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PG&E Letter DCL-06-050

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

Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
10 CFR 50.59 Report of Changes, Tests, and Experiments for the Period January 1, 2004, through December 31, 2005

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 Report for Diablo Canyon Power Plant (DCPP), Units 1 and 2, for the period January 1, 2004, through December 31, 2005. The report provides a summary of all 10 CFR 50.59 evaluations 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 in addition to 10 CFR 50.59, some LBIEs do not include a 10 CFR 50.59 evaluation and, therefore, are not included in this report.

The Plant Staff Review Committee has reviewed the referenced LBIEs and has concurred that the changes do not require prior NRC approval or require changes to the DCPP Technical Specifications.

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

Sincerely,

Donna Jacobs

IE47

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Enclosure
cc: Terry W. Jackson
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cc/enc: Alan B. Wang

Enclosure
PG&E Letter DCL-06-050

10 CFR 50.59 REPORT OF CHANGES, TESTS, AND EXPERIMENTS for the Period January 1, 2004, through December 31, 2005

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

Index of LBIEs

04-004 Unit 2 Turbine Control Replacement

Reference Document No.: DCP J-050625, Rev. 0
Reference Document Title: DEH/P2000 Turbine Control Replacement

Activity Description:

This design change replaces the Unit 2 Westinghouse digital electro-hydraulic P2000 computer turbine control system with a new system manufactured by Triconex. The Westinghouse control system is digital, but has an analog system as a backup. The analog system is also used for overspeed protection. The replacement system is completely digital. Since the critical function performed by the turbine control system is to protect the turbine from overspeed, this design change is treated as a digital upgrade, and has been reviewed in accordance with NRC Regulatory Issue Summary 2002-22, "Use of EPRI/NEI Joint Task Force Report, 'Guideline on Licensing Digital Upgrades: EPRI TR-102348, Revision 1, NEI 01-01: A Revision of EPRI TR-102348 to Reflect Changes to the 10 CFR 50.59 Rule.'"

The Triconex system uses triple modular redundant (TMR) hardware to improve system operation, reduce maintenance, and improve transient response. It will mount in the same cabinets as the existing P2000 computer next to the plant process computer in the computer room in the control room. The existing operator interface consists of an alarm and status panel, a valve indication and test panel, and a control panel. These interfaces are replaced by two touch screens. One touch screen is used for control and the other is used for display purposes. The Triconex system adds programmed ramps (for load shedding) for circulating water pump trip, main feedwater pump trip, and heater drip pump trip. The programmed ramps can be aborted by operator action.

Summary of Evaluation:

The 10 CFR 50.59 criteria affected by the proposed change are whether the proposed change increases the frequency of any accident (criterion [c][2][i]) and whether the proposed change increases the likelihood of any malfunction (criterion [c][2][ii]) previously evaluated in the Final Safety Analysis Report Update (FSARU). Using the guidance provided by the EPRI/NEI Joint Task Force report, the proposed change does not increase either the frequency of any accident or the likelihood of any malfunction previously evaluated in the FSARU.

There are no credited operator actions required by safety analyses for turbine control system operation. The time required to perform valve testing, perform load changes, or parallel the unit to the grid will be comparable to the existing system.

The addition of programmed ramps for events requiring load shedding will not challenge safety systems, but will improve plant response for these events.

Therefore, the proposed change did not require prior NRC approval.

04-008R1 Evaluation of Alternate Source Range Detector Power Supply Requirements, Revision 1

Reference Document No.:	See below
Reference Document Title:	OP B-8DS1, Revision 34, "Core Unloading," OP B-8DS2, Revision 33, "Core Loading," STP I-1A, Revision 94A, "Routine Shift Checks Required By Licenses," and AD8.DC55, Revision 20, "Outage Safety Scheduling"

Activity Description:

The proposed activity deletes the requirement for an alternate source range detector (i.e., N-51 or N-52) to be powered by a different operable vital 120 VAC power supply than the other operable source range detector (i.e., N-31 or N-32). The need for this procedure revision was identified during core offload operations during Unit 2 Twelfth Refueling Outage. The two operable source range detectors at the time (N-31 and N-51) were powered by the same power supply. The procedural requirement to have separate power supplies for two source range detectors during core alterations came from a Pacific Gas and Electric Company (PG&E) commitment made in PG&E Letter DCL-97-035, "Revision of the Bases for Technical Specification 3/4.9.2, Refueling Operations - Instrumentation," dated March 18, 1997. This commitment was not carried over to the Improved Technical Specifications (TSs) that were implemented in 1999.

