UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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August 5, 2009

John T. Conway
Senior Vice President-Energy Supply and
Chief Nuclear Officer
Pacific Gas and Electric Company
P.O. Box 3
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Avila Beach, California 93424

Subject: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2009003 AND 05000323/2009003

Dear Mr. Conway:

On June 26, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Diablo Canyon Power Plant. The enclosed integrated inspection report documents the inspection findings, which were discussed on June 29, 2009, with Mr. James Becker, Site Vice President and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two NRC-identified findings of very low safety significance (Green) and three Severity Level IV violations. All of these findings were determined to involve violations of NRC requirements. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest the violations or the significance of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 E. Lamar Blvd, Suite 400, Arlington, Texas 76011-4125; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Diablo Canyon Power Plant. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the Diablo Canyon Power Plant. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document

Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Vince G. Gaddy, Chief
Project Branch B
Division of Reactor Projects

Docket: 50-275
50-323
License: DPR-80
DPR-82

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NRC Inspection Report 05000275/2009003 and 05000323/2009003
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RIV:RI:DRP/B	SRI:DRP/B	C:DRS/PSB1	C:DRS/PSB2	C:DRS/EB1
MABrown	MSPeck	MShannon	GWerner	TFarnholtz
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8/5/09	8/4/09	7/28/09	7/31/09	7/30/09
C:DRS/EB2	C:DRS/OB	C:/DRP/B		
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000275, 05000323
License:	DPR-80, DPR-82
Report:	05000275/2009003 05000323/2009003
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	March 28 through June 26, 2009
Inspectors:	M. S. Peck, Senior Resident Inspector M. A. Brown, Resident Inspector G. L. Guerra, CHP, Emergency Preparedness Inspector
Approved By:	V. G. Gaddy, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000275/2009003, 05000323/2009003; 3/28/2009 – 6/26/2009; Diablo Canyon Power Plant, Integrated Resident and Regional Report; Identification and Resolution of Problems, Event Follow-up and Other Activities.

The report covered a 3-month period of inspection by resident inspectors and an announced baseline inspection by a region based inspector. Two Green noncited violations of significance and three Severity Level IV noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG 1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Severity Level IV. The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a condition prohibited by technical specifications. The licensee failed to correctly evaluate the March 18, 2009, failure of the Unit 2 control rod demand position indicators for reportability. The inspectors concluded that the failure of control rod position indicators was a condition prohibited by Technical Specification 3.17, "Rod Position Indication."

  This finding is greater than minor because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the regulations in order to perform its regulatory function. This finding affected the mitigating systems cornerstone. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV, noncited violation. The licensee entered this issue into the corrective action program as Notification 50242153. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the Unit 2 control rod demand position indicators for reportability [P.1(c)](Section 40A2).

- Severity Level IV. The inspectors identified a noncited violation of 10 CFR Part 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with current plant design criteria. The Final Safety Analysis Report Update stated that Diablo Canyon was designed to comply with the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, published in July 1967. The inspectors identified that the

Diablo Canyon Safety Evaluation Report stated that the NRC used General Design Criteria published in July 1971 to review the plant design. In addition, during the initial licensing process, the licensee stated that the plant was evaluated against the 1971 design criteria during the licensing process.

The inspectors evaluated this finding using the traditional enforcement process because the failure to update the Final Safety Analysis Report affected the NRC's ability to perform its regulatory function. The inspectors concluded that the failure to update the Final Safety Analysis Report was a Severity Level IV violation based on the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I – Reactor Operations, dated January 14, 2005, because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner [P.1(d)](Section 40A2).

- <u>Green</u>. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, "Corrective Action," after Pacific Gas and Electric failed to adequately correct a nonconforming condition related to the adequacy of design documentation to demonstrate the acceptability of design control for the 500 kV delayed offsite power system. The licensee stated the design control documentation demonstrated that the offsite power system met the design basis was not retrievable. The licensee entered this nonconforming condition into the corrective action system. On October 28, 2008, plant engineers completed an evaluation of the nonconforming condition and concluded the delayed offsite power system design basis was demonstrated by a "road map" of pre-existing analyses created to support other plant functions. The inspectors concluded that the "road map" was less than adequate because the licensee failed to consider the affect of the loss of reactor coolant pump seal cooling and injection anticipated during the time needed to align the offsite power supply to the engineering safety feature buses. The inspectors concluded that the failure of the licensee to promptly correct the nonconforming condition and ensure that the "road map" implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance.

  This finding is more than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors concluded this finding is of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition to ensure that the offsite power system design basis was met [P.1(c)](Section 40A5).

- Severity Level IV. The inspectors identified a noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a thermal hydraulic analysis to determine if prior NRC approval was required for a 30-minute delay time to align offsite power. This analysis, Calculation STA-274, "RETRAN Evaluation of GDC-17 Loss of AC Scenario," Revision 0, demonstrated that the 30-minute delayed offsite power source was acceptable. On December 31, 2008, a Pacific Gas and Electric 10 CFR 50.59 screen concluded that Calculation STA-274 was not required to be evaluated to determine if prior NRC approval was required for the delay time. On March 31, 2009, the inspectors concluded that the licensee was required to evaluate Calculation STA-274 to determine if prior NRC approval was needed. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required for the 30-minute delay time to align offsite power.

  The inspectors concluded that the finding is more than minor because the changes made to the facility required prior NRC review and approval. The finding affected the Mitigating Systems Cornerstone because the change described how the delayed offsite power source met the design basis. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency that did not result in the loss of operability or functionality. Because the issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated this finding using the traditional enforcement process. This issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1(c)](Section 40A5).

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a noncited violation of 10 CFR 50.54(q), after a Pacific Gas and Electric shift manager failed to promptly declare an Unusual Event in accordance with the emergency plan. Station procedures required that the emergency plan be activated within 15 minutes following a fire alarm in the containment building that could not be validated as false.

  This finding is more than minor because it affected the response organization performance attribute of the Emergency Preparedness Cornerstone due to failure to properly recognize plant conditions commensurate with an Unusual Event classification. This finding was of very low safety significance, because it did not meet any higher level emergency plan and implementing procedure notification requirements. The licensee entered this issue into the corrective action program as Notification 50247279. This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee failed to implement the time requirements of the Emergency Plan [H.4(b)](Section 4OA3).

B. Licensee-Identified Violations

A violation of very low safety significance, which was identified by the licensee, has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and corrective action tracking numbers are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, Diablo Canyon Units 1 and 2 were operating at full power. Unit 1 remained at full power throughout the inspection period. The licensee reduced Unit 2 to 50 percent power on May 15, 2009, for condenser cooling water maintenance. The licensee returned Unit 2 to full power on May 22. On June 30, plant operators rapidly shut down Unit 2 following the failure of a main transformer bank cooling system. Unit 2 remained off line for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

1R01 Adverse Weather Protection (71111.01)

Summer Readiness for Offsite and Alternate-ac Power

a. Inspection Scope

The inspectors performed a review of the licensee's preparations for summer weather for selected systems, including conditions that could lead to loss-of-offsite power and conditions that could result from high temperatures. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator and the plant to verify that the appropriate information was being exchanged when issues arose that could affect the offsite power system. Examples of aspects considered in the inspectors' review included:

- Coordination between the transmission system operator and the plant during off-normal or emergency events
- Explanations for the events
- Estimates of when the offsite power system would be returned to a normal state
- Notifications from the transmission system operator to the plant when the offsite power system was returned to normal

During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Final Safety Analysis Report Update and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the 230 kV preferred offsite power system

These activities constitute completion of one readiness for summer weather effect on offsite and alternate power sample as defined in Inspection Procedure 71111.01-05.

b. Findings

Introduction. The inspectors identified an unresolved item related to the acceptability of the 230 kV preferred offsite power system to meet design basis requirements. Additional NRC review is needed to determine if the preferred offsite system has sufficient capacity and capability to supply the engineered safety features buses for all required accidents and transients.

