environment is controlled by the rate of containment leakage.

Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits. Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour. These Completion Times are well within the existing 24-hour Completion Time for Required Action C.3. Thus, the containment function is maintained at the point in time when a RICT would be entered.

Therefore, TS 3.6.2 Condition C meets the requirements for inclusion in the RICT Program.

This note satisfies the requirements of Table 1 of TSTF-505 Revision 2.

Note 4: The fire PRA does not credit the CS system nor the CFCUs.

Note 5: The CS and Containment Cooling systems provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA, to within limits. The CS system is modeled in the Diablo Canyon PRA, with the same success criteria as the design success criteria (i.e., one of two CS subsystems).

Therefore, TS 3.6.6 Condition A meets the requirements for inclusion in the RICT Program.

This note satisfies the requirements of Table 1 of TSTF-505 Revision 2.

Note 6: The design of the MSIVs precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand). With one MSIV inoperable in Condition A, the steam line isolation function of TS 3.7.2 is met by the remaining three operable MSIVs on the other three steam generators to prevent blowdown of more than one steam generator. This note satisfies the requirements of Table 1 of TSTF-505 Revision 2.

Note 7: Condition B of TS 3.7.4 for two inoperable ADVs is not in the proposed Diablo Canyon TSTF-505 scope; therefore, no additional justification needs to be provided per Table 1 of TSTF-505 Revision 2.

Note 8: The 500 kV offsite circuits are only credited for the mitigation of internal events.

Note 9: The PORV success criteria for beyond design basis scenarios corresponds to loss of steam generator cooling events where Bleed and Feed cooling is initiated. Bleed and Feed cooling through the PORVs is successful if 2 out of 3 PORVs open. PORV PCV-474 is not safety related and does not have a backup air accumulator, thus it is only credited for those initiating events where instrument air is credited.

Table A2-1 - Disposition of SPRA Updates Since Peer Review

Table A2-1 - Disposition of SPRA Updates Since Peer Review		
Update Description	**Classification**	**Classification Rationale**
Fragility updates: Minor updates to fragilities for the diesel fuel oil pump, EDG day tanks, local starter boards, 480V auxiliary relay panel and piping components.	Maintenance	(1) No new methodology is introduced - Fragility update utilized peer reviewed methodology and does not constitute a new method. (2) The scope of the PRA is NOT changed as the change only updates fragility parameters (3) The minor change to these fragilities does not impact the significant accident sequences or the significant accident progression sequences. This resolution/change is classified as Maintenance.
Changes related to streamlining model and quantification time improvement: Streamlined event tree modeling for seismically induced fires (Incorporated conditional ignition frequency into non-vital 480V modeling). Removed Bus F EDG potential transformer seismic impacts. This had a negligible contribution to risk and was removed to expedite quantification. Removed SCI to expedite quantification. This	Maintenance	(1) No new methodology is introduced - Streamlining process removed extraneous model components to expedite quantification (2) The scope of the PRA is NOT changed by improving quantification efficiency (3) The minor change to these fragilities does not impact the significant accident sequences or the significant accident progression sequences. This resolution/change is classified as Maintenance.

Table A2-1 - Disposition of SPRA Updates Since Peer Review		
Update Description	**Classification**	**Classification Rationale**
top event does not impact SCDF or SLERF in any way.		
Fragility Impact Grouping Change: Added CFCU fragility into grouping for containment bypass. This models CFCU failure as causing a CCW piping rupture and impacts containment isolation	Maintenance	(1) No new methodology is introduced - Represents a change to fragility impact grouping (2) The scope of the PRA is NOT changed by updating fragility impact grouping (3) This change does not impact the significant accident sequences or the significant accident progression sequences. This resolution/change is classified as Maintenance.
Decorrelation of some containment isolation valves Changes related to decorrelating containment isolation valves that are located at different elevations. Removing excessive conservatism.	Maintenance	(1) No new methodology is introduced - Removed excessive modeling conservatism (2) The scope of the PRA is NOT changed by removing correlation (3) This change does not impact the significant accident sequences or the significant accident progression sequences. This resolution/change is classified as Maintenance.

Table A2-1 - Disposition of SPRA Updates Since Peer Review		
Update Description	**Classification**	**Classification Rationale**
FLEX strategy actions fragility impacts updated: Added hot shutdown panel fragility to seismic pre tree. This new event impacts FLEX action feasibility. Scenarios where control room vertical boards seismically fail are now recoverable using FLEX actions as long as the HSDP connections are available (based on status of SHSP top event).	Maintenance	(1) No new methodology is introduced - Change ensures that recovery actions are appropriately credited based on availability of required equipment. (2) The scope of the PRA is NOT changed by correcting fragility impacts to required equipment (3) This change does not impact the significant accident sequences or the significant accident progression sequences. This resolution/change is classified as Maintenance.
General PRA model logic update: The SSBO (seismic SBO) and TDP (turbine driven AFW pump) event tree logic macros were updated to account for changes in top event nomenclature.	Maintenance	(1) No new methodology is introduced - Minor editorial change to model logic due to top event nomenclature change. (2) The scope of the PRA is NOT changed by including updated top event names. (3) This change does not impact the significant accident sequences or the significant accident progression sequences. This resolution/change is classified as Maintenance.
Corrected dependency for AOV CIVs: Failure of vital DC now results in these valves failing closed (only for AOVs that fail closed when deenergized).	Maintenance	(1) No new methodology is introduced - Model corrected to include the fail closed failure state for AOVs. (2) The scope of the PRA is NOT changed by changing the dependency between DC power and failure state for AOVs. (3) This change does not impact the significant accident sequences or the significant accident progression sequences.

