UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

February 7, 2011

John T. Conway
Senior Vice President and
Chief Nuclear Officer
Pacific Gas and Electric Company
77 Beale Street, B32
San Francisco, CA 94105

Subject: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2010005 AND 05000323/2010005

Dear Mr. Conway:

On December 31, 2010, 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 January 4, 2011, 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 six findings of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. Additionally, five licensee-identified violations, which were determined to be of very low safety significance, are listed in this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these findings as noncited violations, consistent with Section 2.3.2 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 crosscutting aspect assigned to 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.

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/

Donald Allen, Chief
Project Branch B
Division of Reactor Projects

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

Enclosure:

NRC Inspection Report 05000/275/2010005 and 0500323/2010005
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MSPeck	MABrown	RDeese	TRFarnholtz	NFO'Keefe
/DBA for=E/	**/DBA for=E/**	**/RA/**	**/RA/**	**/RA/**
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C:DRS/OB	C:DRS/PSB1	C:DRS/PSB2	C:DRS/TSB	C:DRP/
MSHaire	MPShannon	GEWerner	MCHay	DBAllen
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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000275, 05000323
License:	DPR-80, DPR-82
Report:	05000275/2010005 05000323/2010005
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:	September 26 through December 31, 2010
Inspectors:	M. Peck, Senior Resident Inspector M. Brown, Resident Inspector J. Drake, Senior Reactor Inspector L. Ricketson, P.E., Senior Health Physicist C. Graves, Health Physicist R. Kopriva, Senior Reactor Inspector, Engineering Branch 1 I. Anchondo, Reactor Inspector, Plant Support Branch 2 P. Goldberg, Reactor Inspector, P.E. D. Reinert, Reactor Inspector C. Osterholtz, Senior Operations Engineer B. Latta, Senior Reactor Inspector, Engineering Branch 2
Accompanied By:	E. Wong, Chemical Engineer, Steam Generator Integrity and Chemical Engineering Branch C. Alldredge, Health Physicist, Nuclear Safety Professional Development Program Participant
Approved By:	D. Allen, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000275/2010005, 05000323/2010005; 9/26/2010 – 12/31/2010; Diablo Canyon Power Plant, Integrated Resident and Regional Report; Fire Protection; Operability Evaluations; Plant Modifications, Postmaintenance Testing, and Other Activities.

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by regional based inspectors. Six Green noncited violations of significance 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 Findings and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain the integrity of Door 155 in the rated condition. On December 9, 2010, the inspectors identified that the fire door was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain Door 155 in a configuration that would provide at least a 1½-hour rated fire barrier. The inspectors previously identified that Door 155 was degraded as a fire barrier in 2009. The licensee entered the violation into the corrective action program as Notification 50367381 and took immediate corrective actions to restore the fire barrier to the rated condition and to implement weekly plant fire door walkdowns.

  The inspectors concluded that the finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute and objective to prevent undesirable consequences due to fire. The inspectors determined that the finding was within the fire confinement category and that the fire barrier was moderately degraded. The inspectors concluded that the finding was of very low safety significance (Green) because there was a non-degraded automatic full area water-based suppression system in the exposed fire area. 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 effective corrective actions to following the previous occurrence of the violation [P.1(d)]. (Section 1R05)

- Green. The inspectors identified a noncited violation of Diablo Canyon Unit 2 Facility Operating License Condition 2.C.(5), “Fire Protection,” after Pacific Gas and Electric failed to ensure procedures for controlling flammable and combustible materials adequately incorporated requirements of the fire hazard analysis. On October 18, 2010, the inspectors identified that transient combustible materials staged in the Unit 1 12 kilovolt switchgear room did not have an approved transient combustibles permit. The licensee stated that the combustibles permit procedure did not require a permit for the room while Unit 1 was shutdown. However, the plant fire hazards design basis described safe shutdown equipment in the room that would be needed to support a safe shutdown of the operating unit, specifically the Unit 2 startup bus located in the room. The inspectors determined that the licensee's transient combustibles permit procedure was inadequate because the procedure did not require a permit for the Unit1 12 kilovolt switchgear room when Unit 2 was operating. The licensee entered the issue into the corrective action program as Notification 50366302 and performed an evaluation of the transient combustibles stored in the area.

The inspectors concluded that this finding was more than minor because it affected the Mitigating Systems Cornerstone external factors attribute objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors determined that the finding was within the fire prevention and administrative controls category and represented a low degradation level due to the minimal impact on the effectiveness and reliability of the affected systems. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening, the low degradation rating, and only equipment needed to reach and maintain cold shutdown conditions was affected. This finding had a crosscutting aspect in the area of human performance associated with the resources component because the licensee failed to ensure that the design documentation adequately identified the Unit 2 startup bus as equipment required for safe shutdown for Unit 2 [H.2(c)]. (Section 1R05)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to adequately evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, "Operability Determination." On October 15, 2010, the inspectors identified a less than adequate technical evaluation supporting Prompt Operability Assessment 50350918, "Unit 2 - Insulation in Bio-Wall Penetration." Engineering personnel failed to adequately evaluate the extent of condition after technicians identified about 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous insulation in the Unit 1 reactor coolant loop biological shield wall penetrations. This fibrous material could have potentially been transported and plugged the emergency core cooling containment sump screen. The licensee performed the prompt operability assessment for Unit 2, which was operating at the time. The inspectors concluded that the engineering personnel inappropriately applied the leak-before-break methodology to exclude about 87 percent of this material from the extent of condition review in the prompt operability assessment.

  The second example involved Prompt Operability Assessment Notification 50355265, "RHR Sump Margin," which was completed by the licensee on October 23, 2010. In this example, engineering personnel failed to identify and demonstrate that the specified safety function of the refueling water storage tank could be maintained as required by the plant operability procedure. The inspectors identified that the post accident flow path from the reactor cavity to the containment sump was blocked by a large shield plug. This blockage reduced the amount of post accident inventory available at the containment sump at the time of transition from injection to recirculation mode of emergency core cooling operation. Engineering personnel failed to demonstrate that the safety function to ensure full sump submergence was maintained with the blocked flow path. Full submergence of the sump was used by the NRC as the basis for approval of Technical Specification 3.5.4, "Refueling Water Storage Tank," inventory requirements. The licensee entered the violation into the corrective action program as Notification 50369117 and revised the prompt operability assessments using assumptions consistent with the current licensing bases.

The inspectors concluded that the performance deficiency was more than minor because the finding affected 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. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because Pacific Gas and Electric did not use conservative assumptions in decisions to demonstrate component operability in either example [H.1(b)]. (Section 1R15)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to ensure Calculation STA-255, "Minimum Required Refueling Water Storage Tank Level for GE Sumps," Revision 2, demonstrated adequate available refueling water storage tank inventory. On October 19, 2010, the inspectors identified that emergency core cooling post accident flow path from the reactor cavity to the containment sumps was blocked by a large steel plug on Unit 1. The accident analysis assumed this 35 square foot path was open to allow coolant from a pipe break inside the biological shield to communicate with containment sumps during the recirculation mode of emergency core cooling. The licensee credited the inventory from the reactor cavity when determining the minimum required refueling water storage tank volume in Calculation STA-255. Pacific Gas and Electric used Calculation STA-255 as the basis for determining the minimum required refueling water storage tank volume specified by Technical Specification 3.5.4, "Refueling Water Storage Tank." The inspectors identified that the recirculation flow path was also blocked on Unit 2. The inspectors concluded that the most significant contributor to the violation was inaccurate plant drawings used by plant engineers during the performance of Calculation STA-255. The licensee's corrective actions included completion of a prompt operability assessment justifying continued operation of Unit 2 and replacement of the shield plug with a movable platform on Unit 1 prior to plant restart.

  The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone plant modification design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the performance deficiency involved a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the resources component because Pacific Gas and Electric failed to use complete, accurate and up-to-date drawing for Calculation STA 255 [H.2(c)]. (Section 1R18)

- Green. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," after Pacific Gas and Electric failed to develop and implement an adequate testing program for the emergency diesel generators that met design requirements and recommendations. Specifically, in December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design bases were met. Pacific Gas and Electric updated the load calculations, but failed to make the necessary revisions to Surveillance Test Procedure STP M-9D1, "Diesel Generator Full Load Rejection Test." As a result, Pacific Gas and Electric failed to test several of the emergency diesel generators at the complete load as required by Regulatory Guide 1.108, Revision 1, which is part of the current licensing bases. The licensee entered this into the corrective action program as Notification 50368801, determined there was no loss of safety function for the affected components, and applied the provisions of Surveillance Requirement 3.0.3 for a missed surveillance test. The inspectors concluded the most significant contributor to the finding was less than adequate diesel generator loading evaluations to support corrective action from previous violations associated with the emergency diesel generator testing.

  The inspectors concluded that the performance deficiency was more than minor because the finding affected the equipment control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative surveillance test such that the resolution addressed the fundamental basis for the surveillance [P.1(c)]. (Section 1R19)

- Green The inspectors identified a noncited violation of 10CFR Appendix B, Criterion XVIII, "Audits", which required that a comprehensive system of planned and periodic audits be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program as well as follow up action, including re-audit of deficient areas, where indicated. Contrary to this requirement, Pacific Gas and Electric failed to ensure that a comprehensive system of planned and periodic audits were carried out to verify compliance with all aspects of the quality assurance program, determine the effectiveness of the program, and perform necessary follow up actions. Specifically, the 2008 Quality Verification audit of the corrective action program failed to adequately address an adverse trend in the problem evaluation process documented in NRC Inspection report 2008005, which identified eleven examples of an adverse trend in problem evaluation. The licensee entered this into their corrective action program as Notification 50365083 and determined there was no loss of safety function for the affected components. The inspectors concluded the most significant contributor to the finding was a less than adequate evaluation of the corrective action trending program.

This finding was more than minor because it was associated with the equipment control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the performance deficiency was of very low safety significance (Green) it was a 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 the licensee failed to coordinate and communicate the results from assessments to affected personnel, and track the corrective actions to address issues commensurate with their significance [P.3(c)].

