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PG&E Letter DCL-16-046

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

10 CFR 50.59

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Summary Report of 10 CFR 50.59, "Changes, Tests, and Experiments," for the Period of January 1, 2014, through December 31, 2015

Dear Commissioners and Staff:

Pursuant to 10 CFR 50.59, "Changes, tests, and experiments," Pacific Gas and Electric Company (PG&E) is enclosing the 10 CFR 50.59 Summary Report for Diablo Canyon Power Plant, Units 1 and 2, for the period of January 1, 2014, through December 31, 2015. In accordance with 10 CFR 50.59(d)(2), the Enclosure provides a brief description and a summary of the evaluation for the changes, tests, and experiments performed during this period.

Evaluations performed in accordance with 10 CFR 50.59 are performed as part of PG&E's licensing basis impact evaluation (LBIE) process. Since the LBIE process is used to perform reviews for compliance with regulations other than 10 CFR 50.59, some LBIEs do not include a 10 CFR 50.59 evaluation; therefore, they are not included in this report.

The Plant Staff Review Committee has reviewed the referenced LBIEs and concurred with the determination in each regarding prior NRC approval as specified in the summaries.

PG&E makes no new or revised regulatory commitments (as defined by NEI 99-04) in this submittal.

If you have any questions or require additional information, please contact Mr. Hossein Hamzehee at (805) 545-4720.

Sincerely,

JAN NIMICK
FOR
JAMES WELSCH

James M. Welsch

dqmg/6192/50623971
Enclosure
cc: Diablo Distribution
cc/enc: Marc L. Dapas, NRC Region IV Administrator
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SUMMARY REPORT OF 10 CFR 50.59, "CHANGES, TESTS, AND EXPERIMENTS," for the Period January 1, 2014, through December 31, 2015

Pacific Gas and Electric Company
Diablo Canyon Power Plant, Units 1 and 2
Docket Nos. 50-275 and 50-323

Index of Licensing Basis Impact Evaluations (LBIEs)

* Proposed activity represents a re-evaluation (e.g., LBIEs performed by the Licensing Basis Verification Project) of a legacy item to verify consistency with current 10 CFR 50.59 standards.

13-032 Pressure Temperature Limits Report Revision, Units 1 and 2	
Reference Document No.:	PTLR-1, Revision 13
Reference Document Title:	Pressure and Temperature Limits Report (PTLR)
Activity Description:	
The proposed change revises PTLR-1, Table 2.2-2, to add a note that allows a special administrative exception to the Low Temperature Overpressure Protection (LTOP) requirement to establish an open reactor coolant system (RCS) vent when RCS temperature is less than or equal to 96°F. The note allows reducing this value to greater than 91°F when performing the RCS vacuum refill procedure and when RCS level is less than or equal to 123 ft.	
Summary of Evaluation:	
There has been no change in LTOP system operation such that all LTOP structures, systems, and components will continue to perform in the same manner as described in the Updated Final Safety Analysis Report. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

14-009 Revise Safety Injection (SI) Load Sequence Timer Settings, Unit 2	
Reference Document No.:	DDP 1000024983, Revision 0
Reference Document Title:	SI Load Sequence Timer Settings
Activity Description:	
Containment fan cooler unit (CFCU), component cooling water pump, auxiliary saltwater pump, and containment spray pump SI load sequence timer settings are being changed to allow time for any CFCUs running in high speed (1200 rpm) to coast down to less than 600 rpm before being restarted on low speed during a SI.	
Summary of Evaluation:	
The 10 CFR 50.59 Evaluation concludes that changing the time settings for the SI timers has no discernable effect on the likelihood of timer malfunction. The diesel generator loading remains within UFSAR-described diesel generator capacity. There is no discernable change in accident consequences due to the longer start delay times. The containment integrity analyses determine that containment design basis pressure limit is not exceeded by the proposed activity. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

14-015 Install RCP Shut Down Seal, Unit 1	
Reference Document No.:	DDP 1000000495, Revision 0
Reference Document Title:	Install Reactor Coolant Pump (RCP) Shut Down Seal

Activity Description:

The RCP Seal Number 1 insert is being replaced with a third-generation Westinghouse SHIELD® Passive Thermal Shut Down Seal (SDS) on all Unit 1 RCPs. The RCP shaft sleeve is also replaced with a sleeve that has a surface machined for better sealing with the SDS.