Description. On April 10, 2009, the inspectors identified that the plant electrical design analysis may not be adequate to demonstrate that the 230 kV preferred offsite power system had sufficient capacity and capability to meet station loads following an accident on one unit and concurrent safe shutdown on the remaining unit or for a concurrent safe shutdowns on both units. The Diablo Canyon offsite power sources include a normal supply from the 230 kV distribution system and a delayed supply from the 500 kV distribution system. The normal supply is required to immediately power the engineered safety feature systems following a station accident or a reactor trip. The delayed supply backs up the normal supply and can be aligned to power the engineered safety feature systems in about 30 minutes.

NRC Safety Evaluation Report, “Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Pacific Gas And Electric Company Diablo Canyon Nuclear Power Station, Units 1 and 2 San Luis Obispo County, California Docket Nos. 50-275 And 50-323,” and Final Safety Analysis Report Update, Section 8.1, “Offsite Power Systems,” established IEEE Standard 308-1971, “Class IE Electrical Systems,” as part of the preferred offsite power system design basis. IEEE Standard 308-1971, Section 8.1.1, “Multi-Unit Station Considerations,” stated:

> “Capacity. A multi-unit station may share preferred power supply capacity between units. In such a case, as a minimum the total preferred capability must be sufficient to operate the engineering safety features for a design basis accident on one unit and those systems required for concurrent safe shutdown on the remaining units. The type of accident and shutdown and the unit assumed to have the accident, shall be those which give the largest total preferred capability requirements.”

Pacific Gas and Electric used Design Calculation 357AA-DC, “Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis,” September 24, 2007, to ensure that the preferred offsite power system was capable of meeting design basis electrical load requirements. The inspectors identified that Calculation 357AA-DC did not include load flow cases representing the largest total capability for an accident on one unit and concurrent safe shutdown of the other unit or concurrent safe shutdown of both units. Calculation 357AA-DC modeled the limiting load flow cases as an accident (or unit trip) on one unit while assuming a previous shutdown had occurred on the other unit. The load flow modeling was based on the assumption that plant operators would perform an “orderly shutdown” entailing the manual transfer of electrical loads to the 230 kV system at a time of low electrical demand from the accident or tripped unit.

On June 26, 2009, the licensee completed a preliminary re-evaluation of preferred offsite power supply load flow assuming an accident on one unit and a concurrent safe shutdown on the remaining unit, and for an assumed concurrent safe shutdown on both units. The licensee concluded the voltage at the 4160 Class 1E vital buses would be less than adequate to support operation of the engineering safety features under design conditions. The licensee also analyzed the plant response based on actual available 230 kV switchyard voltages between November 2008, and June 26, 2009. For these cases, the licensee concluded that 4160 Class 1E vital bus voltages would have intermittently dropped below the minimum voltage required for operability of the engineering safety features. The inspectors concluded that actual 230 kV system voltage recovered prior to exceeding the 72-hour action time for Technical Specification 3.8.1, “AC Sources - Operating,” for any single occurrence.

Pacific Gas and Electric had previously identified that the 230 kV offsite power source had insufficient voltage (reported as Licensee Event Report 1-95-007, “230 kV System May Not Be Able to Meet its Design Requirements for all Conditions Due to Personal Error).” The corrective actions included increasing the capability of the startup transformers and installation of large capacitor banks at the plant switchyard and Mesa Substation. When sizing the replacement transformers, the licensee assumed that the preferred offsite power system only needed to have the capacity and capability for an accident or trip on one of the two units. In a licensing position paper for the 230 kV system loading requirements (Letter File 227961, from the Director Licensing to Director, Electrical I&C Engineering September 27, 1995), the licensee added the word “orderly” to the safety shutdown requirements specified in IEEE Standard 308-1971. The licensee did not perform a 10 CFR 50.59 evaluation of this change nor seek prior NRC approval. As a result, the licensee’s previous corrective actions were insufficient to restore the preferred offsite power system to compliance with the design basis.

This issue is unresolved pending NRC review of the 230 kV preferred offsite power system design basis requirements. Unresolved Item: 05000275;323/2009003-01, “Corrective Action Following Degraded Offsite Power System.”

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk significant systems:

- Unit 1 and Unit 2, seismic monitoring and trip systems, May 7, 2009
- Unit 2, containment spray system, June 2, 2009
- Unit 1, auxiliary feedwater system, June 16, 2009

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system; and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, Final Safety Analysis Report Update, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three partial system walkdown samples as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

On April 21 and 22, 2009, the inspectors performed a complete system alignment inspection of the Unit 2 component cooling water system to verify the functional capability of the system. The inspectors selected this system because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. The inspectors reviewed a sample of past and outstanding work orders to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment alignment problems were being identified and appropriately resolved. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one complete system walkdown sample as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Fire Inspection Tours

a. Inspection Scope

The inspectors conducted fire protection walkdowns that were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Units 1 and 2, containment penetration rooms, Fire Area 3-C-C
- Unit 1, residual heat removal pump room, Fire Area 3-B-1
- Units 1 and 2, circulating water pump room, Fire Zone 30-A-5
- Units 1 and 2, intake structure control room, Fire Zone 30-B

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. The inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed, that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

These activities constitute completion of four quarterly fire-protection inspection samples as defined by Inspection Procedure 71111.05-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On April 22 and April 28, 2009, the inspectors observed a crew of licensed operators in the licensee's simulator to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- Licensed operator performance
- Crew's clarity and formality of communications
- Crew's ability to take timely actions in the conservative direction
- Crew's prioritization, interpretation, and verification of annunciator alarms
- Crew's correct use and implementation of abnormal and emergency procedures
- Control board manipulations
- Oversight and direction from supervisors
- Crew's ability to identify and implement appropriate technical specification actions and emergency plan actions and notifications

The inspectors compared the crew's performance in these areas to pre-established operator action expectations and successful critical task completion requirements. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two quarterly licensed-operator requalification program samples as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk significant systems:

- Unit 2, containment fan cooling units, May 11, 2009
- Unit 2, rod control system, June 3, 2009

The inspectors reviewed events such as where ineffective equipment maintenance has resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- Implementing appropriate work practices
- Identifying and addressing common cause failures

- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)
- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed licensee personnel's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk significant and safety related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Notification 50231639, Units 1 and 2, non-conservative Technical Specification 3.8.1, April 9, 2009
- Risk Assessment 09-06, Revision 0, planned maintenance of auxiliary saltwater Pump 2-1, May 12, 2009
- Technical Specification Sheet 2-TS-09-0323, Unit 2, modification to remove the negative rate reactor trip from the reactor protection systems while at power, May 13, 2009
- Order 600011161, Unit 2, control rod testing, May 20, 2009