B. Licensee-Identified Violations

Violations of very low safety significance, which were identified by the licensee, have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers (Notification report numbers) are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

Pacific Gas and Electric Company operated Diablo Canyon Units 1 and 2 at full power at the beginning of the inspection period. Plant operators shutdown Unit 1 for refueling on October 2, 2010. The licensee returned Unit 1 to power operations on November 13, 2010. On November 22, 2010, plant operators reduced Unit 1 to 78 percent power for corrective maintenance on a main feed pump turbine. The licensee restored the unit to full power the following day. Plant operators reduced Unit 2 to 50 percent power to support main condenser circulating water maintenance on November 29, 2010. The licensee restored the unit to full power on December 3, 2010.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors performed a detailed review of the licensee's procedures and preparations for operating the facility during an extended period when ambient outside temperature was high and plant electrical equipment was experiencing elevated temperatures. The inspectors focused on plant-specific design features and implementation of the procedures for responding to or mitigating the effects of these conditions on the operation of the facility's normal and emergency power systems during extreme heat. Inspection activities included a review of the licensee's adverse weather procedures, daily monitoring of the off-normal environmental conditions, and that operator actions specified by plant-specific procedures were appropriate to ensure operability of the facility's normal and emergency cooling systems. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one readiness for impending adverse weather condition sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness for Seasonal Extreme Weather Conditions

a. Inspection Scope

The inspectors performed a review of the adverse weather procedures for winter storm season preparations. The inspectors verified that weather-related equipment deficiencies identified during the previous year were corrected prior to the onset of seasonal extremes, and evaluated the implementation of the adverse weather preparation procedures and compensatory measures for the affected conditions before the onset of, and during, the adverse weather conditions.

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Final Safety Analysis Report Update (FSARU) and performance requirements for the systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the attachment. The inspectors also reviewed corrective action program items to verify that plant personnel were identifying adverse weather issues at an appropriate threshold and entering them into the corrective action program in accordance with station corrective action procedures. The inspectors' reviews focused specifically on the following plant systems:

- Units 1 and 2, Auxiliary saltwater system and circulating water system
- Units 1 and 2, Intake structure

These activities constitute completion of one readiness for seasonal adverse weather sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

.3 Readiness to Cope with External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the FSARU for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed an inspection of the protected area to identify any modification to the site that would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of one external flooding sample as defined in Inspection Procedure 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignments (71111.04)

Partial Walkdown

a. Inspection Scope

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

- Unit 2, Auxiliary feedwater system, October 21, 2010
- Unit 2, Auxiliary building ventilation system, November 15, 2010
- Unit 1, Centrifugal charging pump, December 22, 2010

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, FSARU, 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 inspected 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 in Inspection Procedure 71111.04-05.

b. Findings

No findings 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:

- Fire Zone 14-E, Unit 1, Component cooling water heat exchanger room, October 1, 2010

- Fire Zones 3-T-1 and 3-T-2, Unit 2, Auxiliary feedwater pump rooms, October 18, 2010
- Fire Area 10, Unit 1, 12kV switchgear room, October 18, 2010
- Fire Area 9, Unit 1, Containment, October 22, 2010
- Fire Area 3-CC, Unit 2, Containment penetration area, November 16, 2010
- Fire Area 4-B, Unit 1, December 9, 2010

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 the 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 six quarterly fire protection inspection samples as defined in Inspection Procedure 71111.05-05.

b. Findings

.1 Failure to Maintain a Fire Barrier

Introduction. The inspectors identified a Green noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), Fire Protection," after Pacific Gas and Electric failed to maintain Fire Door 155 in the rated condition.

Description. On December 9, 2010, the inspectors identified that Fire Door 155 was inoperable. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to maintain Door 155 in a configuration that would provide a 1½-hour rated barrier between Fire Areas 4B and S-2. The door was inoperable because the latching mechanism was disengaged. Engagement of the latching was required for the door to perform the rated fire barrier function. The door included clear signage stating that the latch must be engaged. The licensee took immediate corrective action to restore the fire barrier to the rated condition and implemented weekly plant fire door walkdown inspections. The licensee also planned to further evaluate the cause of the violation by completing an apparent cause evaluation. The inspectors previously identified that Fire Door 155 was inoperable on September 1 and September 16, 2009. The inspectors dispositioned these previous occurrences of the violation as

NCV 05000275;323/2009004-01, “Failure to Identify and Correct a Degraded Fire Barrier.” The inspectors concluded the most significant contributor to the violation was less than effective corrective actions following the previous violation.

Analysis. The failure of Pacific Gas and Electric to maintain Door 155 in the rated configuration was a performance deficiency. This finding is more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors used the Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” to analyze this finding. The inspectors determined that the inoperable door was a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated function. The inspectors concluded that the finding was of very low safety significance (Green) because there was a non-degraded automatic full area water-based fire suppression system in the exposed fire area. 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 following the previous violation to address the safety issue in a timely manner [P.1(d)].

Enforcement. Diablo Canyon Facility Operating License DPR-80/DPR-82, License Condition (5), “Fire Protection,” required Pacific Gas and Electric to implement and maintain all provisions of the approved fire protection plan as described by the FSARU. FSARU, Appendix 9.5a, “Fire Hazards Analysis,” and Equipment Control Guideline 18.7, required that the licensee maintain Door 155 as an operable fire area barrier or to implement compensatory actions. Contrary to the above, on December 9, 2010, the inspectors identified that plant personnel failed to maintain Door 155 as an operable fire barrier or implement compensatory actions. Because this finding was of very low safety significance and was entered into the corrective action program as Notification 50367381, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275; 323/201005-01, “Failure to Maintain a Fire Barrier.”

.2 Inadequate Transient Combustibles Procedure

Introduction. The inspectors identified a Green noncited violation of Diablo Canyon Facility Operating License Condition 2.C.(5), “Fire Protection,” after Pacific Gas and Electric failed to ensure that flammable and combustible material control procedures required fire protection engineering review and approval prior to the storage of transient combustibles near safe shutdown equipment.

Description. On October 18, 2010, the inspectors identified transient combustible materials staged in the Unit 1 12 kilovolt switchgear room without an approved Transient Combustibles Permit. The licensee responded that Procedure OM8.ID4, “Control of Flammable and Combustible Materials,” did not require a permit for the area when the Unit 1 was shutdown. The inspectors identified that FSARU, Appendix 9.5G, “Equipment Required for Safe Shutdown,” included the Unit 2 Startup Transformer, as equipment required to bring the plant to a cold shutdown condition as defined by 10 CFR Part 50, Appendix R, Section III.G, but failed to also identify the Unit 2 Startup Bus, located in the Unit 1 12 kilovolt switchgear room. Appendix 9.5H, “Inspection and Testing Requirements and Program Administration,” Section E.1, stated, “Use of combustibles in safety-related areas is to be strictly controlled and is the responsibility of the area or work supervisor." Section E.1 specified that these controls are implemented in plant procedures. Procedure OM8.ID4, "Control of Flammable and Combustible Materials," implemented these requirements to ensure transient combustible materials do not exceed the loadings specified in the fire hazards analysis design. Procedure OM8.ID4, Section 5.6.4.5 identified fire areas for Unit 1 and Unit 2 that require fire protection engineering review and approval prior to storage of any transient or in-situ combustible materials. However, Procedure OM8.ID4 excluded the Unit 1 12 kilovolt switchgear room for Unit 2 operations. The licensee performed an evaluation of the transient combustibles stored in the area and concluded that the loading was acceptable. The licensee entered the issue into the corrective action program as Notification 50366302. The inspectors concluded that the most significant contributor to the finding was that the FSARU, Appendix 9.5G, did not identify equipment in the Unit 1 12 kilovolt switchgear room required for Unit 2 safe shutdown.

Analysis. The inspectors concluded that the licensee's failure to ensure Procedure OM7.ID4, "Control of Flammable and Combustible Materials," was adequate to control all safe shutdown areas was a performance deficiency. The inspectors concluded this finding was more than minor because it affected the Mitigating Systems Cornerstone external factors attribute and objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors used Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," to analyze this finding. The inspectors determined that the inadequate procedure was a fire prevention and administrative controls category finding and assigned a low degradation because of the minimal impact on the effectiveness and reliability of safe shutdown systems. The inspectors concluded the finding was of very low safety significance based on a qualitative screening and because the finding was assigned a low degradation rating and only affected the ability to reach and maintain cold shutdown conditions. This finding had a crosscutting aspect in the area of human performance associated with the resources component because the licensee failed to ensure that design documentation adequately identified the startup bus as equipment required for Unit 2 safe shutdown, which resulted in plant procedures not requiring an approved transit combustibles permit. [H.2(c)]

Enforcement. Diablo Canyon Facility Operating License DPR-82, License Condition 2.C (5), "Fire Protection," required Pacific Gas and Electric to implement and maintain in effect all provisions of the approved fire protection plan as described by the FSARU. FSARU, Appendix 9.5H, "Inspection and Testing Requirements and Program Administration," Section E.1, states "Use of combustibles in safety-related areas is to be strictly controlled and is the responsibility of the area or work supervisor. Specific controls are delineated in plant procedures." Quality related plant Procedure OM8.ID4, "Control of Flammable and Combustible Materials," specified administrative controls required to keep bulk transient combustible materials within the plant fire hazards analysis design basis. Section 5.6.4.5 identified fire areas for Unit 1 and Unit 2 that require fire protection engineering review and approval prior to storage of any transient or in-situ combustible materials. Contrary to this, Procedure OM8.ID4 failed to identify that Fire Area 10 would require fire protection engineering review and approval prior to storage of transient combustible materials while Unit 2 was operating and Unit 1 was shutdown. As a result, transient combustibles were stored in Fire Area 10 without fire protection engineering review and approval. Because this finding was of very low safety significance and was entered into the corrective action program as Notification 50366302, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000323/201005-02, Inadequate Transient Combustibles Procedure.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed the FSARU, the flooding analysis, and plant procedures to assess seasonal susceptibilities involving internal flooding; reviewed the FSARU and corrective action program to determine if licensee personnel identified and corrected flooding problems; inspected underground bunkers/manholes to verify the adequacy of sump pumps, level alarm circuits, cable splices subject to submergence, and drainage for bunkers/manholes; verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and walked down the four areas listed below to verify the adequacy of equipment seals located below the flood line, floor and wall penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, and control circuits, and temporary or removable flood barriers. Specific documents reviewed during this inspection are listed in the attachment.

- October 5, 2010, Unit 1, Circulating Water Pump 1-1 and 1-2 control cabling underground vaults (BPZ40/40A and BPZ41/41A)
- October 7, 2010, Unit 1, 4kV Bus H control cabling vault (BPO2)
- October 12 and 13, 2010, Unit 1, Auxiliary Saltwater Pump 1-1 and 1-2 control cabling underground vaults (BPZ42/42A and BPZ43/43A) and Auxiliary Saltwater System cross-tie valve circuitry vault (BPZ44/44A)
- December 8, 2010, Diesel Generator Fuel Oil Transfer Pit 1-1

These activities constitute completion of one flood protection measures inspection samples and one bunker/manhole sample as defined by Inspection Procedure 71111.06-05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the Units 1 and 2 containment fan cooler units, the ultimate heat sink including the auxiliary salt water system, and the component cooling water heat exchangers. The inspectors verified that performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines," the licensee properly utilized biofouling controls; the licensee’s heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchangers were correctly categorized under 10 CFR 50.65, “Requirements for

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of three heat sink inspection samples as defined in Inspection Procedure 71111.07-05.

b. Findings

Corrosion of the Containment Fan Cooler Unit Cooling Coil Casings

Introduction. The inspectors identified an unresolved item concerning the degradation of the containment fan cooler unit cooling coil casings due to corrosion. Specifically, the issue concerns the licensee's actions to verify the heat removal capability of the containment fan coolers under degraded conditions and the failure to take corrective actions for the repair or replacement of the corroded cooling coil assemblies.