This design change supports the current PG&E strategy in the License Amendment Request that implements National Fire Protection Association (NFPA) Standard 805 (PG&E Letter DCL-13-065, "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," dated June 26, 2013). Specifically, PG&E committed to a modification to reduce the risk of a loss-of-coolant accident resulting from a loss of RCP seal cooling.

This design change also supports the current PG&E strategy in the FLEX Overall Integrated Plan (PG&E Letter DCL-13-007, "Pacific Gas and Electric Company's Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 27, 2013). Specifically, PG&E committed to install low-leakage RCP seals prior to startup from Unit 1 refueling outage 19 to reduce the potential seal leakage.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation has determined the potential for inadvertent SDS actuation has no discernable effect on the frequency of UFSAR-described RCP loss-of-flow as an accident initiator. The predicted likelihood of an SDS malfunction is negligible compared to the likelihood of UFSAR-evaluated RCP malfunctions. Inadvertent SDS deployment will not significantly impact the rotation of the RCPs or leakage from the existing seals.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

14-017* Seismic Trip Upgrade, Units 1 and 2	
Reference Document No.:	DCP J-41746 and J-43064, Revisions 1 and 4, respectively
Reference Document Title:	"Seismic Trip System Upgrade - Local Panels and Associated Circuits" and "Seismic Trip System Upgrade - SSPS," respectively

Activity Description:

This 10 CFR 50.59 Evaluation addresses a modification to the seismic reactor trip system and the resulting impact on compliance with Regulatory Guide (RG) 1.75, Revision 1. The original system consisted of three seismic sensor modules with outputs wired in two trains to relay panels that provided 2-of-3 logic, and the logic outputs were wired to the shunt trip coils of the reactor trip switchgear. The modification eliminated the relay panel, connecting the sensor output through a bypass switch panel (to facilitate testing) to the solid state protection system (SSPS) for the trip logic and additional output signals for status lamps, augmented annunciation, and the plant process computer.

Summary of Evaluation:

The sensor and train redundancy and use of 2-of-3 logic are unchanged by the modification. Added output signals for status indication and augmented annunciation are isolated consistent with RG 1.75. The addition of the bypass panel, partial trip status, and annunciation is aimed at reducing the likelihood of inadvertent trips.

The Reactor Trip System (including the SSPS) is designed to Institute of Electrical and Electronics Engineers Standard 279-1971 requirements, featuring redundancy and diversity. The design has been evaluated in detail, tested with respect to common mode failure, and is approved by the NRC. The SSPS changes have been made consistent with the approved design requirements. Because the design meets alternate criteria approved by the NRC for Diablo Canyon Power Plant, the change is an accepted alternative to RG 1.75 conformance.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

14-018* Low Pressure (LP) Rotor Replace – Missile Analysis, Units 1 and 2

Reference Document No.:	DCP M-49665/M-50665, Revision 0
Reference Document Title:	Units 1 and 2 Replacement of LP Turbine

Activity Description:

The design overspeed missile generation probability analysis of record (AOR) for the original LP turbine uses a Westinghouse methodology applicable to its "shrunk on" disc design. The LP rotor replacement modification required the turbine missile generation probability to be reanalyzed by the supplier, Alstom, for its welded integral disc design.

The AOR determined a missile generation probability for the original LP turbine based on periodic inspections of turbine discs at 5-year intervals. The Alstom analysis at the time of the LP rotor replacement determined a missile generation probability based on periodic inspections of turbine discs at 10-year intervals. Subsequently, the inspection interval was changed from 10 years to the vendor recommended interval of 100,000 equivalent operating hours, or approximately 12 years.