- Technical Specification Sheet 1-TS-09-0603, failure of the Unit 1 protection channel, both motor-driven auxiliary feedwater pumps while auxiliary saltwater Pump 1-2 out of service for maintenance on June 29, 2009

The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that licensee personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When licensee personnel performed emergent work, the inspectors verified that the licensee personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of five maintenance risk assessments and emergent work control inspection samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Notification 50227292, Unit 1, Valve FW-1-LCV-106 packing leak, March 29, 2009

- Notification 50231656, Units 1 and 2, diesel generator power factor testing, April 9, 2009

- Notification 50233583, Unit 2, auxiliary saltwater Pump 2-1 seal leak rate high, April 20, 2009

- Notification 50237951, Unit 1, auxiliary saltwater Pump 1-1 exhaust fan vibration increase, April 29, 2009

- Notification 50248314, Unit 2, emergency diesel generator missing fasteners, June 15, 2009

- Manual actions for motor-driven auxiliary feedwater level control valves, Unit 1 on June 29, 2009

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and Final Safety Analysis Report Update to the licensee's evaluations, to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six operability evaluations inspection samples as defined in Inspection Procedure 71111.15-05

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following postmaintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Measurement of hot gaps of replacement steam generator upper lateral supports, lower lateral supports, reactor coolant pump supports, hot leg rupture restraints and crossover leg rupture restraints as part of the Unit 1 steam generator replacement project, April 6, 2009

- Calibration of Unit 1 wide range steam generator level transmitters installed as part of Unit 1 steam generator replacement project, May 5, 2009

- Calibration of Unit 1 narrow range steam generator level transmitters installed as part of Unit 1 steam generator replacement project, May 7, 2009

- Postmaintenance Test 37.03, Unit 1, replacement steam generator determination of full power reference temperature, May 11, 2009

- Unit 1 reactor coolant system primary coolant flow measurements following steam generator replacement, May 13, 2009

- Postmaintenance Test 04.30, Unit 1, steam generator replacement testing, June 29, 2009

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following:

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed

- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the Final Safety Analysis Report Update, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of six postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. <u>Findings</u>

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. <u>Inspection Scope</u>

The inspectors reviewed the final safety analysis report update, procedure requirements, and technical specifications to ensure that the six surveillance activities listed below demonstrated that the systems, structures, and/or components tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the significant surveillance test attributes were adequate to address the following:

- Preconditioning

- Evaluation of testing impact on the plant

- Acceptance criteria

- Test equipment

- Procedures

- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems
- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints

The inspectors also verified that licensee personnel identified and implemented any needed corrective actions associated with the surveillance testing.

- April 2, 2009, Unit 1, core reactivity routine surveillance test
- April 3, 2009, Unit 2, containment spray Pump 2-2 inservice test
- April 3, 2009, Unit 1, residual heat removal Pump 2-2 inservice test
- April 4, 2009, Unit 1, containment spray Pump 1-2 inservice test
- April 21, 2009, Unit 1, cable spreading room carbon dioxide fire system routine surveillance test
- June 20, 2009, Unit 2, emergency diesel Generator 2-2 routine 24-hour load test

Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three routine surveillance and three inservice testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. <u>Findings</u>

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed an in-office review of Diablo Canyon Emergency Action Level Procedure, EP G-1, “Emergency Classification and Emergency Plan Activation,” Revisions 37A and 38, effective January 22 and February 26, 2009, respectively. These revisions corrected administrative errors introduced during Revision 37, effective July 2, 2008. The discovery and correction of these errors by the licensee was documented in Notification 50205989.

These revisions were compared to its previous revision, to the criteria of NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” Revision 1, to NEI Report 99-01, “Emergency Action Level Methodology,” Revision 4, and to the standards in 10 CFR 50.47(b) to determine if the revision adequately implemented the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two samples as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

Training Observations

a. Inspection Scope

The inspectors observed a simulator training evolution for licensed operators on June 9, 2009, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors’ activities was to note any weaknesses and deficiencies in the crew’s performance and ensure that the licensee evaluators noted the same issues and entered them into the corrective action program. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the attachment.

These activities constitute completion of one sample as defined in Inspection Procedure 71114.06-05.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the first quarter 2009 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator for Diablo Canyon Units 1 and 2 for the period from the first quarter 2008 through the first quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, and NUREG-1022, “Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance were used. The inspectors reviewed the licensee’s operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports, and NRC integrated inspection reports for the period of the first quarter 2008 through the first quarter 2009 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted. The inspectors identified one example where the licensee had not reported a safety system functional failure that occurred in August 2008. The licensee entered this issue into the corrective action program as Notification 50245926.

These activities constitute completion of two safety system functional failures samples as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index - Emergency ac Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Emergency Power System performance indicator for Diablo Canyon Units 1 and 2 for the period from the first quarter 2008 through the first quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports and NRC integrated inspection reports for the period of the first quarter 2008 through the first quarter 2009 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

These activities constitute completion of two mitigating systems performance index emergency ac power system samples as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.4 Mitigating Systems Performance Index - High Pressure Injection Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - High Pressure Injection Systems performance indicator for Diablo Canyon Units 1 and 2 for the period from the first quarter 2008 through the first quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports and NRC integrated inspection reports for the period of the first quarter 2008 through the first quarter 2009 to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

These activities constitute completion of two mitigating systems performance index high pressure injection system samples as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. The inspectors reviewed attributes that included: the complete and accurate identification of the problem; the timely correction, commensurate with the safety significance; the evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews; and the classification, prioritization, focus, and timeliness of corrective actions. Minor issues entered into the licensee's corrective action program because of the inspectors' observations are included in the attached list of documents reviewed.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure, they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's corrective action program. The inspectors accomplished this through review of the station's daily corrective action documents.

The inspectors performed these daily reviews as part of their daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of January 1, 2009, through June 30, 2009, although some examples expanded beyond those dates where the scope of the trend warranted.

The inspectors also included issues documented outside the normal corrective action program in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenge lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's corrective action program trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy. Specific documents reviewed are listed in the attachment.

These activities constitute a single semi-annual trend inspection sample.

b. Findings

Adverse Trend in Problem Evaluation

The inspectors previously identified an adverse trend in problem evaluation in December 2008 (described in Section 4OA2 of Inspection Report 05000275/2008005). The inspectors identified that this adverse trend continued during the first two quarters of 2009. The inspectors analyzed this trend and identified a common theme related to poor licensee management of the plant design/licensing bases and inconsistent implementation of regulatory administrative processes. The inspectors concluded that some issues identified in the trend could indicate the existence of a more significant concern affecting the NRC's ability to regulate the licensee:

NRC-identified issues related to poor licensing and design basis management:

- Violation of design control requirements associated with the failure to maintain adequate capacity and capability of the preferred offsite power system, as described in Section 1R01 of this report. This finding illustrated the failure of the licensee to maintain the plant design basis for a period of time. The finding also revealed that the licensee's evaluation process was less than adequate to reconstruct the historic licensing basis changes to identify the underlying cause of the problem.