Description. The containment fan cooler units function during normal plant operation to maintain the containment atmosphere at design conditions. During accident conditions, the cooler units automatically initiate to maintain containment operability. Diablo Canyon Units 1 and 2 each have five cooler units installed inside the containment building. Each cooler unit has two cooling coil banks with six coils stacked in each bank. Each of the coils is mounted on sheet metal casings and the casings are mounted within the cooler unit frame. The casings act to prevent air bypass between the coils in the banks and as structural support for the coil tubes and fins.

The inspectors reviewed Diablo Canyon Power Plant Health Issue Reports 2002-S023-002 and 2002-S023-003 which identified corrosion of the containment fan cooler unit cooling coil casings. The power plant health issue reports acknowledged that continued casing corrosion would decrease the available design margin of the heat removal capacity of the cooler unit cooling coils. Containment Fan Cooler Unit Coil Study, Phase 1, Revision 0, recommended that the corroded coil casing be repaired or replaced whenever possible to avoid impacting the cooler unit heat removal capacity. The licensee's initial recommended corrective action, as described in the power plant health issue reports, was to replace the cooling coil assemblies in Units 1 and 2 beginning in Refueling Outage 1R13 (Fall 2005). The replacement plans were not implemented and the licensee currently plans to begin replacement of the cooling coil assemblies during upcoming Refueling Outages 1R18 and 2R18. All cooling coil assemblies in Units 1 and 2 are scheduled to be replaced by Refueling Outage 1R20 and 2R20, respectively.

The inspectors determined that additional information was needed to resolve this issue. The inspectors were unable to clearly determine the design basis function of the cooling coil casings based upon documentation provided by the licensee during the inspection. Additionally, the licensee has not quantified the effect of the corrosion to verify that the cooling coil casing functions would be maintained under the current degraded conditions and has not provided a technical justification for the acceptability of the proposed coil assembly replacement schedule. The licensee has agreed to provide this information for NRC's review.

Because more information is necessary to resolve this issue, it is considered an unresolved item pending further NRC's review. The NRC will review the licensee's evaluation to determine:

- If the licensee’s failure to verify the heat removal capability of the containment fan cooler units is a performance deficiency

- If the licensee’s decision to delay taking corrective actions for repair or replacement of the corroded cooling coils constitutes a violation of NRC requirements

The licensee entered this issue into its corrective action program as Notification 50364991. This unresolved item is identified as URI 05000275; 323/2010005-03, “Corrosion of Containment Fan Cooler Unit Cooling Coil Casings.”

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control (71111.08-02.01)

a. Inspection Scope

The inspectors observed 17 nondestructive examination activities and reviewed three nondestructive examination activities that included six types of examinations. The inspectors also reviewed four examinations with relevant indications that were identified in the previous outage for Unit 1 that had been accepted by licensee personnel for continued service. The licensee did not identify any relevant indications accepted for continued service during the nondestructive examinations performed during the current Refueling Outage 1R16.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Steam Generators	Magnetic examination test of pipe support lugs attached to the feedwater pipe Line 1-K16-556-16 IV Hanger # 1044-7V	Magnetic Examination Test - Dry Particle
Reactor Coolant System	Loop 1 cold leg RTD connection Line # S6-1140-2SPL	Penetrant Examination Test - Contrasting Dye
Reactor Coolant System	Reactor coolant pump 1-2 weld attachment - support lug Lug Weld # 3	Penetrant Examination Test - Contrasting Dye
Component Cooling Water System	Post-freeze seal liquid penetrant exam. Component ID: CCW-1-RV-47	Penetrant Examination Test - Contrasting Dye

Chemical Volume and Control System	Valve CVCS-1-8402-A	Radiograph Examination Test - Digital Process
Chemical Volume and Control System	Valve CVCS 1-8403	Radiograph Examination Test - Digital Process
Reactor Coolant System	Reactor cold leg, nozzle to safe-end dissimilar metal weld Weld # WIB-RC-3-18 SE	Ultrasonic Examination Test - Phased Array
Reactor Coolant System	Reactor cold leg, nozzle to safe-end dissimilar metal weld Weld # WIB-RC-2-20 (SE)	Ultrasonic Examination Test - Phased Array
Reactor Coolant System	Steam generator 1-1 circumferential weld, FW-11.07.01	Ultrasonic Examination Test
Steam Generator	Steam generator feedwater nozzle to shell inner radius inspection, FWN-IR	Ultrasonic Examination Test
Reactor Coolant System	Steam generator 1-1 manway bolting and seal leakage	Visual Examination Test - VT-2
Reactor Coolant System	Steam generator 1-2 manway bolting and seal leakage	Visual Examination Test - VT-2
Residual Heat Removal System	Hanger support for residual heat removal, safety injection system to reactor coolant system loop 2 hot leg, Line # 1-S6-2576-8 A Drawing # 049308	Visual Examination Test - VT-3
Reactor Coolant System	Pressurizer auxiliary spray pipe support, hanger # 0181-29	Visual Examination Test - VT-3
Containment	Containment - concrete examination of Unit 1	Visual Examination Test - VT-3 C
Steam Generators	Supports and hangers for the main steam piping outlet from steam generator 1-2, Line 1-145-227-28V, Hanger # 1022-IV	Visual Examination Test - VT-3
Steam Generators	Supports and hangers for the main feedwater piping to steam	Visual Examination Test - VT-3

generator 1-4
Line 1-K16-556-16 IV
Hanger # 1044-7V

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	IDENTIFICATION	EXAMINATION TYPE
Component Cooling Water System	Pre-freeze seal liquid penetrant exam Component ID# CCW-1-RV-47	Penetrant Examination Test - Contrasting Dye
Residual Heat Removal System	Surface examination of studs and nuts for valve RHR-1-8740A	Visual Examination Test - VT-1
Residual Heat Removal System	Visual examination of valve internals for valve RHR-1-8740A	Visual Examination Test - VT-3

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with the ASME Code requirements and applicable procedures. The inspectors compared any indications identified during previous examinations and verified that licensee personnel dispositioned the indications in accordance with the ASME Code and approved procedures. The inspectors also verified the qualifications of all nondestructive examination technicians performing the inspections were current.

The inspectors observed two welds and reviewed three welds on the reactor coolant system pressure boundary.

The inspectors directly observed a portion of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
High Pressure Safety Injection	Valve CVCS-1-8402-A	Gas Tungsten Arc Welding
High Pressure Safety Injection	Valve CVCS-1-8403	Gas Tungsten Arc Welding

The inspectors reviewed the records of the following welding activities:

SYSTEM	WELD IDENTIFICATION	WELD TYPE
Residual Heat Removal System	Component ID - RHR-1-930 Socket weld - install ¾ inch valve.	Gas Tungsten Arc Welding
Safety Injection System	SI-1-161 valve	Gas Tungsten Arc Welding
Safety Injection System	SI-1-8917C valve	Gas Tungsten Arc Welding
Component Cooling Water	Replacement of component cooling water valve CCW-1-698	Gas Tungsten Arc Welding

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of requirements as defined in Inspection Procedure 71111.08-02.01.

b. Findings

No findings were identified.

.2 Vessel Upper Head Penetration Inspection Activities (71111.08-02.02)

a. Inspection Scope

The licensee replaced the reactor pressure vessel head during this Refueling Outage 1R16 for Unit 1. All related nondestructive testing inspection is documented in this report in Section 4OA5, Other Activities, under "Reactor Vessel Head Replacement Inspection (71007)."

These actions constitute completion of requirements as defined in Inspection Procedure 71111.08-02.02.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control Inspection Activities (71111.08-02.03)

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure STP R-8C, "Containment Walkdown for Evidence of Boric Acid Leakage," Revision 9, and ER1.ID2, "Boric Acid Corrosion Control Program," Revision 2. The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors reviewed 11 engineering evaluations for those components where boric acid was identified to ensure that the ASME Code wall thickness limits were properly maintained. The evaluations were reviewed for the causes and corrective actions. The inspectors also reviewed 19 notifications to confirm that the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code. Specific documents reviewed during this inspection are listed in the attachment.

These actions constitute completion of requirements as defined in Inspection Procedure 71111.08-02.03.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities (71111.08-02.04)

a. Inspection Scope

This was the first inspection performed after replacing the steam generators per the EPRI guidelines and it consisted of a 100 percent inspection of all tubes.

The inspectors assessed the in-situ screening criteria to assure consistency between assumed nondestructive examination flaw sizing accuracy and data from the EPRI examination technique specification sheets. At the time of this inspection, no conditions had been identified that warranted in-situ pressure testing. The inspectors reviewed the licensee's report for Unit 1 "Steam Generator Degradation Assessment," dated October 13, 2010. This review determined that the remaining screening parameters were consistent with the EPRI guidelines.

In addition, the inspectors reviewed both the licensee site-validated and qualified acquisition and analysis technique sheets used during this refueling outage and the qualifying EPRI examination technique specification sheets to verify that the essential variables regarding flaw sizing accuracy, tubing, equipment, technique, and analysis had been identified and qualified through demonstration. The inspectors reviewed acquisition technique and analysis technique sheets.

Technical Specification 5.5.9.d requires inspection of 100 percent of the tubes in each steam generator during the first refueling outage following steam generator replacement.

In Refueling Outage 1R16, the first refueling outage following steam generator replacement, 100 percent of the tubes in each steam generator were inspected full length by bobbin coil probe. Following the bobbin coil probe inspection, plus point probe inspections were conducted on the following:

- 100 percent of bobbin coil reported "I" codes, which included DNI (dent/ding with possible indication), ADI (absolute drift indication), DSI (distorted support indication), distorted tube sheet indication (DTI), and NQI (non-quantifiable indication)
- 100 percent of greater than or equal to 1 volt dents
- 92 percent of greater than or equal to 1 volt dings
- 100 percent of bobbin coil reported proximity indications
- 100 percent of bobbin coil reported potential loose part indications
- 100 percent of bobbin coil reported manufacturing burnish mark indications
- 100 percent of percent through-wall indications reported by bobbin coil
- Three tubesheet locations where permeability variation signals were reported in preservice inspection
- One tubesheet bulge that was reported in the preservice inspection

Also included was a secondary side visual inspection that included:

- Pre-lance visual inspection
- Post- lance visual inspection
    i. 100 percent of trough region
    ii. 100 percent of the outer periphery tubes
    iii. Center 10 columns of the hot leg top of tube sheet region and columns 20, 40, 80, and 100 in both legs

A new damage mechanism, anti-vibration bar wear, was identified during this inspection. At the time of the inspection, a preliminary assessment of the new degradation mechanism was provided and reviewed by the inspectors. The inspectors verified that the licensee fully enveloped the degradation mechanism in its analysis of extended conditions including operational concerns.