This 50.59 Evaluation was performed to evaluate the changes in inspection interval with respect to the turbine missile analysis methodology and associated inspection interval.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation has determined that the activity does not result in a departure from a method of evaluation described in the UFSAR because the alternative method has been approved by the NRC for the intended application. The criteria contained in NEI 96-07, Revision 1, Section 4.3.8.2, relating to the use/application of an alternative method of evaluation have been satisfied, including a thorough understanding of the method of evaluation, the terms of the existing application, and the conditions/limitations on its use.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

14-019* Low Pressure (LP) Rotor Replacement Digital Upgrades, Units 1 and 2	
Reference Document No.:	DCP M-49665/M-50665, Revision 0
Reference Document Title:	Units 1 and 2 Replacement of LP Turbine
Activity Description:	
The replacement of the LP turbine rotor involved upgrades to the digital electro-hydraulic (DEH) control system for the main turbine. The upgrades utilized software capabilities of the existing DEH to: replace/augment mechanical turbine trips used for the original LP turbine with pressure transmitters and 2-out-of-3 digital trip circuit logic; provide for a variable setpoint to meet design requirements of the replacement turbine rotor; revise the turbine runback rate to accommodate the small increase in full load; provide other logic changes needed to support operation with the replacement rotor; and provide associated human-machine interface changes (i.e., to screen displays).	
Summary of Evaluation:	
The 10 CFR 50.59 Evaluation has determined that each of the DEH changes for turbine trip functions and design features of the replacement LP rotor demonstrates the changes are acceptable and do not require prior NRC approval.	

14-026 Revise Safety Injection (SI) Load Sequence Timer Settings, Unit 1	
Reference Document No.:	DDP 1000025064, Revision 0
Reference Document Title:	Unit 1 SI Load Sequence Timers
Activity Description:	
Containment fan cooler unit (CFCU), component cooling water pump, auxiliary saltwater pump, and containment spray pump SI load sequence timer settings are being changed to allow time for any CFCUs running in high speed (1200 rpm) to coast down to less than 600 rpm before being restarted on low speed during a SI.	
Summary of Evaluation:	
The 10 CFR 50.59 Evaluation concludes that changing the time settings for the SI timers has no discernable effect on the likelihood of timer malfunction. The diesel generator loading remains within UFSAR-described diesel generator capacity. There is no discernable change in accident consequences due to the longer start delay times. The containment integrity analyses determine that containment design basis pressure limit is not exceeded by the proposed activity. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-002* **Reactor Coolant Pressure Boundary (RCPB) analysis using ANSYS, Units 1 and 2**	
Reference Document No.:	SAPN DA 50678458
Reference Document Title:	ANSYS Inadequately Justified Under Prior 10 CFR 50.59
Activity Description:	
The activity involves changing the computer program used for the structural analysis of RCPB components, structures, and supports, and reactor vessel internals to ANSYS. ANSYS was used to evaluate stresses in the replacement reactor vessel closure head (RVCH), replacement steam generators (RSGs), and integrated head assembly (IHA), as well as to evaluate the impact of the replacement RVCH, RSGs, and IHA on the reactor vessel and internals.	
Summary of Evaluation:	
The 10 CFR 50.59 Evaluation has determined the activity does not result in a departure from a method of evaluation described in the UFSAR because the alternative method has been approved by the NRC for the intended application. The criteria relating to the use/application of an alternative method of evaluation, as described in NEI 96-07, Revision 1, Section 4.3.8.2, have been satisfied. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-003* **Exchange Primary and Backup Reactor Protection Trips, Units 1 and 2**	
Reference Document No.:	Updated Final Safety Analysis Report (UFSAR) Section 15.3.4
Reference Document Title:	Complete Loss of Forced Reactor Coolant Flow

Activity Description:

The activity consists of changing the primary protection for a complete-loss-of-flow accident (CLOFA) from 12 kV bus undervoltage (UV)/underfrequency (UF) reactor trip to a reactor coolant loop low flow trip. The 12 kV UV/UF trip will be characterized as backup protection. This is based on a revision to the analysis of record, which credits the Design Class I reactor coolant low-flow reactor trip instead of the Design Class II UV and UF reactor trips for providing primary protection during a CLOFA.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation concludes that changing the primary protection for a CLOFA from 12 kV bus UV/UF reactor trip to a reactor coolant loop low-flow trip does not involve any modifications to the plant's electrical supplies, the reactor coolant pumps, or the reactor trip system. No new failure modes or system interactions are introduced and there are no changes in the operation or performance of any plant systems.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

15-004 Establish 76°F as Design Basis Outdoor Ambient Temperature, Units 1 and 2	
Reference Document No.:	Updated Final Safety Analysis Report (UFSAR) Section 9.4
Reference Document Title:	Heating, Ventilation, and Air Conditioning (HVAC) Systems
Activity Description:	
The proposed activity consists of changing the current licensing basis to establish 76°F as the design basis outdoor ambient temperature for HVAC, replacing the original design value of 82°F and an estimated maximum temperature of 90°F.	
Summary of Evaluation:	
This 10 CFR 50.59 Evaluation has determined that the changes in the calculation of the design basis outdoor ambient temperature for HVAC systems do not involve a departure from a method of evaluation described in the UFSAR. The criteria relating to changing one or more elements of a method of evaluation, as described in NEI 96-07, Revision 1, Section 4.3.8.1, have been satisfied. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-006 Spurious Safety Injection (SI) for Pressurizer Filling Reanalysis, Units 1 and 2

Reference Document No.:	CN-TA-12-67, Revision 0
Reference Document Title:	Westinghouse Calculation, Spurious SI Analysis for Pressurizer Filling

Activity Description:

The spurious safety injection accident for the pressurizer filling concern (Updated Final Safety Analysis Report (UFSAR), Section 15.2.15.3) has been reanalyzed with respect to liquid relief through the pressurizer safety valves.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation concludes there is no more than a minimal increase in the frequency of occurrence, likelihood of malfunction, or consequences of accidents/malfunctions; no possibility of creating accidents of a different type or malfunctions with a different result; no affect on design basis limits for fission product barriers; and no departure from a method of evaluation described in the UFSAR.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

15-008* Replace Solid State Protection System (SSPS) Printed Circuit Boards, Units 1 and 2	
Reference Document No.:	RPE 8*4616, Revision 0
Reference Document Title:	Replace SSPS Circuit Boards
Activity Description:	
Replacement Parts Evaluation (RPE) 8*4616 allows replacement of Westinghouse SSPS original design printed circuit boards (PCBs) with the Westinghouse new design of PCBs in SSPS (Train A and B).	
Summary of Evaluation:	
The new design of SSPS PCBs use complex programmable logic device technology as a replacement for the Motorola high threshold logic (MHTL) technology that is used in the original SSPS PCBs. MHTL-based PCBs are obsolete and can no longer be manufactured. The replacement PCBs have undergone extensive review per WCAP-17867-P-A, Revision 1. The NRC has issued a Final Safety Evaluation for the replacement PCBs. The 10 CFR 50.59 Evaluation results are consistent with Westinghouse guidance provided by the Pressurized Water Reactor Owners Group (PWROG) in Enclosure 1 to Letter OG-14-409 dated November 10, 2014. The 10 CFR 50.59 Evaluation concludes that prior NRC approval is not required.	

15-015 Install Reactor Coolant Pump (RCP) Shut Down Seal, Unit 2	
Reference Document No.:	DDP 1000024851, Revision 0
Reference Document Title:	Unit 2 RCP Shutdown Seal

Activity Description:

The RCP Seal Number 1 insert is being replaced with a third-generation Westinghouse SHIELD® Passive Thermal Shut Down Seal (SDS) on all Unit 2 RCPs. The RCP shaft sleeve is also replaced with a sleeve that has a surface machined for better sealing with the SDS.

This design change supports the current PG&E strategy in the NFPA 805 License Amendment Request (PG&E Letter DCL-13-065, "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," dated June 26, 2013). Specifically, PG&E committed to a modification to reduce the risk of a loss-of-coolant accident resulting from a loss of RCP seal cooling.