- Failure to maintain the final safety analysis report update, as described in Section 4OA2 of this report. This violation illustrated the weakness in the licensee's programmatic processes to evaluate problems associated with maintaining the plant licensing basis. After the inspectors identified the issue, the licensee initially closed the corrective action document without fixing the problem (as discussed in Notification 50202606).

- Failure to perform a 50.59 evaluation for spent fuel pool special test, as described in Notification 50041356. This minor violation illustrated the licensee's failure to implement the industry 50.59 program as described in NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations."

- Failure to perform an adequate 50.59 evaluation for modifications to the special protection scheme for the 500 kV switchyard, as described in Notification 50041356. This minor violation illustrated the licensee's failure to implement the industry 50.59 program as described in NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations."

- Failure to perform a 50.59 evaluation, as described in Section 4OA5 of this report. This violation illustrated the licensee's failure to implement the industry 50.59 program as described in NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations."

- An inadequate 50.59 evaluation for the Unit 1 containment sump modification, as discussed in Section 4OA2.2 of Inspection Report 05000275/2007005. This violation illustrated a failure to understand when prior NRC approval is required for change to the facility as described in the Final Safety Analysis Report Update.

- Violation of design control requirements associated with the failure to maintain adequate capacity and capability of the preferred offsite power system while sharing a startup transformer, as described in Section 4OA2.5.2 of Inspection Report 05000275/2008005. This finding illustrated the failure of the licensee to understand and apply the plant design and licensing basis to offsite power system operability.

- Violation of the station 50.59 evaluation procedure, as described in Section 1R05 of Inspection Report 05000275/2008004. This finding illustrated the failure of the licensee to recognize a condition outside of the plant design basis associated with an explosive mixture of oxygen and hydrogen discovered in the Unit 2 reactor coolant drain tank, waste gas surge tank, and interconnecting piping.

- Violation of design control associated with the failure to maintain adequate capacity and capability of the emergency diesel generators, as described in Section 4OA5 of Inspection Report 05000275/2008005. This finding illustrated the failure of the licensee to understand and apply the plant design and licensing basis to onsite emergency power system.

- Violation of the reactor coolant leakage detection technical specification, as described in Section 1R15 of Inspection Report 05000275/2008004. This finding illustrated the failure of the licensee to recognize design basis requirements establishing operability of the reactor coolant leakage detection system.

- Violation of 10 CFR 50.71 following failure to update the Final Safety Analysis Report Update with current plant design basis criteria, as described in Section 40A2 of this report.

- Violation of 10 CFR 50.71 following failure to update plant changes in the Final Safety Analysis Report Update, as described in Section 2PS2 of Inspection Report 05000275/2008009.

- Violation of design control associated with the failure to maintaining adequate capacity and capability of the delayed offsite power system, as described in Section 4OA5 of this report. This finding illustrated the failure of the licensee to apply generic industry safety analysis to a plant specific application. In this case, plant engineers did not thoroughly evaluate the plant licensing basis to identify the applicability of a fire protection analysis addressing the loss of reactor coolant pump seal cooling and injection flow to the loss of ac power.

NRC-identified issues related to other regulatory administrative functions:

- Failure to meet NRC reporting requirements, as described in Section 4OA2 of this report. This violation illustrated the weakness in the licensee's programmatic processes to evaluate problems for reportability. After the inspectors identified the failure of Pacific Gas and Electric to initiate a licensee event report, the licensee initially deposited the issue as not reportable, as discussed in Notification 50242153.

- Failure to accurately report performance indicator data to the NRC, as discussed in Notification 50245926. This issue illustrated the weakness in the licensee's programmatic processes to collect and report the performance indicator data for the fourth quarter of 2008 to the NRC. After the inspectors identified that a safety system functional failure was missed on the performance indicator submittal, Pacific Gas and Electric initially dispositioned the issue as not required to be reported, as discussed in Notification 50245926.

- Violation of maintenance rule scoping criteria, as described in Section 1R12 of Inspection Report 05000275/2008003. This finding illustrated the failure of the licensee to understand and implement maintenance rule scoping criteria for equipment used in the emergency operating procedures.

- Violation of maintenance rule scoping criteria, as described in Section 1R12 of Inspection Report 05000275/2007003. This finding also illustrated the failure of the licensee to understand maintenance rule scoping criteria for equipment used in the Emergency Operating Procedures.

- Violation of maintenance rule performance monitoring criteria, as described in Section 1R12 of Inspection Report 05000275/2008002. This finding also illustrated the failure of the licensee to understand maintenance rule scoping criteria as applied for equipment used in the emergency operating procedures.

Adverse Trend in Design Margin and Capability of ac Power Systems

The inspectors concluded that the adverse trend related to the availability, reliability, and capability of station ac power systems, originally identified in December 2008 (described in Section 4OA2 of Inspection Report 05000275/2008005) continued through the first two quarters of 2009. The inspectors concluded that this trend could indicate the existence of a more significant concern because all three of the plant ac power systems were affected:

- Less than adequate 230 kV preferred offsite power system capacity and capability, as described in Section 1R01 of this report

- Less than adequate evaluation and corrective actions for a 500 kV delay offsite power system affect the design basis, as described in Section 4OA5 of this report

- Less than adequate emergency diesel generator testing, as described in Section 4OA7 of this report

- Degraded Unit 2 main transformer bank cooling, May 15, 2009, as described in Notification 60014618

- Degraded Unit 2 main transformer bank cooling, June 30, 2009, resulting a forced reactor shutdown, as described in Notification 60017210

- Degraded Unit 1 main transformer bank cooling, July 2, 2009, as described in Notification 60017228

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

During a review of items entered in the licensee's corrective action program, the inspectors recognized a corrective action item documenting:

- Notification 50202606, conformance with 1971 NRC general design criteria, February 16, 2009

- Notification 50085862, dual unit trip evaluation, November 18, 2008

- Notification 50212570, Unit 2, failure of control rod bank counters, March 17, 2009

These activities constitute completion of three in-depth problem identification and resolution samples as defined in Inspection Procedure 71152-05.

b. Findings

1. Failure to Submit a Licensee Event Report for a Condition Prohibited by the Plant's Technical Specifications

Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a Licensee Event Report within 60 days following discovery of a condition prohibited by technical specifications.

Discussion. Technical Specification 3.1.7, "Rod Position Indication," required that the licensee maintain the demand position indication system operable during power operations. On March 8, 2009, Unit 2 Shutdown Bank B, Group 1, demand bank indicator failed. On March 17, Unit 2 Shutdown Bank B, Group 2, demand bank indicator also failed. At 8:25 a.m., control room operators recognized the existence of a condition prohibited by technical specifications, and entered Technical Specification Limiting Condition for Operations 3.0.3, and initiated action to place the unit in Mode 3 within the next seven hours. At 9:40 a.m., control room operators completed the reactor shutdown brief and were preparing to reduce reactor power. Maintenance technicians completed repairs and returned the indicator to operable at 10:18 a.m. The licensee entered the failed bank counters into the corrective action program as Notification 50212570. On May 26, 2009, the inspectors identified that the licensee had not reported the condition as required by 10 CFR 50.73. The inspectors concluded that the licensee failed to thoroughly evaluate the bank counter failures to ensure that reportability requirements were met.