The inspectors evaluated the recommended steam generator tube eddy current test scope established by technical specification requirements and the licensee's degradation assessment report. The inspectors compared the recommended test scope to the actual test scope and found that the licensee had accounted for all known flaws and had, as a minimum, established a test scope that met technical specification requirements, EPRI guidelines, and commitments made to the NRC.

The inspectors also confirmed the following:

- All known areas of potential degradation were inspected

- No repair processes were used or needed

- Steam generator leakage was not greater than three gallons per day

- One loose part was identified during eddy current testing and that loose part was subsequently removed during sludge lancing

- All eddy current results were acceptable and there were no questionable results

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

These actions constitute completion of requirements as defined in Inspection Procedure 71111.08-02.04.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems (71111.08-02.05)

a. Inspection scope

The inspectors reviewed 22 notifications which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues to be appropriate. The specific condition reports reviewed are listed in the documents reviewed section. From this review the inspectors concluded that the licensee has an appropriate threshold for entering inservice inspection issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry inservice inspection operating experience.

These actions constitute completion of requirements as defined in Inspection Procedure 71111.08-02.05.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

The licensed operator requalification program involves two training cycles that are conducted over a two-year period. In the first cycle, the annual cycle, the operators are administered an operating test consisting of job performance measures and simulator scenarios. In the second part of the training cycle, the biennial cycle, operators are administered an operating test and a comprehensive written examination.

.1 Annual Inspection (71111.11B)

a. Inspection Scope

The inspectors conducted an in-office review of the annual requalification training program operating test results for 2010. The licensee examined 89 operators (39 reactor operators and 50 senior reactor operators) during this requalification cycle. In addition, 16 operating crews were examined on the facility's simulator. 16 of the operating crews passed the simulator scenarios and 88 operators passed the operating tests.

One senior reactor operator failed the operating test, and was remediated and re-examined prior to returning to licensed duties.

b. Findings

No findings were identified.

.2 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On November 16, 2010, the inspectors observed a crew of licensed operators in the plant'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 one quarterly licensed-operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Intake structure/auxiliary feedwater, Notification 50033853
- Feedwater heating, Notification 50274626
- Plant process computer, Notification 50340508

The inspectors reviewed events 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 three quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings 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:

- Technical Specification Tracking Sheet 1-TS-10-0701, Vital Battery 1-1 Cell-14 less than required voltage, September 30, 2010

- Technical Specification Tracking Sheets 2-TS-10-0650 and 2-TS-10-0651, 230 Preferred offsite power outages, October 15 and 23, 2010

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 two maintenance risk assessments and emergent work control inspection samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Unit 2, Residual heat removal containment sump margin, October 10, 2010, Notification 50355265

- Unit 2 Insulation in Bio-Wall Penetration, October 14, 2010, Notification 50350918

- Unit 1, Auxiliary Feedwater Pump 1-2 , Discharge Level Control Valve control problems, November 5, 2010, Notification 50358757

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 FSARU to the licensee personnel'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 three operability evaluations inspection samples as defined in Inspection Procedure 71111.15-04

b. Findings

Inadequate Operability Determinations

Introduction. The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to adequately evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, "Operability Determination."

Description. The inspectors identified two examples of less than adequate operability evaluations of nonconforming technical specification required equipment. On October 15, 2010, the inspectors identified a less than adequate technical evaluation supporting a Prompt Operability Assessment 50350918, "Unit 2 – Insulation in Bio-Wall Penetration." Pacific Gas and Electric provided justification for continued operation of Unit 2 after identifying approximately 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous insulation in the Unit 1 reactor coolant loop biological shield wall penetrations. The licensee had not accounted for this additional fibrous material in the containment sump screen design. This additional material was a concern because the post accident environment could reduce the insulation to fibrous debris affecting the capability of the emergency core cooling system pumps to take suction from the containment sumps. Plant engineers initially concluded that the containment sumps were operable because the additional amount of fibrous material was within the bounds of the sump screen test, "Upscale Plant Debris Loading and Maximum Test Head Loss Test, 14-S-PSG." However, the licensee only considered fibrous material from one of the eight penetrations, or about 13 percent of the total material in the prompt operability assessment. The licensee excluded material from the other seven penetrations based on the assumption that the reactor cavity would not pressurize following a rupture of the reactor coolant system piping. The licensee's assumption was based on an inappropriate application of the leak-before-break methodology. The methodology stated that dynamic sub compartment pressurization no longer needed to be included within the plant design basis. The inspectors concluded that plant engineers failed to include the limitations of the leak-before-break methodology, as described in the current licensing basis, before using the methodology as a basis to exclude debris generation. FSARU Section 3.6.2, “Design Basis Piping Break Criteria,” stated that leak-before-break methodology only applied to the dynamic structural effects of main reactor coolant loop piping (pipe whip, pipe break reaction forces and dynamic effects associated with postulated pipe ruptures missiles, pipe whipping). The licensee estimated between ten and twenty pounds per square inch (psi) differential pressure would be required to dislodge the fibrous insulation from all eight penetrations. The inspectors identified that FSARU Table 6.2-24, “Containment Pressure Differential Compartment Pressures,” stated that the calculated differential pressure in the lower reactor cavity would be 51 psi. The inspectors concluded that the actual differential pressure would likely be greater due to the blocked vent path (discussed in Section IR18 of this report). FSARU Table 6.2-22, “Containment Pressure Differential Elements for Pipe Annulus Analysis Model,” stated that the peak differential pressure was calculated based on an assumed 48 square foot vent area. The inspectors identified that 35 square foot of this vent area was blocked by a shield plug. The failure of the licensee to evaluate the extent of condition on the capability of the component to perform its specified safety function, and fully consider the current licensing bases in Prompt Operability Assessment 50350918 was a violation of Procedure OM7.ID12, Section 5.3, “Write the POA.”

The inspectors identified a second example of a less than adequate prompt operability assessment. On October 23, 2010, Pacific Gas and Electric completed Prompt Operability Assessment Notification 50355265, “Residual Heat Removal Sump Margin,” following discovery of the blocked post accident flow path from the Unit 1 reactor cavity to the containment sump. The flow path assumed in the accident analysis was blocked by a large steel plug. The blocked flow path resulted in a reduction of refueling water storage tank inventory available for recirculation mode of emergency core cooling. Plant engineers concluded that the containment sump was operable even though the blocked flow path resulted in inadequate inventory to fully submerge the sump for the large break loss of coolant case. The inspectors concluded that the prompt operability assessment was inadequate because the licensee failed to identify the specified safety function of the refueling water storage tank. NRC Safety Evaluation Report “Licensee Amendment Number 199, Diablo Canyon Power Plant – Issuance of Amendments RE: Technical Specification 3.5.4,” March 26, 2008, stated that the basis for the approval of Technical Specification 3.5.4 included that the refueling water storage tank had adequate inventory to ensure that the new sump screens would be fully submerged at the initiation of recirculation for the large break loss of coolant accident. The specified safety function of the refueling water storage tank was to ensure adequate inventory for full sump screen submergence. The failure of the licensee to identify and demonstrate how the refueling water storage tank specified safety function was met in Prompt Operability Assessment 50355265 was a violation of Procedure OM7.ID12, Section 5.3, “Write the POA.”

Analysis. The inspectors concluded that the failure of licensee personnel to perform an adequate technical evaluation of non-conforming plant equipment, in accordance with plant procedures, was a performance deficiency. This performance deficiency is more than minor because the finding affected 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. This finding was similar to examples 3.i and 3.j of Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," The inspectors used Inspection Manual Chapter 609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," to analyze the finding because the performance deficiency involved a design or qualification deficiency. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because Pacific Gas and Electric did not use conservative assumptions in decisions to demonstrate component operability in either example [H.1(b)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," required that activities affecting quality be accomplished in accordance with instructions or procedures. Quality Procedure OM7.ID12, Section 5.3.3, required prompt operability assessments to identify the "specified safety function" of affected components to a determination of the extent of the condition on the capability of the component to perform the specified safety function. Contrary to the above, on October 23, 2010, licensee personnel failed to identify the specified safety function of the refueling water storage tank in Prompt Operability Assessment 5035929, "Residual Heat Removal Sump Margin," and on October 31, 2010, failed to determine the extent of the condition affecting additional fibrous material in Prompt Operability Assessment 50350918, "Unit 2 – Insulation in Bio-Wall Penetration." Because this finding was of very low safety significance and was entered into the corrective action program as Notifications 50369117, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275; 323/2010005-04 "Inadequate Operability Determinations."

1R18 Plant Modifications (71111.18)

Permanent Modifications

a. Inspection Scope

The inspectors reviewed key affected parameters associated with energy needs, materials, replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flow paths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the permanent modifications listed below.

- November 8, 2010, Unit 1, Containment recirculation sump flow path modifications
- December 31, 2010, Unit 1 Replacement reactor head

The inspectors verified that modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; post modification testing will maintain the plant in a safe configuration during testing by verifying that unintended system interactions will not occur; systems, structures and components' performance characteristics still meet the design basis; the modification design assumptions were appropriate; the modification test acceptance criteria will be met; and licensee personnel identified and implemented appropriate corrective actions associated with permanent plant modifications. Specific documents reviewed during this inspection are listed in the attachment.

These activities constitute completion of two samples for permanent plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

Less than Adequate Containment Recirculation Sump Design Control

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric engineering personnel failed to properly determine inventory unavailable for cold leg recirculation emergency core cooling in Calculation STA-255, "Minimum Required Refueling Water Storage Tank Level for GE Sumps," Revision 2, dated September 28, 2007. The inspectors identified that the assumed post accident recirculation path from the reactor cavity to the containment sump was blocked by a large steel shield plug.