This design change also supports the current PG&E strategy in the FLEX Overall Integrated Plan (PG&E Letter DCL-13-007, "Pacific Gas and Electric Company's Overall Integrated Plan in Response to March 12, 2012, Commission Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events (Order Number EA-12-049)," dated February 27, 2013). Specifically, PG&E committed to install low-leakage RCP seals prior to startup from Unit 2 refueling outage 19 to reduce the potential seal leakage.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation has determined the potential for inadvertent SDS actuation has no discernable effect on the frequency of UFSAR-described RCP loss-of-flow as an accident initiator. The predicted likelihood of an SDS malfunction is negligible compared to the likelihood of UFSAR-evaluated RCP malfunctions. Inadvertent SDS deployment will not significantly impact the rotation of the RCPs or leakage from the existing seals.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

15-016 Disable Loop 3 T-hot Input to Reactor Vessel Level Instrumentation System (RVLIS) and Subcooled Margin Monitor (SCMM), Unit 1	
Reference Document No.:	SAPN Order 60079651
Reference Document Title:	Temporary Modification (TMOD) EVAL-Remove 1-TE-433
Activity Description:	
The proposed activity will disable (remove from scan) the Unit 1 T_{hot} resistance temperature detector (RTD) Loop 3 input to the RVLIS Train A. This is a software change performed within the existing capabilities of the RVLIS program. RVLIS calculates, outputs, and displays the reactor coolant system subcooled margin. Hence, the SCMM Unit 1 Train A Loop 3 T_{hot} RTD input is disabled as well. The proposed activity is associated with a TMOD.	
Summary of Evaluation:	
The 10 CFR 50.59 Evaluation determined that the proposed activity does not impact train redundancy, diversity, separation, or independence of RVLIS/SCMM. Train B is not impacted by this TMOD for either RVLIS or SCMM. The proposed activity will not initiate a new malfunction, nor will it affect, alter, or increase the likelihood of occurrence of any malfunction previously evaluated in the UFSAR. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-017 Updated Final Safety Analysis Report (UFSAR) Pressurizer Level Increase at 20 Percent Rated Thermal Power (RTP), Unit 1	
Reference Document No.:	UFSAR Sections 5.5.9.3.2, 15.1.3, and 15.1.1 0
Reference Document Title:	"Pressurizer Performance," "Optimization of Control Systems," and "References," respectively
Activity Description:	
This proposed activity implements two UFSAR changes related to the control of pressurizer level, which allows operation with an increased pressurizer level in Mode 1 with less than or equal to 20 percent RTP and Mode 2, but only when the unit is shutting down to Mode 3. This activity is not currently described in the UFSAR. There is no change to the UFSAR-described pressurizer level control during Mode 1 operation with power greater than 20 percent RTP, or in the shutdown Modes 3, 4, and 5.	
Summary of Evaluation:	
Technical evaluations of the proposed changes conclude that all applicable structural and thermal hydraulic safety analyses currently described in the UFSAR conservatively bound the plant conditions at a reduced power of ≤ 20 percent with the increased pressurizer level. Therefore, the results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-018* Updated Final Safety Analysis Report (UFSAR) 8.2, Units 1 and 2	
Reference Document No.:	UFSAR Section 8.2
Reference Document Title:	Offsite Power System
Activity Description:	
The manual action time change in Section 8.2 of the UFSAR replaces the description that the 500 kV system "could be placed in service in approximately 30 seconds, plus operator time," with a more technically accurate description of "could be placed in service within 30 minutes."	
Summary of Evaluation:	
The manual action is explicitly reflected in plant emergency operating procedures and training, the action has been demonstrated to be able to be completed in significantly less time than the proposed UFSAR manual action time, there is more than adequate time and indications to facilitate operator recovery from credible errors, and there is no effect on other plant systems. The 10 CFR 50.59 Evaluation concluded that this change does not require prior NRC approval.	