The proposed activity screened in, requiring a 10 CFR 50.59 evaluation, because removal of the requirement to have redundant power supplies was considered adverse.

Summary of Evaluation:

In Mode 6, the purpose of the source range monitors is twofold; to monitor reactivity changes associated with fuel movements, and to provide early detection of boron dilution events.

The monitoring of reactivity changes associated with fuel movement requires that adding or removing fuel assemblies from the core be monitored. During offload and reload there are stages when a substantial number of fuel movements are monitored by a single detector. In the case of offload, the final stages consist of a number of fuel assemblies being monitored by a single detector. The physical location of the second detector precludes its use as a redundant monitor.

The detection of the boron dilution accident as credited in FSARU Section 15.2.4.2.2, "Dilution During Refueling," assumes that the operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation.

Boron dilution is a slow event in which credit is taken for operator action. No automatic mitigative actions are credited. The expected duration of this event is approximately 30 minutes. This provides sufficient time for the operator to recognize a high count rate and isolate the reactor makeup water source by closing valves and stopping the primary water supply pumps. A single operable source range channel will provide audible count rate information to ensure that significant changes in count rate are detected in time to preclude a boron dilution event.

TS 3.9.3, "Refueling Operations - Nuclear Instrumentation," requires that two source range neutron flux monitors be operable (two visual indications and one audible count rate circuit) to ensure that monitoring capability is available to detect changes in core reactivity. To be operable, each monitor must provide visual indication and at least one of the two monitors must provide audible alarm and rate indication in the control room.

TS Bases 3.8.2, "AC Sources - Shutdown," states that when the unit is shutdown, the TSs still ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and the concurrent loss of all offsite or onsite power is not required.

Thus, the power sources for the credited monitoring capability required to support offload and reload activities are not required to be independent. The use of a common power supply for the two source range channels will not result in more than a minimal increase in the frequency of an accident or likelihood of a malfunction, and has no impact on the consequences of an accident previously evaluated in the FSARU.

Therefore the proposed change did not require prior NRC approval.

05-002 Replace Unit 1 Positive Displacement Pump with a Centrifugal Charging Pump

Reference Document No.:	DCP M-49704, Rev. 0
Reference Document Title:	Replacement of the Positive Displacement Pump

Activity Description:

To reduce unscheduled maintenance, and improve equipment availability, the Unit 1 positive displacement pump (PDP) is being replaced with a centrifugal charging pump (designated CCP 1-3) similar to those already in use for the two high-head emergency core cooling system (ECCS) centrifugal charging pumps (designated CCP 1-1 and CCP 1-2). The following aspects of this activity are considered adverse, requiring a 10 CFR 50.59 evaluation:

1. The new CCP 1-3 will provide higher flow than the PDP. To limit pump flow at low reactor coolant system pressures, under low temperature overpressure protection (LTOP) conditions, the CCP 1-3 discharge will be manually aligned through a new pressure reducing orifice.

2. The new CCP 1-3 will provide higher flow than the PDP during certain accidents or events. Under these conditions, increased total charging flow (above that currently assumed in analyses) could have an adverse impact.

3. The new CCP 1-3 has a larger horsepower motor than the PDP and will place more load on its vital bus.

Summary of Evaluation:

1. The need to manually align the discharge of the new CCP 1-3 for LTOP conditions was evaluated and determined to be acceptable. The LTOP alignment is needed during plant cooldown evolutions where there is ample time for the operator to perform the lineup. There is, at most, only a minimal increase in the likelihood that the LTOP function will fail to be enabled properly.

2. The performance of the new CCP 1-3, with its higher flow, was evaluated for its effect on the following accident analyses:

    - Large-break loss-of-coolant accident (LOCA)
    - Inadvertent actuation of the safety injection system
    - Boron dilution accidents

- Steam generator tube rupture
- Pressurized thermal shock events caused by mass injection.

In each case, the results of the evaluation showed that the mitigation of the accident or event is maintained within the current dose limits for each event. There is no increase in dose consequences.