Description. On October 19, 2010, the inspectors identified that the Unit 1 emergency core cooling post accident flow path from the reactor cavity to the containment sumps was blocked by a large steel plug. The accident analysis assumed this 35 square foot path was open to allow coolant from a pipe break inside the biological shield to communicate with containment sumps during the recirculation mode of emergency core cooling. The licensee credited the inventory from the reactor cavity when determining the minimum required refueling water storage tank volume in Calculation STA-255. Pacific Gas and Electric used Calculation STA-255 as a basis for determining the minimum required refueling water storage tank volume for Technical Specification 3.5.4, "Refueling Water Storage Tank," as discussed in "Licensee Amendment Number 199, Diablo Canyon Power Plant – Issuance of Amendments RE: Technical Specification 3.5.4," March 26, 2008. The inspectors concluded that Calculation STA-255 was inadequate because the licensee failed to account for the inventory unavailable for emergency core cooling recirculation due to the blocked flow path. The inspectors identified that the recirculation flow path was also blocked on Unit 2. The shield plugs were installed prior to initial plant licensing. However, the licensee had not updated plant drawings to reflect that the shield plugs were installed. The inspectors concluded that the most significant contributor to the violation was inaccurate plant drawings used for determining containment post accident flow paths and inventory hold up locations used in Calculation STA-255. The licensee's corrective actions included the completion of a prompt operability assessment justifying continued operation of Unit 2 and replacement of the shield plug with a movable platform on Unit 1 during the refueling outage.

Analysis. The inspectors concluded that the failure of engineering personnel to include the unavailable inventory due to the blocked flow path in Calculation STA-255 was a performance deficiency. This finding was more than minor because the performance deficiency affected the Mitigating Systems Cornerstone plant modification design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors used Inspection Manual Chapter 609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," to analyze the finding. The inspectors concluded that the finding was of very low safety significance (Green) because the finding involved a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the resources component because Pacific Gas and Electric failed to use complete, accurate and up-to-date design documentation for Calculation STA 255 [H.2(c)].

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, "Design Control," required measures be established to assure that applicable regulatory requirements and the design basis be correctly translated into specifications. Contrary to the above, on September 28, 2007, Pacific Gas and Electric failed to implement adequate measures to assure that applicable regulatory requirements and the design basis of the refueling water storage tank was correctly translated into specifications. Specifically, Calculation STA-255, "Minimum Required Refueling Water Storage Tank Level for GE Sumps," Revision 2, failed to account for post-accident inventory unavailable for emergency core cooling recirculation flow due to a blocked flow path from the cavity area to the containment sumps. Because this finding was of very low safety significance and was entered into the corrective action program as Notification 5035265, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275; 323/2010005-05 "Less than Adequate Containment Recirculation Sump Design Control."

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:

- Unit 1, Emergency diesel generator 1-2 maintenance outage, Order 64025252, September 30, 2010
- Unit 1, Replacement reactor head digital rod indication system data cabinet cable verification test, Order 68008462, October 30, 2010
- Unit 1, Replacement reactor head control rod drive mechanism test, Order 68008483-0010, October 30, 2010
- Unit 1, Replacement reactor head control rod drive mechanism coil stack operational test, Order 68007794, October 30, 2010
- Unit 1, Replacement reactor head thermocouple cross calibration test, Order 64026499, November 6, 2010
- Unit 1, Replacement reactor head control rod coil verification test, Order 64027300, November 7, 2010
- Unit 1, Replacement reactor head digital metal impact monitoring system functional test, Order 68008242, November 7, 2010

- Unit 1 Reactor head replacement project report of Section XI system pressure test, Order 64024880, November 8, 2010

- Unit 1, Replacement reactor head rod drop test, Order 64025376, November 8, 2010

- Unit 1, Auxiliary Feedwater System Level Control Valve LCV-111 repairs, Order 60030601, November 13, 2010

- Unit 1, Replacement reactor head control rod system test, Orders 64026734 and 64026817, November 17, 2010

- Unit 1 Reactor head replacement project reactor vessel level indication system test, Order 64025082, November 18, 2010

- Unit 1 Reactor head replacement project testing, (Orders 68007794 and 68009187), November 21 2010

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 (as applicable):

- 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 FSARU, 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 13 postmaintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

Inadequate Emergency Diesel Generator Surveillance Testing

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," after Pacific Gas and Electric failed to incorporate the requirements and acceptance limits contained in applicable design documents into the emergency diesel generator test procedures.

Description. In December 2008, the inspectors identified that the emergency diesel generator loading calculations were inadequate to demonstrate that the system design bases were met. The inspectors dispositioned this issue as noncited violation 05000275/2008005-04; 323/2008005-04, “Inadequate Design Control for the Emergency Diesel Generator.” On January 9, 2009, the licensee entered this condition into the corrective action program as Notifications 50163396 and 5017902. Plant engineers revised Design Calculation 9000037760-21 to update the diesel generator accident loading analysis and concluded the limiting loading cases were Bus F at 2710 KW, Bus G at 2660 KW, and Bus H at 2602 KW. In February 2009, the NRC component design bases inspection determined that Pacific Gas and Electric had performed an inadequate 10 CFR 50.59 evaluation and inappropriately adopted Regulatory Guide 1.9, Revision 3 as the basis for the emergency diesel generator testing. The licensee was committed to Safety Guide 9, “Selection of Diesel Generator Set Capacity for Standby Power Supplies,” and Regulatory Guide 1.108, “Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants,” Revision 1. On March 9, 2009, the licensee concluded that Technical Specification Surveillance Requirements 3.8.1, “AC Sources – Operating,” was nonconservative (Notification 50207912). On April 9, 2009, Pacific Gas and Electric concluded that Technical Specification Surveillance Requirement 3.8.1 was not adequate to preserve safety and applied the provisions of Technical Specification Surveillance Requirement 3.0.3, and Administrative Letter 98-10, “Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety.” However, the licensee did not update the test parameters in Surveillance Test Procedure STP M-9D1, “Diesel Generator Full Load Rejection Test,” to match the values determined in Calculation 9000037760-21. During the review of Surveillance Test Procedure STP M-9D1, “Diesel Generator Full Load Rejection Test,” the inspectors determined the licensee failed to incorporate the revised loading values identified in Calculation 9000037760-21 into the test procedure. Regulatory Guide 1.108, Revision 1, Section C.2.a(4) stated in part, “Demonstrate proper operation during diesel generator load shedding, including a test of the largest single load and of complete loss of load and verify that the voltage requirements are met and that the overspeed limits are not exceeded. Step 12.4.2 of STP M-9D1 stated that the load reject test shall be performed at 2600 KW. The inspectors determined that STP M-9D1 did not test a complete loss of load from the diesel generator sets based on worst case load calculations. The inspectors concluded the most significant contributor to the finding was less than adequate diesel generator loading evaluations to support corrective action from previous violations associated with the emergency diesel generator testing.

Analysis. The inspectors determined that the licensee’s failure to properly translate the requirements contained in applicable design documents for the emergency diesel generators into the testing program was a performance deficiency. The finding was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of the emergency diesel generators to respond to initiating events to prevent undesirable consequences. The inspectors determined the significance of the finding using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative surveillance test such that the resolution addressed the fundamental basis for the surveillance [P.1(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," required, in part, that a test program shall be established to ensure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service in accordance with written test procedures which incorporate the requirements and acceptable limits contained in applicable design documents. Contrary to the above, prior to January 4, 2011, Pacific Gas and Electric failed to incorporate the requirements and acceptance limits contained in applicable design documents into the emergency diesel generator test procedures. Specifically, Surveillance Test Procedure STP M-9D1, "Diesel Generator Full Load Rejection Test," did not require testing of the emergency diesel generators at the full load values determined in Calculation 9000037760-21. Because this violation was of very low safety significance (Green) and had been entered into the corrective action program as Notification 50368801, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275; 323/2010005-06, "Inadequate Emergency Diesel Generator Surveillance Testing."

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the outage safety plan and contingency plans for Unit 1 refueling outage conducted on October 2, 2010, to confirm that licensee personnel had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense in depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below.

- Configuration management, including maintenance of defense in depth, was commensurate with the outage safety plan for key safety functions and compliance with the applicable technical specifications when taking equipment out of service.

- Clearance activities, including confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.

- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.

- Status and configuration of electrical systems to ensure that technical specifications and outage safety-plan requirements were met, and controls over switchyard activities.

- Monitoring of decay heat removal processes, systems, and components.

- Verification that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.

- Reactor water inventory controls, including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.

- Controls over activities that could affect reactivity.

- Maintenance of secondary containment as required by the technical specifications.

- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.

- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.

- Licensee identification and resolution of problems related to refueling outage activities.

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

These activities constitute completion of one refueling outage and other outage inspection sample as defined in Inspection Procedure 71111.20-05.

b. <u>Findings</u>

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. <u>Inspection Scope</u>

The inspectors reviewed the FSARU, procedure requirements, and technical specifications to ensure that the 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.

- September 28, 2010, Unit 1, Inservice test of component cooling water pump-12
- October 26, 2010, Unit 1, Penetration 54 containment isolation valve leak testing
- October 27, 2010, Unit 1, Penetration 21 containment isolation valve leak testing
- November 7, 2010, Unit 1, Reactor coolant leakage test
- November 14, 2010, Unit 1, Inservice test of turbine-driven auxiliary feedwater pump 1-1

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

These activities constitute completion of five surveillance testing inspection samples as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on December 15, 2010, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the Technical Support Center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package.

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

b. Findings

No findings were identified.

.2 Training Observations

a. Inspection Scope

The inspectors observed a simulator training evolution for licensed operators on December 7, 2010, 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 were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

This area was inspected to: (1) review and assess licensee's performance in assessing the radiological hazards in the workplace associated with licensed activities and the implementation of appropriate radiation monitoring and exposure control measures for both individual and collective exposures, (2) verify the licensee was properly identifying and reporting Occupational Radiation Safety Cornerstone performance indicators, and (3) identify those performance deficiencies that were reportable as a performance indicator and which may have represented a substantial potential for overexposure of the worker.

The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspectors also reviewed activities associated with the reactor head replacement to fulfill the inspection requirements of Inspection Procedure 71007, "Reactor Vessel head Replacement inspection." During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed walkdowns of various portions of the plant, performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation reported by the licensee in the Occupational Radiation Safety Cornerstone

- The hazard assessment program, including a review of the licensee's evaluations of changes in plant operations and radiological surveys to detect dose rates, airborne radioactivity, and surface contamination levels

- Instructions and notices to workers, including labeling or marking containers of radioactive material, radiation work permits, actions for electronic dosimeter alarms, and changes to radiological conditions

- Programs and processes for control of sealed sources and release of potentially contaminated material from the radiologically controlled area, including survey performance, instrument sensitivity, release criteria, procedural guidance, and sealed source accountability

- Radiological hazards control and work coverage, including the adequacy of surveys, radiation protection job coverage, and contamination controls; the use of electronic dosimeters in high noise areas; dosimetry placement; airborne radioactivity monitoring; controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools; and posting and physical controls for high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

- Audits, self-assessments, and corrective action documents related to radiological hazard assessment and exposure controls since the last inspection

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

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.01-05.

b. Findings

No findings were identified.