15-019 Emergency Diesel Generator (EDG) Steady State Loading, Units 1 and 2

Reference Document No.:	DDP 1*25089, Revision 0
Reference Document Title:	EDG Steady State Loading

Activity Description:

This proposed change consists of:

1) Revising documentation as listed in design change package number DDP 1*25089 to support the Technical Specification (TS) changes requested by License Amendment Request (LAR) 14-01 (DCL 14-018), "Revision to Technical Specification 3.8.1, AC Sources – Operating," dated March 27, 2014.

2) Changing the control room ventilation system (CRVS) Bus F (opposite unit) normal power supply to Bus G (same unit). The CRVS Bus H (same unit) normal power supply is also changed to Bus H (opposite unit). Bus F (for either unit) will not be used to power the CRVS unless EDG capacity is available.

3) Delaying manual restart of a spent fuel pool (SFP) pump following an accident with a loss of offsite power for up to three hours unless EDG capacity is available.

The License Amendment requested per LAR 14-01 is required to be approved by the NRC prior to implementation of this change.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation has determined that requiring that EDG capacity be verified before aligning the CRVS to Bus F power has no discernable effect on the likelihood of CRVS malfunction. The SFP temperature increase which occurs during the delay in starting for the SFP pump has no discernable effect on the likelihood of failure for affected systems, structures, or components. The Class IE bus loading remains within UFSAR Section 8.3.1.1.6.3.13-described diesel generator capacity. The modified manual actions do not introduce any time-critical or unrecoverable steps.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

15-024* Revised Uncontrolled Rod Cluster Control Assembly (RCCA) Analysis, Units 1 and 2

Reference Document No.:	CN-TA-08-099, Revision 0
Reference Document Title:	Revised Uncontrolled RCCA Analysis

Activity Description:

UFSAR Section 15.2.2, "Uncontrolled Rod Cluster Control Assembly Bank Withdrawal At Power," was revised to address the Westinghouse Nuclear Safety Advisory Letter (NSAL) 09-1.

Summary of Evaluation:

The revised UFSAR Section 15.2.2 overpressure analysis uses the same methodology approved by License Amendment (LA) 167/168; however, the initial power level was changed to use a more conservative limiting power level (80 percent) to address the Westinghouse NSAL 09-1.

The revised overpressure analysis demonstrates that the pressurizer safety valves and the reactor protection system power range high positive nuclear power rate reactor trip continue to prevent the reactor coolant system (RCS) pressure from exceeding RCS design pressure analysis limits and therefore there is no impact on the radioactive consequence of the accident; malfunction of structures, systems, and components; or an impact on the fuel and containment fission product barrier limits.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

15-025* Reanalyze Locked Rotor Event, Units 1 and 2	
Reference Document No.:	CN-TA-0 4- 116, Revision 0
Reference Document Title:	Steam Generator Tube Plugging (SGTP)/ Reactor Coolant System (RCS) Flow Asymmetry
Activity Description:	
The proposed activity is a reanalysis of the "Single Reactor Coolant Pump Locked Rotor" accident for the steam generators to assess the impact of increased limits for SGTP and loop-to-loop plugging asymmetry on maximum RCS pressure and maximum clad average temperature.	
Summary of Evaluation:	
The reanalysis determined that the calculated maximum RCS pressure and the maximum clad average temperature remain less than their respective safety analysis limits. The limits for fuel clad and RCS boundary are not exceeded in a locked rotor accident with increased tube plugging and the containment boundary is not affected by this accident. The codes used for the reanalysis are NRC-approved and the version updates follow a validation process that provides assurance that the updated code versions produce essentially the same results as the analysis of record based on identical inputs. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-026* Revised Boron Dilution Analyses, Units 1 and 2	
Reference Document No.:	CN-TA-14-27, Revision 0
Reference Document Title:	Revised Boron Dilution Analyses

Activity Description:

Diablo Canyon Power Plant is proposing to revise Updated Final Safety Analysis Report (UFSAR), Section 15.2.4, "Uncontrolled Boron Dilution." This change replaces the refueling and startup mode sections of CN-TA-87-122 with WEC CN-TA-14-27, Revision 0, "Diablo Canyon Units 1 and 2 Boron Dilution Analysis in Lower Modes," which reanalyzes the boron dilution event for startup and refueling (Modes 2 and 6, respectively) and adds hot standby, hot shutdown, and cold shutdown (Modes 3, 4, and 5, respectively).