3. The impact of the larger CCP motor on 4kV vital bus loading has been evaluated. The impact on fault protection, bus capacity, bus transfer, bus transient voltage during loading, and emergency diesel generator loading have been determined to be within design limits and, therefore, acceptable.

Therefore, the proposed replacement of the PDP with a CCP did not require prior NRC approval.

05-003 Replacement Steam Generator Seismic Damping Values

Reference Document No.:	FSAR Update, Rev. 15
Reference Document Title:	FSAR Update Section 3.7.1.3, "Critical Damping Values"

Activity Description:

PG&E is procuring replacement steam generators (RSG). The RSGs will require seismic qualification to meet Diablo Canyon Power Plant (DCPP) licensing requirements. An important parameter used in the seismic qualification is the percentage of critical damping (damping values) used in the seismic analysis.

The damping values used for the DCPP original steam generator (OSG) qualification were one percent for the design earthquake (DE) and double design earthquake (DDE). These values have been changed by licensing activities over many years. The current OSG damping values are two percent for DE, four percent for DDE, four percent for Hosgri Earthquake, and five percent for the Long Term Seismic Program.

The proposed activity is to use the current OSG damping values for the RSGs. This activity conservatively screened in, requiring a 10 CFR 50.59 evaluation, to demonstrate the applicability of the OSG licensing basis for damping values to the RSG seismic analysis.

Summary of Evaluation:

The damping values used in the current OSG design and licensing bases and safety analyses have been previously approved by the NRC.

The current licensing basis was reviewed to verify the NRC acceptance of each licensing activity, and whether it imposes any limitations for use on RSGs. The evaluation concluded that applying the OSG damping values to the RSGs does not constitute a departure from a method of evaluation described in the FSARU.

Therefore the licensing basis associated with the OSG damping values can be applied to the RSG seismic analysis without prior NRC approval.

05-006 Replace Unit 2 Positive Displacement Pump with a Centrifugal Charging Pump

Reference Document No.:	DCP M-50704, Rev. 0
Reference Document Title:	Replacement of the Positive Displacement Pump

Activity Description:

To reduce unscheduled maintenance, and improve equipment availability, the Unit 2 PDP is being replaced with a centrifugal charging pump (designated CCP 2-3) similar to those already in use for the two high-head ECCS centrifugal charging pumps (designated CCP 2-1 and CCP 2-2). The following aspects of this activity are considered adverse, requiring a 10 CFR 50.59 evaluation:

1. The new CCP 2-3 will provide higher flow than the PDP. To limit pump flow at low reactor coolant system pressures, under LTOP conditions, the CCP 2-3 discharge will be manually aligned through a new pressure reducing orifice.

2. The new CCP 2-3 will provide higher flow than the PDP during certain accidents or events. Under these conditions, increased total charging flow (above that currently assumed in analyses) could have an adverse impact.

3. The new CCP 2-3 has a larger horsepower motor than the PDP and will place more load on its vital bus.

Summary of Evaluation:

1. The need to manually align the discharge of the new CCP 2-3 for LTOP conditions was evaluated and determined to be acceptable. The LTOP alignment is needed during plant cooldown evolutions where there is ample time for the operator to perform the lineup. There is, at most, only a minimal increase in the likelihood that the LTOP function will fail to be enabled properly.

2. The performance of the new CCP 2-3, with its higher flow, was evaluated for its effect on the following accident analyses:

   - Large-break LOCA
   - Inadvertent actuation of the safety injection system
   - Boron dilution accidents
   - Steam generator tube rupture
   - Pressurized thermal shock events caused by mass injection.

   In each case, the results of the evaluation showed that the mitigation of the accident or event is maintained within the current dose limits for each event. There is no increase in dose consequences.

3. The impact of the larger CCP motor on 4kV vital bus loading has been evaluated. The impact on fault protection, bus capacity, bus transfer, bus transient voltage during loading, and emergency diesel generator loading have been determined to be within design limits and therefore acceptable.

Therefore, the proposed replacement of the PDP with a CCP did not require prior NRC approval.