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

This area was inspected to assess performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspectors also reviewed activities associated with the reactor pressure vessel head replacement to fulfill the inspection requirements of Inspection Procedure 71007, "Reactor Vessel head Replacement inspection." During the inspection, the inspectors interviewed licensee personnel and reviewed the following items:

- Site-specific ALARA procedures and collective exposure history, including the current 3-year rolling average, site-specific trends in collective exposures, and source-term measurements

- ALARA work activity evaluations/post job reviews, exposure estimates, and exposure mitigation requirements

- The methodology for estimating work activity exposures, the intended dose outcome, the accuracy of dose rate and man-hour estimates, and intended versus actual work activity doses and the reasons for any inconsistencies

- Records detailing the historical trends and current status of tracked plant source terms and contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Audits, self-assessments, and corrective action documents related to ALARA planning and controls since the last inspection

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

These activities constitute completion of the one required sample as defined in Inspection Procedure 71124.02-05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the third Quarter 2010 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 were identified.

.2 Reactor Coolant System Specific Activity (BI01)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity performance indicator for Units 1 and 2 for the period from the fourth quarter 2009 through the third quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's reactor coolant system chemistry samples, technical specification requirements, issue reports, event reports, and NRC integrated inspection reports for the period of October, 2009 through September 2010 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 for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two reactor coolant system specific activity samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Reactor Coolant System Leakage (BI02)

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system leakage performance indicator for Unit 1 and Unit 2 for the period from the fourth quarter 2009 through the third quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator logs, reactor coolant system leakage tracking data, issue reports, event reports, and NRC integrated inspection reports for the period of October, 2009 through September 2010 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 for this indicator and none were identified. Specific documents reviewed are described in the attachment to this report.

These activities constitute completion of two reactor coolant system leakage samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.4 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the fourth quarter 2009 through the third quarter 2010. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed corrective action program records associated with high radiation area (greater than 1 rem/hr) and very high radiation area non-conformances. The inspectors reviewed radiological, controlled area exit transactions greater than 100 mrem. The inspectors also conducted walkdowns of high radiation areas (greater than 1 rem/hr) and very high radiation area entrances to determine the adequacy of the controls of these areas.

These activities constitute completion of the occupational exposure control effectiveness sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.5 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences (PR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the fourth quarter 2009 through the third quarter 2010. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

The inspectors reviewed the licensee's corrective action program records and selected individual annual or special reports to identify potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose.

These activities constitute completion of the radiological effluent technical specifications/offsite dose calculation manual radiological effluent occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings 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 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 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 July 2010 through December 2010 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/challenges 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.

These activities constitute completion of a single semi-annual trend inspection sample as defined in Inspection Procedure 71152-05.

b. <u>Findings and Observations</u>

<u>Continuation of an Adverse Trend in Problem Evaluation</u>

The inspectors concluded that an adverse trend associated with the thoroughness of Pacific Gas and Electric's problem evaluation continued through December 2010. The inspectors first identified this adverse trend in September 2008 (described in Section 4OA2 of Inspection Report 05000275;323/2008005). The NRC subsequently identified a substantive crosscutting issue associated with the licensee's problem evaluation thoroughness in the 2009 annual assessment. Current examples of this adverse trend included:

- July 2010, two examples of less than adequate problem evaluation of nonconforming conditions affecting equipment operability (NCV 05000275; 323/201004-04, Inadequate Operability Determination).

- September 2010, five examples of less than adequate problem evaluation resulting in a failure to submit complete and accurate information to the NRC. This example illustrated less than adequate application and understanding of the current licensing basis (NRC 05000275; 05000323/2010006-03, Failure to Submit Complete and Accurate Information for a Requested License Amendment).

- September 2010, one example of a failure to appropriately evaluate and correct a residual heat removal system testing procedure (NCV 05000323/2010006-05, Failure to Appropriately Evaluate Failed Residual Heat Removal Surveillance Test).

- December 2010, one example of the failure to adequately evaluate long standing problems with emergency diesel generator testing as described in Section 1R19 of this report. This example illustrated less than adequate application and understanding of the current licensing basis (NCV 05000275; 323/2010005-06, Inadequate Emergency Diesel Generator Surveillance Testing).

In May 2010, Pacific Gas and Electric completed a second root cause of this adverse trend (Notification Order 60024480, "Adverse Trend in Thoroughness of Problem Evaluation,"). The licensee concluded that the leadership team has not provided adequate standards, effectively demonstrated or reinforced behaviors, or established sustainable programs in the area of evaluation. The root cause team recommended the following corrective actions:

- Provide expectations to the senior leadership team on coaching standards and responsibility for implementing an effective evaluation program

- Establish generic governance for evaluation programs

- Train program sponsors and program owners on the structure of an effective program governance

- Program implementation to ensure evaluation programs incorporate the essential elements for their sustainability

The inspectors will continue to monitor the licensee's progress to address this adverse trend.

<u>Adverse Trend Related in Human Performance Conservative Assumptions in Decision Making</u>

The inspectors concluded that the adverse trend associated with the use of conservative assumptions in decision making continued through December 2010. The inspectors first discussed examples of nonconservative decision making in February 2010 (described in Section 4OA2 of Inspection Report 05000275;323/2009005**).** The NRC subsequently identified an adverse theme related to failure to use conservative assumptions in decision making in the 2010 Mid-Cycle Performance Review completed in September 2010. Current examples of this adverse trend included:

- September 2010, nonconservative decision making resulted in the failure to maintain adequate design control associated with the emergency diesel generating air system design control measures. This example illustrated less than adequate application and understanding of the current licensing basis
- (NRC 05000275/2010006-01; 05000323/2010006-01, Inadequate Design Control for the Emergency Diesel Generator).

- September 2010, nonconservative decision making resulted in the failure to ensure that operators were able to implement specified actions in response to operational events and accidents. This example illustrated a less than adequate understanding of the current licensing basis requirements (NCV 5000275/2010006-02; 05000323/2010006-02, Failure to Maintain Proficiency of Operators to Meet the Time Critical Operator Actions).

- September 2010, two examples of nonconservative decision making resulted in the failure to promptly identify and correct nonconforming conditions related to the emergency diesel generators meeting the design basis. This example included elements of a less than adequate understanding of the current licensing basis (NCV 05000275/2010006-04; 05000323/2010006-04, Untimely and Inadequate Corrective Actions for the Emergency Diesel Generators).

- October 2010, two examples of nonconservative decision making used in prompt operability determinations, as discussed in Section 1R15 of this report. This example illustrated a less than adequate application and understanding of the current licensing basis (NCV 05000275; 323/2010005-04 Inadequate Operability Determinations).

In October, Pacific Gas and Electric completed a common cause analysis of the adverse trend (Notification Order 60026627, Common Cause Analysis: Nonconservative Assumptions). The licensee concluded the following apparent causes contributed to the adverse trend:

- Depth and/or breadth of evaluations were limited due to a lack of follow-through

- Inattentiveness to current licensing bases requirements
- Less than adequate change management

Inspectors Assessment of Licensee's Actions

The inspectors performed an in-depth review of both adverse trends including the licensee's root causes and corrective actions. The inspectors concluded that the attributes of poor problem evaluation and nonconservative decision making had similar roots. Most of the inspection examples of poor problem evaluation involved the use of incorrect assumptions at either the individual or first organizational level. Many of the inspection examples involving nonconservative assumptions also had roots in the use of incorrect assumptions or poor problem evaluation. The inspectors identified two common threads associated with many of the examples in these adverse trends:

- Less than adequate licensing/design bases documentation

- Less than adequate application or understanding of the current licensing and design basis

Pacific Gas and Electric made a substantive commitment to correct station licensing/design bases documentation as part of the Licensing Bases Verification Program. These corrective actions are ongoing and long term. However, these trends also illustrated examples of either poor problem evaluation or decision making when correct licensing/design bases information was available to licensee personnel. The licensee's casual analysis acknowledged the need for changes to address these behaviors. However, the licensee has not implemented effective corrective actions to successfully address these concerns.

Effectiveness of Independent Assessment

The inspectors concluded that the licensee's independent assessments should have identified these adverse trends prior to the NRC. The inspectors reviewed past Qualify Verification and Nuclear Safety Oversight Committee audits and reports and interviewed members of the independent assessment and line organizations to determine why the independent assessments were not effective in identifying these trends. The inspectors identified:

- Poor line management accountability for resolution of independent assessment identified issues

- As discussed in FSARU Section 17.2.3, "Independent Review Program," the independent assessment function relied heavily on the Nuclear Safety Oversight Committee. However, the effectiveness of the committee to perform independent assessment functions was limited by the time the committee meets onsite, approximately 15 days per year.

- Poor resources dedicated to independent assessment functions (people and skill mix)

.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 2 corrective action items documenting:

- Notification 50301167, Unanalyzed Condition 230 kV
- Notification 50341634, Unit 1, Failure of steam flange required rapid power reduction

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

b. Findings

No findings were identified.

4OA3 Event Follow-up (71153)

.1 (Closed) Licensee Event Report 1-2010-004-00 Diablo Canyon Power Plant Pressurizer Level Control During Ramps and Degassing Operations

On June 29, 2010, Pacific Gas and Electric identified that a spurious safety injection could result in a pressurizer overfill condition and challenge the power-operated relief valves. The licensee concluded this condition was outside of the bounds of the plant safety analysis. The licensee determined that an inappropriate change to the plant procedures regarding the pressurizer program control band and volume control tank degassing led to the unanalyzed condition. Corrective actions included procedure changes to ensure pressurizer level was controlled within the FSARU described control band. The enforcement aspects of the performance deficiency are discussed in Section 4OA7. This licensee event report is closed.

.2 (Closed) Licensee Event Report 1-2010-001-01 Common Cause Control Room Ventilation Radiation Detector Failures

On October 13, 2009, the control room ventilation system automatically transferred to the pressurization mode of operation following a high radiation signal. Technicians observed indication of increasing radiation levels on Radiation Detectors 1-RM-25, 1-RM-26, and 2-RM-25. Technicians subsequently discovered that the high indicated radiation resulted from a malfunction caused by water intrusion following the heavy rains. Pacific Gas and Electric revised this licensee event report to include additional information related to manufacturer workmanship flaws and to address the cause of the water intrusion. The inspectors previously inspected this issue and documented this in Section 40A3 of NRC Integrated Inspection Report 05000275/2010002 and 05000323/2010002. The inspectors concluded that no other performance deficiencies existed. This licensee event report is closed.