The proposed activity calculates operator action times for Modes 2 and 6, and the minimum initial boron concentration ratio that must be maintained in Modes 3, 4, and 5 in order to ensure that the acceptance criteria of 15 minutes are met.

Summary of Evaluation:

The 10 CFR 50.59 Evaluation has determined that incorporating a new analysis of the uncontrolled boron dilution event in the startup, hot standby, hot shutdown, cold shutdown, and refueling modes (Modes 2, 3, 4, 5, and 6, respectively) does not involve any modifications to any plant systems, or introduction of any new failure modes or system interactions.

The Modes 3, 4, and 5 analyses included a determination of minimum boron concentrations which ensure that operator action time is unchanged and meets the acceptance criterion. Although operator action response times for Modes 2 and 6 are reduced, NEI 96-07, Revision 1, Section 4.3.2 criteria for acceptability of a modified operator action are all satisfied. Therefore, the proposed activity does not result in any increase in the likelihood of occurrence of a malfunction of structures, systems, or components important to safety previously evaluated in the UFSAR.

The revised analysis demonstrated that shutdown margin will be protected, no return to criticality will occur, safety limits will be maintained, and fission product barrier limits will not be exceeded or altered.

The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.

15-027* Reactor Internals Square-Root-of-the-Sum-of-the-Squares (SRSS) Method, Units 1 and 2	
Reference Document No.:	Updated Final Safety Analysis Report (UFSAR) Section 3.9
Reference Document Title:	Mechanical Systems and Components
Activity Description:	
The activity involves changing the method for combining the loss-of-coolant accident (LOCA) and seismic dynamic loading used for the structural analysis of reactor vessel (RV) internal components from absolute summation to square-root-of-the-sum-of-the-squares (SRSS) for the combination of seismic faulted condition dynamic loads with LOCA dynamic loads.	
Summary of Evaluation:	
This 10 CFR 50.59 Evaluation is limited to the use of SRSS for structural analysis of RV internals for combining the LOCA and seismic dynamic loading. The 10 CFR 50.59 Evaluation concludes the activity does not result in a departure from a method of evaluation described in the UFSAR because the alternative method has been approved by the NRC for the intended application. The criteria relating to the use/application of an alternative method of evaluation, as outlined by NEI 96-07, Revision 1, Section 4.3.8.2, have been satisfied. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	

15-028* **Incorporate Square-Root-of-the-Sum-of-the-Squares (SRSS)/Absolute Summation (ABSUM) for Seismic and Loss-of-Coolant-Accident (LOCA) Loads, Units 1 and 2**	
Reference Document No.:	N/A – See LBIE 2013-029
Reference Document Title:	Incorporate SRSS/ABSUM for Seismic and LOCA Loads
Activity Description:	
Diablo Canyon Power Plant (DCPP) is proposing a change to the current licensing basis to describe the SRSS method and the ABSUM method as alternative acceptable methods for the combination of seismic faulted condition loads with LOCA loads in the evaluation of American Society of Mechanical Engineers Section III, Class 1, "Reactor Coolant System Components and Supports." This activity is a reperformance of Licensing Basis Impact Evaluation 2013-029, per the DCPP corrective action program.	
Summary of Evaluation:	
The 10 CFR 50.59 Evaluation concludes the activity does not result in a departure from a method of evaluation described in the Updated Final Safety Analysis Report because the alternative method has been approved by the NRC for the intended application. The criteria relating to the use/application of an alternative method of evaluation, as outlined by NEI 96-07, Revision 1, Section 4.3.8.2, have been satisfied. The results of the 10 CFR 50.59 Evaluation show that the changes do not require prior NRC approval.	