Analysis. The failure of Pacific Gas and Electric to report a condition prohibited by technical specifications was a performance deficiency. The inspectors evaluated this violation using the traditional enforcement process because the issue affected the NRC's ability to perform its regulatory function. The inspectors concluded that the failure to make a required Licensee Event Report was Severity Level IV violation using the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I - Reactor Operations, dated January 14, 2005. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV noncited violation because the licensee failed to issue a required Licensee Event Report. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the bank indicator to ensure reportability requirements were met [P.1(c)].

Enforcement. Title 10 of the Code of Federal Regulations 50.73(a)(1) required, in part, that the licensee submit a Licensee Event Report for any event of the type described in this paragraph within 60 days after the discovery of the event. Title 10 of the Code of Federal Regulations 50.73(a)(2)(i)(B) required, in part, that the licensee report any operation or condition prohibited by the plant's technical specification. Contrary to the above, the licensee failed to submit a required Licensee Event Report within 60 days after discovery of a condition prohibited by the plant's Technical Specification on March 17, 2009. This is a Severity Level IV noncited violation consistent with

Section 7.10 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy. Because this finding is of very low safety significance and has been entered into the corrective action program as Notification 50242153, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000323/2009003-02, “Failure to Submit a Licensee Event Report for a Condition Prohibited by the Plant’s Technical Specifications.”

.2 Failure to Update the Final Safety Analysis Report Update with Current Plant Design Criteria

Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR Part 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with current plant design criteria.

Description. The inspectors identified that the Final Safety Analysis Report Update failed to include the appropriate design basis for the plant. The failure of the Final Safety Analysis Report Update to reflect the current plant design basis had an adverse impact on the plant modification process, the licensee’s ability to assess operability for degraded plant systems, and the NRC’s ability to ensure that regulatory requirements were met. Final Safety Analysis Report Update, Section 3.1, “Conformance with AEC General Design Criteria,” stated that Diablo Canyon was designed to comply with the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, published in July 1967. Final Safety Analysis Report Update Sections 3.1.1 through 3.1.10 provided a description of how the plant conformed to each of the individual 1967 General Design Criteria. Final Safety Analysis Report Update, Appendix 3.1.A, “AEC General Design Criteria - 1971,” stated that Final Safety Analysis Report Update, Section 3.1 described the plant’s conformance with the 1967 criteria and Appendix 3.1.A provided a discussion of 1971 General Design Criteria “made solely to supply an informative comparison.” Appendix 3.1.A included a discussion of how the plant design met the “intent” of portions of the 1971 General Design Criteria.

The inspectors concluded that Final Safety Analysis Report Update, Chapter 3, was inadequate to describe the Diablo Canyon design bases related to the 1971 General Design Criteria. Pacific Gas and Electric submitted a Letter (To F.J. Miraglia, Division of Licensing, US NRC, from P. A. Crane, Pacific Gas and Electric, CHRON 131464, Description of PG&E’s compliance with the requirements 10 CFR 50, September 10, 1981) to the NRC. This letter stated that Diablo Canyon complies with the 1971 General Design Criteria requirements except for those specific cases where exemptions have been approved by NRC Staff. This letter stated:

> “The General Design Criteria (GDC) provided in 10 CFR 50 Appendix A which were promulgated in 1967 were used as the design criteria for obtaining the Diablo Canyon construction permit. In 1971 a new set of General Design Criteria were promulgated by the AEC. To be responsive to the NRC staff’s requirement to evaluate the Diablo Canyon Power Plant design against present rules and regulations, PG&E has done the evaluation using the GDC promulgated in 1971.”

This submittal included a detailed description of how Pacific Gas and Electric met each of the 1971 General Design Criteria. Also, the NRC Safety Evaluation Report, “Safety

Evaluation by The Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Pacific Gas And Electric Company Diablo Canyon Nuclear Power Station, Units 1 And 2 San Luis Obispo County, California Docket Nos. 50-275 And 50-323," stated:

> "Any exceptions to the 1971 GDC which have been taken because of earlier design or construction commitments are identified in the FSAR in the discussion of the corresponding criterion (see Appendix 3.IA of the FSAR). As a result, our review assessed the plant against the General Design Criteria now in effect (1971), and we have concluded that the plant design conforms to the intent of these newer criteria."

In January 2009, the inspectors identified that the final safety analysis report update did not adequately reflect the licensee's commitment to the NRC acceptance review of the 1971 General Design Criteria. The licensee entered this problem into the corrective action system as Notification 50202606. On April 30, 2009, Pacific Gas and Electric closed the notification without taking corrective action. The licensee concluded that corrective action was not necessary because the 1981 submittal provided the information and did not change the principle design criteria for the plant. The inspectors concluded that while this performance deficiency was related to a latent issue, the licensee had a reasonable opportunity to correct the problem. The inspectors also concluded that the most significant contributor to the finding, the licensee's failure to adequately manage the current licensing basis, is reflective of current plant performance.

Analysis. The failure of Pacific Gas and Electric to maintain the current plant design basis in the Final Safety Analysis Report Updated was a performance deficiency. The inspectors concluded that the finding is more than minor because incorrect Final Safety Analysis Report Update design basis information affected all of the reactor safety cornerstone design control attributes and because the failure to update the Final Safety Analysis Report Update had a material impact on safety and licensee activities. Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The issue was classified as Severity Level IV because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner [P.1(d)].

Enforcement. Title 10 of the Code of Federal Regulations 50.71(e) required Pacific Gas and Electric to periodically update the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal was required to contain all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submittal of the original Final Safety Analysis Report, or as appropriate, the last update to The Final Safety Analysis Report under this section. Contrary to the above, Pacific Gas and Electric failed to update the Final Safety Analysis Report originally submitted to assure that the information detailing compliance to the 1971 General Design Criteria provided to the Commission in letter CHRON 131464. The licensee reopened corrective action program Notification 50202606 to update the Final Safety Analysis Report Update with information included in letter CHRON 131464. Because this finding is of very low safety significance and was entered into the corrective action program, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000275;323/2009003-03, "Failure to Update the Final Safety Analysis Report Update with Current Plant Design Criteria."

4OA3 Event Follow-up (71153)

.1 <u>(Closed) Licensee Event Report 05000275/2009-001-00: Replacement Steam Generator Support Inadequate Due to Improper Washer Plate Installation</u>

On March 22, 2009, Pacific Gas and Electric identified that two of six seismic washer support plates on Steam Generator 1-3 were not seated due to interference with weld material. The inspectors reviewed this issue during the first quarter of 2009 and documented a licensee-identified violation in Section 4OA7 of the Inspection Report 05000275;323/2009002. This Licensee Event Report is closed.

.2 <u>Unusual Event for Smoke Alarm in Unit 1 Containment on June 2, 2009</u>

On June 2, 2009, Pacific Gas and Electric declared an Unusual Event due to a smoke alarm in the protected area inside the Unit 1 containment. The inspectors responded to the site and reviewed the licensee actions with respect to the Site Emergency Plan.

<u>Introduction</u>. The inspectors identified a Green, noncited violation of 10 CFR 50.54(q), after a Pacific Gas and Electric shift manager failed to promptly declare an Unusual Event in accordance with the emergency plan. Diablo Canyon Power Plant Emergency Plan, Appendix D, Section 1, required that the emergency plan be activated after 15 minutes following a fire alarm in the containment building, that could not be validated as false.