.3 (Closed) Licensee Event Report 2-2010-002-00 Diablo Canyon Power Plant Unit 2 Safety Injection Test Line Unanalyzed Condition

On May 14, 2010, Pacific Gas and Electric discovered a Unit 1 design deficiency that could result in the loss of reactor coolant inventory exceeding the normal makeup capacity. The licensee identified the issue during the preparation of the safety injection test line optimization modification design change package. Plant engineers identified that the modification did not include the required flow restrictors at the reactor coolant pressure boundary isolation valves. The licensee concluded that the potential for excessive loss of reactor coolant system inventory only existed during shutdown conditions (Mode 4) when reactor coolant system check valve are tested. On June 14, 2010, the licensee determined that this design deficiency also impacted Unit 2. The safety injection modification package had been implemented during the most recent refueling outage. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50316384 and initiated corrective actions to install travel limiting stops on all of the new safety injection test line manual valves to limit the opening size prior to valve testing. The enforcement aspects of the performance deficiency are discussed in Section 4OA7.

.4 Unusual Event Following High Winds

On December 25, 2010, Pacific Gas and Electric declared an Unusual Event on Units 1 and 2 after a greater than 80 miles per hour wind gust occurred within the protected area. The high wind gust occurred as a winter storm front moved past the site. The licensee declared the Unusual Event at 4:25 p.m. and exited the event at 8:21 p.m. after the storm front had passed the plant. The resident inspectors responded to the site and performed an independent assessment of plant conditions, which were used in making decisions on NRC's responses to the event.

4OA5 Other Activities

.1 (Closed) Temporary Instruction (TI) 2515/179, "Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207"

a. Inspection Scope

An NRC inspection was performed to confirm that the licensee had reported the initial inventories of sealed sources pursuant to 10 CFR 20.2207 and to verify that the National Source Tracking System database correctly reflected the Category 1 and 2 sealed sources in custody of the licensee. Inspectors interviewed the licensee personnel and performed the following:

- Reviewed the licensee's source inventory
- Verified the presence of any Category 1 or 2 sources
- Reviewed procedures for and evaluated the effectiveness of storage and handling of sources
- Reviewed documents involving transactions of sources

- Reviewed adequacy of licensee maintenance, posting, and labeling of nationally tracked sources

b. Findings

No findings were identified.

.2 Reactor Vessel Head Replacement Inspection (71007)

Design and Planning Inspections (Section 02.02)

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 71007, "Reactor Vessel Head Replacement Inspection," to perform the following reactor vessel head design and planning inspection activities.

Engineering and Technical Support

The inspectors reviewed engineering and technical support activities performed prior to, and during the reactor pressure vessel head replacement outage. This review verified that selected design changes and modifications to structures, systems, and components described in the FSARU for transporting the new and old reactor vessel heads were reviewed in accordance with 10 CFR 50.59. Additionally, key design aspects and modifications associated with the reactor vessel head replacement were also reviewed. Finally, the inspectors determined that the licensee had confirmed that the new reactor pressure vessel head conformed to design requirements and that there were no fabrication deviations from design requirements.

Lifting and Rigging

The inspectors reviewed engineering design, modification, and analysis associated with reactor vessel head lifting and rigging activities. This included: (1) crane and rigging equipment; (2) reactor vessel head component drop analysis; (3) safe load paths; and (4) load lay-down areas.

Radiation Protection

The inspectors observed and reviewed the licensee's radiological controls for removal and storage of the old reactor head, and the movement of the new reactor head into containment. Radiation technicians were diligent in controlling the entire evolution of transporting the old head out of containment to the permanent storage facility.

Security Considerations and Adverse Impact to Other Unit

The inspectors observed security controls and reviewed security plans to verify that any potential adverse impacts on Unit 2 (the operating unit) caused by outage activities were minimized. The inspectors made frequent observations of security actions to verify that the licensee had implemented the appropriate controls for affected vital and protected area barriers during the reactor head replacement activities.

b. Findings

No findings were identified.

.3 Reactor Vessel Head Fabrication Inspections at Licensee Facility (Section 02.03)

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 71007 to perform a document review of the following reactor pressure vessel head fabrication inspection activities.

Heat Treatment

The inspectors reviewed documents to confirm that the material heat treatment used to enhance the mechanical properties of the reactor vessel head material carbon, low alloy, and high alloy chromium steels, were conducted per ASME Code and approved vendor procedures, consistent with the applicable ASME Code, Section III requirements. Also, the inspectors reviewed documents to assure that the following requirements were met: (1) furnace atmosphere; (2) furnace temperature distribution and calibration of measuring and recording devices; (3) thermocouple installation; (4) heating and cooling rates; (5) quenching methods; and (6) record and documentation requirements.

Nondestructive Examination

The Inspectors reviewed documents to ensure the manufacturing control plan included provisions for monitoring nondestructive examinations to ascertain that the nondestructive examinations were performed in accordance with applicable code, material specification, and contract requirements.

Welding

The inspectors reviewed the documentation for the weld overlay operations that established a layer of stainless steel cladding on the inside of the reactor vessel head to determine if it was accomplished per design. The inspectors also selected a sample of dome-to-flange and control rod drive mechanism flange-to-nozzle welds and reviewed the following items: (1) certified mill test reports of the dome, flange, weld material rods, and control rod drive mechanism nozzles; (2) certified mill test reports for the welding material for the reactor vessel head cladding; (3) cladding weld records, weld rod material control requisitions, traceability of weld material rods, weld procedure qualification, welder qualifications, and nonconformance reports; (4) control rod drive mechanism nozzle cladding welding inspection records, weld rod material control requisitions, traceability of weld material rods, weld procedure qualification, welder qualifications, and nonconformance reports; (5) control rod drive mechanism to nozzle welding and welds inspection records, weld rod material control requisitions, traceability of weld material rods, weld procedure qualification, welder qualifications, and nonconformance reports; and (6) nondestructive examination procedures, nondestructive examination records of the welds, nondestructive examination personnel qualifications, and certification of the nondestructive examination solvents.

Procedures

Inspections were completed to ensure that repair procedures had been established and that these procedures were consistent with applicable ASME Code, material specification, and contract requirements by verifying: (1) repair welding was conducted in accordance with procedures qualified to Section IX of the ASME Code; (2) all welders were qualified in accordance with Section IX of the ASME Code; (3) records of the repair were maintained; and (4) that requirements had been established for the preparation of certified material test reports and that the records of all required examinations and tests were traceable to the procedures to which they were performed.

Code Reconciliation

The inspectors reviewed the required documentation, supplemental examinations, analysis, and ASME Code documentation reconciliation to ensure that the original ASME Code N-Stamp remained valid, and that the replacement head complied with appropriate NRC rules and industry requirements. The inspectors also ensured that the design specification was reconciled and a design report was prepared for the reconciliation of the replacement head, verifying that the reports were certified by professional engineers.

Quality Assurance Program

The inspectors reviewed documentation to ensure that machining processes were carried out under a controlled system of operation, a drawing/document control system was in use in the manufacturing process, and that part identification and traceability was maintained throughout processing and was consistent with the manufacturer's quality assurance program.

Compliance Inspection

The inspectors verified that the original ASME Code, Section III, data packages for the replacement reactor vessel head were supplemented by documents included in the ASME Code Section XI (pre-service inspection) data packages; examined selected manufacturing and inspection records of the finished machined reactor vessel head; and verified compliance with applicable documentation requirements.

b. Findings

No findings were identified.

.4 Reactor Vessel Head Removal and Replacement Inspections

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 71007 to perform the following reactor vessel head removal and replacement inspection activities:

Lifting and Rigging

The inspectors reviewed preparations and procedures for rigging and heavy lifting including crane and rigging inspections, testing, equipment modifications, lay down area preparations, and training for the following activities:

- Area preparation for the outside systems
- Lattice boom crawler crane assembly
- Gantry lift system
- Outside bridge and trolley transfer system
- Reactor vessel head lift rig and polar crane
- Downender/upender fixture
- Old reactor vessel head removal
- New reactor vessel head placement

Major Structural Modifications

Pacific Gas and Electric did not make any major structural modifications that were made to facilitate reactor pressure vessel head replacement.

Containment Access and Integrity

The inspectors observed there were no modifications to the existing containment access structure or integrity to allow for the reactor vessel head to be removed and installed. The new and old reactor vessel heads were moved in and out of containment using the existing equipment hatch.

Outage Operating Conditions

The inspectors reviewed and observed the establishment of conditions including: (1) defueling; (2) reactor coolant system draindown; (3) system isolation; (4) safety tagging; (5) radiation protection controls; (6) controls for excluding foreign materials in the reactor vessel; (7) verification of the suitability of reinstalled (reused) components for use; and (8) the installation, use, and removal of temporary services. Section 1R20 of this report documents additional activities that were performed during the outage.

Storage of Removed Reactor Vessel Head

The inspectors reviewed the radiological safety plans and observed the transport, storage, and radiological surveys of the old reactor vessel head to its onsite storage location.

b. Findings

No findings were identified.

.5 Post Installation Verification and Testing Inspections

a. Inspection Scope

The inspectors used the guidance in Inspection Procedure 71007 to perform the following post installation verification and testing inspection activities: (1) containment testing; (2) licensee's post installation inspections and verifications program and its implementation; (3) reactor coolant system leakage testing and review of test results; (4) procedures required for equipment performance testing to confirm the design and to establish baseline measurements; and (5) pre-service inspection of new welds.

b. Findings

No findings were identified.

.6 IP 92723 Follow Up Inspection for Three or More Severity Level IV Traditional Enforcement Violations in the Same Area in a 12-Month Period

a. Inspection Scope

Consistent with the guidance provided in Inspection Procedure 92723, the inspectors evaluated the licensee's response to multiple Severity Level (SL) IV violations that occurred within a single traditional enforcement area. Specifically, the inspectors examined the licensee's response to a number of recent SL IV violations associated with impeding the regulatory process. These violations involved the following regulatory issues:

- Inadequate licensing basis impact evaluations (10 CFR 50.59)
- Accuracy and completeness of USFSAR (10 CFR 50.71(e))
- Failure to report LER's in accordance with 10 CFR 50.73
- Accuracy and completeness of information provided to the NRC (10 CFR 50.9 (a))

Documents reviewed by the inspectors are listed in the attachment.

b. Findings and Observations

1. Assessment

Based on the review of the licensee's Root Cause Evaluation Report, "Adverse Trend in Thoroughness of Problem Evaluation," and the associated Common Cause Analysis, "Impeding the Regulatory Process," it was determined that an extensive assessment of these conditions had been performed including the identification of corrective and preventive measures. It was also determined that the licensee had implemented a wide range of actions to address the adverse trend in problem identification. However, the inspectors were unable to verify that these actions have been effective in substantially mitigating the associated conditions.

2. Inadequate Quality Assurance Audits of the Corrective Action Program

Introduction. The inspectors identified a Green noncited violation of 10 CFR Part 50 Appendix B, Criterion XVIII, "Audits", which required that a comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. Contrary to this requirement, from August 2008 until December 09, 2010, audits performed by Pacific Gas and Electric had not been effective in identifying programmatic deficiencies in the implementation of the corrective action program.