05-008 Unit 2 Digital Feedwater Control System Replacement

Reference Document No.: DCP-J-050731, Rev. 0
Reference Document Title: Replace Digital Feedwater Control System

Activity Description:

This activity replaces the Unit 2 Westinghouse digital feedwater control system (DFWCS) with Triconex TMR hardware to improve system operation, reduce maintenance and improve transient response. Existing videographic recorders and hardwired manual/auto (MA) stations on the control room vertical boards and control console will be replaced with touch-screen flat panel monitors to provide a new human machine interface (HMI).

The activity screens in, requiring a 10 CFR 50.59 evaluation, because it fundamentally changes data presentation and creates new potential failure modes for operator interaction with the system. In addition, the new HMI is dependent upon the processors for operation. The existing DFWCS control valve and feedwater pump slave MA stations are independent of the processors.

Summary of Evaluation:

The new HMI will provide the same information to the operators as the existing system, but in a different format. The format change in itself is not considered adverse and will be addressed through operator training. The new DFWCS implements TMR processors and input/output equipment that will continue to function in the presence of a single fault within the TMR architecture. All three processors must fail for the system to fail. The existing system will fail if two processors fail. The new DFWCS system will not exhibit performance characteristics that more than minimally increase the need for operator intervention or increase operator burden to support operation of the system in normal or off-normal conditions.

Therefore the proposed DFWCS replacement did not require prior NRC approval.

05-009 Unit 2 Upflow Conversion and Upper Head Temperature Reduction

Reference Document No.:	DCP N-50449, Rev. 0
Reference Document Title:	Reactor Internals Upflow Conversion and Upper Head Temperature Reduction

Activity Description:

This activity implements the reactor internals upflow conversion and upper head temperature reduction (UC/UHTR) design change. This involves modifying the Unit 2 reactor internals to provide an upflow design and to introduce cold leg fluid into the upper head of the reactor vessel. The upflow conversion will eliminate the pressure differential across the baffle plates to reduce the potential for baffle jetting-related fuel assembly degradation. The upper head temperature reduction is being done to help mitigate upper head primary water stress corrosion cracking related to Alloy 600 materials.

Evaluations have been performed on the impact of the modifications to the reactor internals on fuel assembly integrity, maintaining a coolable core geometry, stresses from a seismic event, LOCA loading, and safe shutdown capability. These evaluations utilized approved analytical methodologies.

The input parameters were altered for the modified internals configuration, resulting in conservatively screening this activity in for a 10 CFR 50.59 evaluation.

Summary of Evaluation:

The UC/UHTR design change meets all design criteria for maintaining the design basis functions of the internals. The changes do not increase the frequency of evaluated events, or introduce any new plant operating conditions or system configurations, that are precursors of evaluated accidents, or accidents of a different type.

Evaluations of the proposed activity on the dose consequences for accidents described in the FSARU concluded that that there was no increase in any radiological source term for any event (e.g. LOCA, rod ejection, locked rotor). Also, there was no impact on any secondary steam releases that are used to determine offsite doses. There was no increase in any of the analyzed dose consequences.

Therefore, the UC/UHTR modification did not require prior NRC approval.

05-010 Replace Unit 2 Reactor Makeup Control System

Reference Document No.: DCP J-50779, Rev. 0
Reference Document Title: Replace Reactor Makeup Control System

Activity Description:

The reactor makeup control system (RMCS), and the boric acid and primary water batch controllers will be replaced with a digital process control system that has anticipatory control capability. The change will improve accuracy and reliability of the system, and assist in reducing batch errors.

The activity screens in, requiring a 10 CFR 50.59 evaluation, because it is a digital upgrade that changes some manual actions to automatic actions.

Summary of Evaluation:

The FSARU analyzed accident that is affected by the RMCS replacement is an uncontrolled boron dilution of the reactor coolant system. The new system does not change the failure mode of any valve on loss of power or instrument air. The operator maintains the capability to manually operate pump and valve controls. Therefore, there is no increase in the frequency of the dilution accident.

The new system maintains the availability of an alternate boration flow path to the charging pumps, maintains post accident monitoring instrumentation, and does not affect the ability to control or monitor the volume control tank levels or letdown flows.

Although there is a fundamental change in the way information is presented to the operator, and the way the operator interacts with the system, these changes are not adverse and will be addressed in training provided to the operators.

Therefore, the RMCS replacement did not require prior NRC approval.