Table A2-1 - Disposition of SPRA Updates Since Peer Review		
Update Description	**Classification**	**Classification Rationale**
		This resolution/change is classified as Maintenance.

Table A2-2 ASME SPRA Standard Gap Assessment

Diablo Canyon performed a gap assessment between Addendum A and B consistent with that from Southern Nuclear Operating Company, Inc., letter to NRC, NL-17-1201, “Vogtle Electric Generating Plant Units 1 and 2 Response to Supplemental Information Needed for Acceptance of Systematic Risk-Informed Assessment of Debris Technical Report,” dated July 11, 2017 (ML17192A245). NRC acceptance of the assessment was documented in a letter to Southern Nuclear Operating Company, Inc., “Vogtle Electric Generating Plant, Units 1 and 2 – Issuance of Amendments Regarding Application of Seismic Probabilistic Risk Assessment Into the Previously Approved 10 CFR 50.69 Categorization Process (EPID L-2017-LLA-0248),” dated August 10, 2018 (ML18180A062). In the Vogtle Assessment, all but six of the Addendum B SRs have been shown to either be equal to the corresponding Addendum A SRs or have been shown to envelop the corresponding Addendum A SRs. The remaining six SRs (SHA-B3, SHA-C3, SFR-C3, SFR-C6, SFR-G3, and SPR-B1) were assessed and the DCPP SPRA was shown to conform to these Addendum A SRs (See table below).

<table>
<tr><th colspan="6">Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B</th></tr>
<tr><th>SR</th><th>Standard Rev.</th><th>Capability Category I</th><th>Capability Category II</th><th>Capability Category III</th><th>Basis for Assessment</th></tr>
<tr><td rowspan="2">SHA-B3</td><td>ASME/ANS RA-Sa–2009</td><td>[Assessment focused on CC II/III only]</td><td colspan="2">As a part of data collection, COMPILE a catalog of historically reported, geologically identified, and instrumentally recorded earthquakes. USE reference [5-30] requirements or equivalent.</td><td rowspan="2">DCPP Conforms to Addendum A (CC-II/III)

DCPP SPRA meets the Addendum B SR requirement at Cat III. Addendum B has an option to “AUGMENT an existing catalog” instead of compiling the catalog.

Earthquake catalogs and geologic data were compiled as part of the Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 process to support the Seismic Source Characterization (SSC) for Diablo Canyon. Alternative catalogs prepared by others were also considered and, as appropriate, used. Catalogs included historical and instrumental data. Geologic data were used to characterize fault slip rates, as appropriate.

Additional descriptive wording was also added to Addendum B that does not impact the requirement.

Based on this, the DCPP seismic source characterization conforms to Addendum A.</td></tr>
<tr><td>ASME/ANS RA-Sb–2013</td><td colspan="2">[Assessment focused on CC III only]</td><td>As a part of data collection, COMPILE a catalog (or AUGMENT an existing catalog) of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes reported through geological investigations. USE reference [5-30] requirements or equivalent.</td></tr>
</table>

Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment
SHA-C3	ASME/ANS RA-Sa–2009	[Assessment focused on CC II/III only]	The seismic sources are characterized by source location and geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations.		**DCPP Conforms to Addendum A (CC-II/III)** Addendum B includes the language, “where significant” in the CC II/III requirement as well as additional clarification. A SSHAC Level 3 analysis was used to define the seismic hazard and included characterization of epistemic uncertainties and aleatory variability in seismic sources and ground motion models. The overall Seismic Source Characterization model included, for “Primary and Connected” faults, assessments of uncertainty in fault geometry, fault slip rate, how different faults or fault segments might rupture together, slip rate allocation among faults, magnitude distribution, and recurrence, including its possible time-dependency. Sensitivity analyses were used to inform the degree to which uncertainty was characterized. Epistemic uncertainty was represented using a logic tree structure as documented in the DCPP SSHAC SSC Report.
	ASME/ANS RA-Sb–2013	[Assessment focused on CC II/III only]	The seismic sources are characterized by alternative source representation and source geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations, where significant.		

Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment
					Based on this, the DCPP seismic source characterization conforms to Addendum A.
SFR-C3	ASME/ANS RA-Sa–2009	[Assessment focused on CC III only]		Addressed in Requirement SFR-C2	**DCPP Conforms to Addendum A** The change from Addendum A to Addendum B (for CC-I/II) involved the deletion of the word "design" from "existing design response analysis." In the DCPP SPRA peer review, this SR was "N/A". The CC I/II assessments were addressed in SFR-C2. Probabilistic response analysis was performed for the Containment Structure, Auxiliary Building and Turbine Buildings, using the guidance provided in NUREG/CR-2015 and ASCE 4-13. New finite element structural models were developed and used in the development of structural responses. Latin hypercube sampling
	ASME/ANS RA-Sb–2013	[Assessment focused on CC III only]		Addressed in Requirement SFR-C2	

Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B					
SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment
					approach was used with 30 trials, the minimum number specified in ASCE-4 to achieve stable responses. The variables included: structural stiffness, damping, soil/rock profiles and ground motions (time histories). The median 50% and 84% non- exceedance probability (NEP) spectral accelerations were determined. Forces and moments for the determination of fragilities of major structural components were also based on the probabilistic seismic response analyses. The peer review team also concluded that; considering the use of plant specific PSHA based seismic response spectra, development of new 3D finite element building models, use of probabilistic seismic response analysis, detailed walk downs, and the use of separation of variables approach to derive fragility values for all but two (2) SSCs of the SEL (other than rugged and robust components), the fragility values used in the S-PRA are indeed realistic.

Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B

SR	Standard Rev.	Capability Category I	Capability Category II	Capability Category III	Basis for Assessment
SFR-C6	ASME/ANS RA-Sa–2009	[Assessment focused on CC III only]		Addressed in Requirement SFR-C2	**DCPP Conforms to Addendum A** In the DCPP SPRA peer review, this SR was "N/A". The CCI/II assessments were addressed in SFR-C2. Soil-structure interaction (SSI) analysis was performed of the Containment Structure, Auxiliary Building and Turbine Buildings. For the Intake Structure and Outdoor Tanks, the effects of SSI were judged to be insignificant. A probabilistic analysis approach was used using 30 random soil profiles (same as those used in the PSHA) and 30 time histories. Spatial incoherency of ground motion was judged to be insignificant given the site conditions, and frequency range of interest (1 to 10 Hz).
	ASME/ANS RA-Sb–2013	[Assessment focused on CC III only]		Addressed in Requirement SFR-C2	
SFR-G3	ASME/ANS RA-Sa–2009	**DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic fragility analysis.**			**DCPP Conforms to Addendum A** Addendum B deleted this SR however, the DCPP seismic PRA documentation includes a significant amount of detail concerning analysis assumptions and associated sources of modeling
	ASME/ANS RA-Sb–2013	**Deleted.**			

Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment**
					uncertainties. An extensive set of sensitivity analyses were used to assess the impact of these modeling uncertainties. As a result, the DCPP SPRA conforms to Addendum A.
SPR-B1	ASME/ANS RA-Sa–2009	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. **DEVELOP a defined** basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g) use of expert judgment **When the Part 2 requirements are used, FOLLOW the Capability Category designations in Part 2, and for consistency USE the same Capability Category in this analysis.**			**DCPP Conforms to Addendum A** Addendum B removed the last sentence of this SR in response to an EPRI 2011 comment on the Addendum B ballot. The last sentence was removed in Addendum B because it was determined to be confusing as well as containing inappropriate specificity to require all new aspects in the SPRA to meet the exact same CCs of Part 2 SRs. In addition, Addendum B changed the action verb to be consistent with accepted verb usage across SRs. The

Table A2-2: Comparison of Supporting Requirements of Addendum A and Addendum B					
SR	**Standard Rev.**	**Capability Category I**	**Capability Category II**	**Capability Category III**	**Basis for Assessment**
	ASME/ANS RA-Sb–2013	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. **SPECIFY** a basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are (a) initiating-event analysis (b) accident-sequence analysis (c) success-criteria analysis (d) systems analysis (e) data analysis (f) human-reliability analysis (g)use of expert judgment			Addendum B SR requirement clarifications are appropriate. Regardless, the DCPP 1&2 SPRA builds upon the internal events PRA and uses the same general methodologies as used for Part 2 where applicable and the same Capability Category designations; therefore, the DCPP SPRA conforms to Addendum A.

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. More than one block valve inoperable.	------------------NOTE------------------ Required Actions do not apply when block valve is inoperable solely as result of complying with Required Actions B.2 or E.3. ---	
	F.1 Place associated PORVs in manual control.	1 hour
	AND	
	F.2 Restore one block valve for a Class I PORV to OPERABLE status.	2 hours
	AND	
	F.3 Restore remaining block valve for a Class I PORV to OPERABLE status.	72 hours OR In accordance with the RICT Program
	OR	
	F.4 If the remaining block valve is associated with the non-Class I PORV, close the block valve and remove its power.	72 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Initiate action to restore block valve(s) to OPERABLE status.	Immediately
	AND	
	G.2 Be in MODE 3.	6 hours
	AND	
	G.3 Be in MODE 4.	12 hours