<u>Description</u>. Emergency Plan Implementing Procedure EP G-1, "Emergency Classification and Emergency Plan Activation," Revision 39, defined Unusual Event HU2.1 as a fire in buildings or areas, including containment, not extinguished within 15 minutes of control room notification or validation of a control room alarm. Diablo Canyon Power Plant Emergency Plan, Appendix D, Section 1, stated "that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present."

At 12:56 a.m. on June 2, 2009, control room operators received a Unit 1 containment fire alarm. Plant operators were dispatched to the containment building hatch to validate the alarm. However, plant operators were unable to enter the containment airlock because the outer hatch was posted "Airborne Area." The shift manager declared an Unusual Event at 1:22 a.m. At 2:48 a.m. fire brigade personnel entered the containment building and confirmed that the alarm was spurious. At 3:06 a.m. the shift manager terminated the Unusual Event. The inspectors concluded that the shift manager failed to meet the

time requirements of the emergency plan when he did not declare an Unusual Event within 15 minutes of receiving the control room alarm. The shift manager stated that he understood that the emergency plan provided for 15 minutes to validate the fire alarm and an additional 15 minutes to make the Emergency Plan declaration.

Analysis. The inspectors determined that the failure to promptly classify an Emergency Action Level was a performance deficiency. Traditional enforcement does not apply because the issue did not have any actual safety consequences or potential for impacting the NRC's regulatory function and was not the result of any willful violation of NRC requirements or licensee procedures. The finding was more than minor because it is associated with the emergency response organization performance attribute of the emergency preparedness cornerstone and affected the cornerstone objective of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency in that delays in activation and notification of emergency conditions could adversely affect the health and safety of the public. The inspectors determined the finding was associated with an actual event implementation problem, and assessed the significance using Inspection Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process." Using the emergency preparedness significance determination process Sheet 2, "Actual Event Implementation Problem," the inspectors determined the finding was of very low safety significance (Green) because the licensee failed to implement a risk significant planning standard (10 CFR 50.47(b)(4)) during an actual Unusual Event. This finding has a crosscutting aspect in the area of human performance associated with the work practices component because a shift manager failed to implement the time requirments of the emergency plan [H.4(b)].

Enforcement. Title 10 of the Code of Federal Regulations 50.54(q), requires that licensees follow and maintain their emergency plans. Diablo Canyon Power Plant Emergency Plan, Appendix D, Section 1, states in part, “that emergency classifications are to be made as soon as conditions are present and recognizable for the classification, but within 15 minutes or less in all cases of conditions present.” Contrary to the above, on June 2, 2009, because an Unusual Event was not classified within 15 minutes a shift manager failed to follow the Emergency Plan. Specifically, when a Unit 1 containment fire alarm was received and not verified as valid within 15 minutes; conditions had been met for classification as a Notice of Unusual Event in accordance with Emergency Plan Implementing Procedure EP G-1 “Emergency Classification and Emergency Plan Activation.” However, it was not classified until 26 minutes after the alarm was received. Because this finding is of very low safety significance and has been entered into the corrective action program as Notification 50247279, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000275/2009003-04, “Failure to Promptly Declare an Unusual Event.”

4OA5 Other Activities

.1 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors performed observations of security force personnel and activities to ensure that the activities were consistent with Pacific Gas and Electric Company's security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item 05000275/2008005-01, 05000323/2008005-01: Acceptability Assumed 500 kV Delay Time

a. Inspection Scope

The inspectors previously identified an unresolved item related to the acceptability of the assumed delay time needed for plant operators to align the alternate offsite power source to the engineered safety feature bus following a loss of the 230 kV offsite power source. The licensee stated that the 500 kV power source can be made available by manual initiation in approximately 30 seconds, as described in NRC Safety Evaluation Report, "Safety Evaluation By The Directorate of Licensing U.S. Atomic Energy Commission In The Matter of Pacific Gas And Electric Company Diablo Canyon Nuclear Power Station, Units 1 And 2 San Luis Obispo County, California Docket Nos. 50-275 And 50-323." The NRC initially concluded that the alternate offsite power source was acceptable because the circuits provided sufficient assurance that redundant and independent sources of offsite power are provided, as required by General Design Criteria 17. The licensee increased the delay time to 30 minutes in Revision 12 of Final Safety Analysis Report Update. This issue was unresolved pending NRC review to verify that the 500 kV design basis was still met with the increase in delay time.

On December 31, 2008, the licensee completed a thermal hydraulic analysis, Calculation STA-274, "RETRAN Evaluation of GDC-17 Loss of AC Scenario," Revision 0, examining the plant response during the delay time. The new analysis concluded that about 1,800 gallon of reactor coolant inventory would be lost during the first 30 minutes and up to 4,500 gallons for an hour, following the loss of reactor coolant pump seal cooling and injection. On May 27, 2009, Pacific Gas and Electric concluded that prior NRC approval was required for the increase in delay time. The NRC acceptability review of the alternate offsite power source design will be performed by the licensee amendment process.

This unresolved item is closed.

b. Findings

.1 Inadequate Corrective Actions Following the Loss of Design Control for the 500 kV Offsite Power Source

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, “Corrective Action,” after Pacific Gas and Electric failed to adequately correct a nonconforming condition related to the design control of the 500 kV qualified offsite power system.

Description. The inspectors identified that Pacific Gas and Electric failed to implement adequate corrective actions following discovery that design control measures for the 500 kV alternate offsite power system were not retrievable. The alternate offsite power circuit design basis included General Design Criteria 17, “Electric Power Systems,” as defined in NRC Safety Evaluation Report, “Safety Evaluation by the Directorate of Licensing U.S. Atomic Energy Commission in the Matter of Pacific Gas and Electric Company Diablo Canyon Nuclear Power Station, Units 1 and 2 San Luis Obispo County, California Docket Nos. 50-275 And 50-323.” General Design Criteria 17 required the alternate delayed supply be aligned to the engineering safety features electrical buses in sufficient time to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded, assuming the loss of all onsite alternating current power supplies and the preferred offsite electric power circuit. Final Safety Analysis Report Update, Section 8.2, “Offsite Power System,” stated that the alternate offsite transmission circuit could be aligned to the safety related buses within 30 minutes.

On August 26, 2008, the inspectors reviewed the design control measures demonstrating that the alternate offsite power source met the design basis. The licensee was required to maintain this documentation by 10 CRF Part 50, Appendix B, Criteria III, “Design Control.” The licensee stated that the requested documents were not retrievable and entered the nonconforming condition into the corrective action program as Notification 50083989. On October 28, 2008, plant engineers reevaluated the alternate offsite power system and concluded General Design Criteria 17 compliance by a “road map.” The “road map” consisted of pre-existing quality related analyses created to support other plant design bases.

The inspectors identified that the “road map” was less than adequate to demonstrate that the 500 kV offsite power system time was in compliance with the design basis. A loss of reactor coolant pump seal cooling and injection is anticipated to occur at the beginning of the delay time and last until operators align the alternate offsite power supply to the engineering safety feature buses. The licensee’s “road map” failed to analyze the affect the resulting loss of reactor coolant through the reactor coolant pump seals had on the design basis. NRC Information Notice 2005-14, “Fire Protection Findings on Loss of Seal Cooling to Westinghouse Reactor Coolant Pumps,” and

Westinghouse Technical Bulletin, TB-04-22, “Reactor Coolant Pump Seal Performance, Appendix R Compliance and Loss of All Seal Cooling,” Revision 1, provided deterministic and probabilistic models for reactor coolant pump seal performance following the loss of seal injection and cooling. Based on this information, the inspectors estimated that a loss of reactor coolant inventory of about 12 gallons per minute, during the first 8 minutes, and increasing to about 88 gallons per minute, would be lost through the reactor coolant pump seals until seal injection could be reestablished.

The failure of Pacific Gas and Electric to maintain documentation demonstrating the alternate offsite delayed power source met the design basis was an old design issue. The inspectors concluded that the failure of the licensee to promptly correct the non-conforming condition and to ensure that the “road map” implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance.

Analysis. The finding is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors concluded that the failure of the licensee to promptly correct the nonconforming condition and to ensure that the “road map” implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance. The inspectors used Inspection Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” to analyze the significance of this finding. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the problem of missing documentation to ensure that the design basis for the alternate offsite power source was met [P.1(c)].

Enforcement. Title 10 of Code of Federal Regulations, Part 50, Appendix B, Criteria XVI, “Corrective Action,” required that Pacific Gas and Electric establish measures to assure that conditions adverse to quality, including nonconformances, are promptly identified and corrected. Contrary to the above, Pacific Gas and Electric did not establish measures to assure that a condition adverse to quality, a nonconformance, was corrected. On October 28, 2008, the licensee failed to adequately correct a nonconformance related to missing documentation demonstrating that the alternate offsite power source met the design basis. The licensee also failed to ensure that the “road map” design was verified or checked. Because this finding is of very low safety significance and was entered into the corrective action program as Notification 50083989, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000275;323/2009003-05, “Inadequate Corrective Actions Following the Loss of Design Control for the 500 kV Offsite Power Source.”

.2 Failure to Evaluate a Change to the Facility as Described in the Final Safety Analysis Report Update Associated with the Alternate Offsite Power Source

Introduction. The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an evaluation of a thermal hydraulic analysis demonstrating acceptable plant response during the alternate offsite delayed power source delay time. This analysis was used by the licensee to support the conclusion that the 500 kV offsite power system met the design basis.

Description. On December 31, 2008, a Pacific Gas and Electric 50.59 screen failed to determine that Calculation STA-274, “RETRAN Evaluation of GDC-17 Loss of AC Scenario,” Revision 0, was required to be evaluated to determine if prior NRC approval was required. Calculation STA-274 verified that the delayed offsite power source met General Design Criteria 17, “Electric Power Systems,” design basis. Calculation STA-274 included a thermal hydraulic analysis of a new plant transient resulting from the anticipated loss of reactor coolant pump seal cooling and injection during the delay time while plant operators aligned the 500 kV offsite power system to the engineering safety feature buses. Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” stated that NEI 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” Revision 1 provides methods that are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. The licensee’s screen concluded that a 50.59 evaluation was not required because Calculation STA-274 did meet the NEI-96-07 criteria to be screened out from an evaluation.

On March 31, 2009, the inspectors concluded that Calculation STA-274 was required to be evaluated for prior NRC approval. The analysis described significant loss of reactor coolant system inventory through the reactor coolant pump seals during the delay time needed for 500 kV power alignment. Because the loss of inventory introduced an unwanted or previously unreviewed system or material interaction, NEI 96-07 required that the licensee perform an evaluation. NEI 96-07 screening criteria also required an evaluation because the analysis included a change that had the potential to increase the likelihood of a malfunction of the reactor coolant pump seals, which potentially increased the consequences of or created a new accident. The licensee entered the failure to complete a 50.59 evaluation into the corrective action system as Notification 50228928. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required because the new analysis:

- Created a possibility for a malfunction of structures, systems, and components important to safety with a different result from previously evaluated in the Final Safety Analysis Report Update

- Resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update used in establishing the design bases or in the safety analyses

Analysis. The failure of Pacific Gas and Electric to perform a 50.59 evaluation of Calculation STA-274, in accordance with NEI 96-07, was a performance deficiency. The inspectors concluded that the finding is more than minor because the changes made to the facility as described in the final safety analysis report update, required prior NRC review and approval. The inspectors evaluated the significance of the finding using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," to analyze the significance of this finding. The finding affected the Mitigating Systems Cornerstone because the change described how alternate delayed offsite power source met the design basis. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency that did not result in the loss of operability or functionality. Because the issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated this finding using the traditional enforcement process. The issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1(c)].

Enforcement. Title 10 of the Code of Federal Regulations, Part 50.59(c)(1) requires, in part, that a licensee may make changes in the facility as described in the Final Safety Analysis Report without obtaining a license amendment only if a change to the technical specifications incorporated in the license is not required. Contrary to the above, on December 31, 2008, Pacific Gas and Electric made changes to the facility as described in the Final Safety Analysis Report without obtaining a license amendment. Specifically, Pacific Gas and Electric approved an evaluation that demonstrated the 500 kV offsite power source met the intended functions that also created the possibility for a malfunction of the reactor coolant system with a different result than any previously evaluated in the Final Safety Analysis Report Update and resulted in a departure from the method of evaluation described in the Final Safety Analysis Report Update to establish the design. The licensee has initiated corrective actions to submit the change to the NRC for approval. Because this finding is of very low safety significance and was entered into the corrective action program as Notification 50228928, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000275;323/2009003-06, "Failure to Evaluate a Change to the Facility as Described in the Final Safety Analysis Report Update Associated with 500 kV Offsite Power Source."

4OA6 Meetings

Exit Meeting Summary

On May 26, 2009, the inspectors conducted a telephonic exit meeting to present the results of the in-office inspection of changes to the licensee's emergency action levels to Ms. M. Zawalick, Senior Emergency Planning Coordinator, Mr. S. Hamilton, Supervisor Regulatory Services, and other members of the licensee's staff. The licensee acknowledged the issues presented.

On June 29, 2009, the inspectors presented the inspection results to Mr. J. Becker, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a noncited violation.

- Title 10 of the Code of Federal Regulations 50, Appendix B, Criteria XI, “Test Control,” required Pacific Gas and Electric to establish a test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to this, on March 9, 2009, Pacific Gas and Electric discovered that emergency diesel generator surveillance testing did not reflect actual design basis loading conditions. Engineers discovered that the power factor requirements of Surveillance Test Procedures STP M-9G, “Diesel Generator 24-Hour Load Test and Hot Restart Test,” and STP M-9D1, “Diesel Generator Full Load Rejection Test;” and Technical Specification Surveillance Requirements SR 3.8.1.10 and SR 3.8.1.12 were not adequate, based on reviews of Calculations 015-DC, “Diesel Generator Loading for 4160V Vital Bus Loads, Unit 1,” Revision 19 and 125-DC, “Diesel Generator Loading for 4160V Vital Bus Loads, Unit 2,” Revision 13. Pacific Gas and Electric entered SR 3.0.3 and implemented administrative controls in accordance with NRC Administrative Letter 98-10, “Disposition of Technical Specifications that are Insufficient to Assure Plant Safety.” Pacific Gas and Electric revised the surveillance test procedures and successfully completed the surveillance tests for all emergency diesel generators June 21, 2009. The licensee entered this condition into the correction action program as Notification 50232184, “Diesel Generator Power Factor.” This finding is of very low safety significance because the condition did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Becker, Site Vice President
W. Guldemond, Director, Site Services
S. Ketelsen, Manager, Regulatory Services
K. Peters, Station Director
M. Somerville, Manager, Radiation Protection
T. Swartzbaugh, Manager, Operations
J. Welsch, Director, Operations Services

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

05000275/2008005-01; 05000323/2008005-01	URI	Acceptability Assumed 500 kV Delay Time (Section 4OA5)
05000275/2009-001-00	LER	Replacement Steam Generator Support Inadequate due to Improper Washer Plate Installation

LIST OF DOCUMENTS REVIEWED

Section 1RO4: Equipment Alignments

PROCEDURES

NUMBER	TITLE	REVISION
STA-220	RHR System Pressurization Due to INPO OE 20893SBLOCA Scenario	0
DCM S-10	Residual Heat Removal System	17
P&ID 106710	Residual Heat Removal System	38

ACTION REQUEST

A0643107

Section 1R011: Licensed Operator Requalification Program

PROCEDURES

NUMBER	TITLE	REVISION / DATE
ECA00-A	Instructor Lesson Guide, Loss of All AC	4/3/09
E3ECA33-D	Instructor Lesson Guide, Steam Generator Tube Rupture	4/3/09
OP1.DC10	Conduct of Operations	16A

Section 1R12: Maintenance Effectiveness

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
MA1.ID17	Maintenance Rule Monitoring Program	21
System Report	System 41 – Rod Control Units 1 and 2	5/6/09
PC-41B-01	Maintenance rule scoping worksheet – rod control system	4
PC-23A1-01	Maintenance rule scoping worksheet – CFCUs	4
Meeting Minutes	MR Expert Panel Meeting 156 Minutes	1/22/09
Summary Report	(a)(1) Goal Setting Summary Report – System 23A1 – CFCUs – Unit 2	2/16/09

ACTION REQUESTS

50232567 50212570 50044669

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

PROCEDURES

NUMBER	TITLE	REVISION / DATE
50232136	SR 3.0.3 extension based on draft calculations	4/13/2009
50231727	Document special PRA eval week 0916	4/11/2009
Calc PRA 09-02	Evaluation of the PRA Impact of Missed Surveillances Due to Emergency Diesels tested outside Tech Spec required parameter	0
TS3.NR1	Probabilistic Risk Assessment	5
09-06	Risk Assessment for Signal Component MOW Week 0920 with Elevated Risk Due to EDGs SR 3.0.3	1
09-06	Risk Assessment for Signal Component MOW Week 0920 with Elevated Risk Due to EDGs SR 3.0.3	2

NOTIFICATION

50234252

Section 1R15: Operability Evaluations

PROCEDURES

NUMBER	TITLE	REVISION
STP M-9D1	Diesel Generator Full Load Rejection Test	13
STP M-9G	Diesel Generator 24-Hour Load Test and Hot Restart Test	44
STP P-ASW-21	Routine Surveillance Test of Auxiliary Saltwater Pump 2-1	26
Calc M-270	Flood Level of the Auxiliary Saltwater Pump Vault Due to Pipe	6

	Crack and Floor Drain Plugging	
OM7.ID12	Operability Determination	12

NOTIFICATIONS

50237951	50244173	50248314	50248315	50248319
50248207				

Section 1R19: Postmaintenance Testing

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
WP 1-4520A	SG 1-1 Hot Gap Measurements	3/25/09
WP 1-4520B	SG 1-2 Hot Gap Measurements	3/24/09
WP 1-4520C	SG 1-3 Hot Gap Measurements	3/25/09
WP 1-4520D	SG 1-4 Hot Gap Measurements	3/24/09
68001000	LT-501 (STP)I-4-L501 CAL SG1 WR LVL	3/11/09
68001406	LT-502 (STP)I-4-L502 CAL SG2 WR LVL	3/13/09
68000789	LT-503 (STP)I-4-L503 CAL SG3 WR LVL	3/8/09
68000810	LT-504 (STP)I-4-L504 CAL SG4 WR LVL	3/7/09
68001401	LT-517 (STP)I-4-L517 CAL SG1 NR LVL	3/12/09
68001402	LT-518 (STP)I-4-L518 CAL SG1 NR LVL	3/12/09
68001404	LT-519 (STP)I-4-L519 CAL SG1 NR LVL	3/12/09
68001407	LT-527 (STP)I-4-L527 CAL SG2 NR LVL	3/13/09
68001409	LT-528 (STP)I-4-L528 CAL SG2 NR LVL	3/15/09
68001411	LT-529 (STP)I-4-L529 CAL SG2 NR LVL	3/13/09
68000795	LT-537 (STP)I-4-L537 CAL SG3 NR LVL	3/10/09
68000800	LT-538 (STP)I-4-L538 CAL SG3 NR LVL	3/10/09
68000806	LT-539 (STP)I-4-L539 CAL SG3 NR LVL	3/11/09
68000816	LT-547 (STP)I-4-L547 CAL SG4 NR LVL	3/10/09
68000842	LT-548 (STP)I-4-L548 CAL SG4 NR LVL	3/10/09
68000846	LT-549 (STP)I-4-L549 CAL SG4 NR LVL	3/10/09
PMT 37.03	RSG Functional Test: Verification of Full Power Tref	0
STP R-26	RCS Primary Coolant Flow Measurements	28
PMT 04.30	Steam Generator Replacement Testing	0
PMT 04.28	RSG Warranty Test: Moisture Carryover	0

NOTIFICATIONS

50237641	50236391

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION / DATE
STP R-4	Evaluation of Core Reactivity	15
STP P-RHR-12	Routine Surveillance Test of RHR Pump 1-2	22
STP P-CSP-12	Routine Surveillance Test Containment Spray Pump 1-2	12
STP P-CSP-12	Routine Surveillance Test Containment Spray Pump 2-2	10A
STP M-39B	Routine Surveillance Test of Cable Spreading Room Carbon Dioxide Fire System Operation	23
50233728	Failed U1 CSR CO2 Test STP M-39B	April 21, 2009
STP M-9G	Diesel Generator 24-Hour Load Test and Hot Restart Test	47

Section 1EP4: Emergency Action Level and Emergency Plan Changes

PROCEDURES

NUMBER	TITLE	REVISION
EP G-1	Emergency Classification and Emergency Plan Activation	37A
EP G-1	Emergency Classification and Emergency Plan Activation	38

ACTION REQUEST

50205989

Section 1EP6: Drill Evaluation

DOCUMENTS

NUMBER	TITLE	REVISION
FRS1-B	ATWS	14

Section 4OA2: Identification and Resolution of Problems

PROCEDURES

NUMBER	TITLE	REVISION
EP G-1	Emergency Classification and Emergency Plan Activation	37A
EP G-1	Emergency Classification and Emergency Plan Activation	38

NOTIFICATIONS

50203149 50232567 50212570