Description. In December 2010, the inspectors identified a performance deficiency in the audits performed from 2008 to the present. These audits had not been effective in identifying programmatic deficiencies in the implementation of the licensee's corrective action program. Specifically, the quality verification audit performed in August 2008 evaluating the effectiveness of the corrective action program concluded that this activity was satisfactorily implemented. However, this audit failed to adequately address an adverse trend in the problem evaluation process documented in NRC Integrated Inspection Report 2008005, which identified eleven examples of an adverse trend in problem evaluation. Although subsequent actions have been implemented to enhance the quality verification oversight process, there was no objective evidence to establish that the noted inadequacies in the licensee's audit program had been specifically addressed in the licensee's root cause evaluation report. Section 17.18 of the licensee's Final Safety Analysis, Updated, "Quality Assurance", required that the adequacy and effectiveness of the QA program shall be continuously monitored through a comprehensive system of internal and supplier audits. The audit system implemented by the Quality Verification organization shall include all aspects of the QA program including identifying any deficiencies or nonconformances in the QA program, and shall assess the adequacy and effectiveness of the QA program. However, from August of 2008 until the present, the licensee had failed to ensure that a comprehensive system of planned and periodic audits was implemented to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. It was also noted that the scope of the licensee's corrective and preventive actions may not have been comprehensive enough to address all of the leadership issues identified in the root cause evaluation report. Specifically, issues related to how plant personnel apply current licensing basis information to emergent issues including prompt operability assessments continue to manifest themselves.

Analysis. The inspectors determined that the licensee's failure to properly implement the requirements contained in their quality assurance program and 10 CFR Part 50 Appendix B, Criterion XVIII was a performance deficiency. The finding was more than minor because it is associated with the procedure quality attribute of the Mitigating Systems Cornerstone, and affected the cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the significance of the finding using IMC 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," and determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to coordinate and communicate the results from assessments to affected personnel, and track the corrective actions to address issues commensurate with their significance [P.3(c)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion XVIII, "Audits," required, in part, that a comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits were required to be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audit results were required to be documented and reviewed by management having responsibility in the area audited. Follow up action, including re-audit of deficient areas, was required to be taken where indicated. Contrary to this requirement, from August 2008 until December 9, 2010, Pacific Gas and Electric failed to ensure that a comprehensive system of planned and periodic audits were carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. Because this violation was of very low safety significance (Green) and has been entered into the licensee's corrective action program as Notification 50365083, this violation is being treated as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275/2010005-07, 05000323/2010005-07, "Inadequate Quality Verification Audits."

4OA6 Meetings

Exit Meeting Summary

On October 20, 2010, the inspectors presented the inspection results of the review of inservice inspection activities to Mr. L. Sharp, Senior Director, Engineering Services, 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. The inspectors acknowledged review of proprietary material during the inspection which had been or would be returned to the licensee, or appropriately destroyed.

On October 21, 2010, the inspectors presented the results of the radiation safety inspections 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.

On November 4, 2010, the inspectors briefed the results of the annual licensed operator requalification program inspection to Mr. Bill Hendy, Operations Training Manager. The licensee representative acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

On December 15, 2010, the inspectors presented the inspection results of the heat sink performance inspection 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.

On January 4, 2011, 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 violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as noncited violations:

.1 Technical Specification 5.4.1.a required that the licensee establish, implement, and maintain written procedures covering the applicable activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 recommended procedures required for shutdown of pressurized water reactors. Contrary to this, Pacific Gas and Electric failed to properly implement procedures for shutdown, as required by Technical Specification 5.4.1.a. Specifically, on October 5, 2010, operators failed to follow Operating Procedure OP L-6, "Cold Shutdown/Refueling", Step 5.9.6 and establish a reactor coolant system vent path before reducing reactor coolant system temperature below 90°F. As a result, operators declared the Low Temperature Overpressure Protection System inoperable in accordance with Technical Specification 3.4.12 and restored temperature above 90°F. Pacific Gas and Electric entered the issue into the corrective action program as Notification 50347713. Using Appendix G of Inspection Manual Chapter 0609, "Shutdown Operations Significance Determination Process," the inspectors concluded this finding was of very low safety significance because the licensee maintained an adequate mitigation capability during shutdown and the issue did not require a quantitative assessment.

.2 Title 10 CFR 50.59 c(2)(vi) required that a licensee obtain a license amendment pursuant to Sec. 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would create a possibility for a malfunction of a structures, systems, and components important to safety with a different result from any previously evaluated in the FSARU. Contrary to this, Pacific Gas and Electric failed to obtain a license amendment prior to implementing revisions to plant operating procedures. Specifically, on June 29, 2010, Pacific Gas and Electric identified that revisions to Operating Procedure OP L-4, "Normal Operation at Power," and OP L-5, "Plant Cooldown from Minimum Load to Cold Shutdown" allowed operators to control pressurizer level above the program control band during brief periods. The licensee concluded that operation above the program control band could result in a pressurizer overfill condition that could challenge the performance of the pressurizer power-operated relief valves if a spurious safety injection occurred. The licensee entered the issue into the corrective action program as Notification 50320032. The inspectors concluded this finding was of very low safety significance because it is a design deficiency confirmed not to result in the loss of operability or functionality.

.3 Title 10 CFR Part 50, Appendix B, Section III, "Design Control," required, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to this, Pacific Gas and Electric failed to establish measures to ensure that applicable regulatory requirements and the design bases were correctly translated into specifications, drawings, procedures, and instructions. Specifically, on May 14, 2010, Pacific Gas and Electric discovered a design deficiency during the final preparation of the Diablo Canyon Power Plant Unit 1 design change package for the safety injection test line optimization modification. The package lacked the required flow restrictors for the reactor coolant pressure boundary isolation valves, resulting in the potential to create loss of reactor coolant system inventory in Mode 4 which would exceed normal charging makeup capability. On June 14, 2010, the licensee determined that this design deficiency impacted the Unit 2 safety injection modification that had been implemented during the most recent refueling outage. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50316384 and initiated corrective actions to install travel limiting stops on all of the new safety injection test line manual valves to limit the opening size prior to valve testing. The inspectors concluded this finding was of very low safety significance because it is a design deficiency confirmed not to result in the loss of operability or functionality.

.4 Title 10 CFR Part 50, Appendix B, Section XI, “Test Control,” required, in part, that the test program demonstrate that systems and components will perform satisfactorily in service and is performed in accordance with written test procedures that incorporate the requirement and acceptance limits contained in applicable design documents. Contrary to this, Pacific Gas and Electric failed to ensure that testing demonstrated that systems and components would perform satisfactorily in service and perform in accordance with written test procedures which incorporated the requirements and acceptance limits into applicable design documents. Specifically, on November 14, 2010, the licensee performed comprehensive pump testing of Unit 1 turbine-driven auxiliary feedwater pump following replacement of the turbine governor during the most recent refueling outage. The licensee adjusted governor speed settings to meet the pump speed criteria of Procedure STP P-AFW-A11, “Comprehensive Testing of Turbine-Driven Auxiliary Feedwater Pump 1-2,” Revision 6. Testing personnel identified that the turbine speed settings differed from the previous governor. Following an engineering evaluation, operators performed a surveillance test of the pump and discovered that turbine speed exceeded the test acceptance criteria. The licensee concluded that the new governor had different performance characteristics from the previous governor. Pacific Gas and Electric failed to revise the test procedures to reflect the new governor performance characteristics. As a result, the licensee inappropriately adjusted the governor settings. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50316384 and initiated corrective actions to re-perform comprehensive testing to the pump and adjust the governor to meet pump acceptance criteria. The inspectors concluded this finding was of very low safety significance because it did not result in an actual loss of safety function of the pump.

.5 Title 10 CFR Part 50, Appendix B, Section XVI, “Corrective Action,” required measures be established to ensure that conditions adverse to quality are promptly identified and corrected. Contrary to this, Pacific Gas and Electric failed to ensure corrective actions to identify and correct fibrous material inside containment, as described in licensee Letter DCL-08-059, “Supplemental Response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors" (July 10, 2008). Subsequently, on October 14, 2010, the licensee identified approximately 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous material installed in the Unit 1 reactor coolant loop biological shield wall penetrations. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50355265 and initiated corrective actions to remove the fibrous insulation material. The inspectors concluded that this finding was of very low safety significance because it was a design deficiency confirmed not to result in the loss of operability or functionality.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Baldwin, Manager, Regulatory Services
J. Becker, Site Vice President
S. David, Director, Site Services
J. Fields, Auditor, Quality Verification
M. Gibbons, Manager, Maintenance
B. Hendy, Operations Training Manager
L. Hopson, Site Services Manager
N. Jahangir, Manager, Engineering
R. Lovell, Senior Consulting Engineer, Design Engineering
M. McCoy, NRC Interface, Regulatory Services
E. Nelson, Engineering Services Senior Manager
J. Nimick, Manager, Operations
K. Peters, Station Director
D Petersen, Quality Verification Director
L. Sharp, Senior Director, Engineering Services
M. Somerville, Manager, Radiation Protection
J. Welsch, Director, Operations Services
S. Westcott, Director, Engineering

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000275; 05000323/2010005-03	URI	Corrosion of Containment Fan Cooler Unit Cooling Coil Casings (Section 1R07)

Opened and Closed

05000275; 05000323/2010005-01	NCV	Failure to Maintain a Fire Barrier (Section 1R05)
05000323/2010005-02	NCV	Inadequate Transient Combustibles Procedure (Section 1R05)
05000275; 05000323/2010005-04	NCV	Inadequate Operability Determinations (Section 1R15)
05000275; 05000323/2010005-05	NCV	Less than Adequate Containment Recirculation Sump Design Control (Section 1R18)
05000275; 05000323/2010005-06	NCV	Inadequate Emergency Diesel Generator Surveillance Testing (Section 1R19)
05000275; 05000323/2010005-07	NCV	Inadequate Quality Verification Audits (Section 4OA5)

Closed

1-2010-004-00	LER	Diablo Canyon Power Plant Pressurizer Level Control During Ramps and Degassing Operations
1-2010-001-01	LER	Common Cause Control Room Ventilation Radiation Detector Failures
2-2010-002-00	LER	Diablo Canyon Power Plant Unit 2 SI Test Line Unanalyzed Condition

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
AR PK15 05	Ambient Temperature PPC Alarm	18
OP O-28	Intake Management	11
CP M-12	Stranded Plant	4

NOTIFICATIONS

50349892	50345653

Section 1R04: Equipment Alignments

PROCEDURES/DRAWINGS

NUMBER	TITLE	REVISION
107703, Sheet 3	Auxiliary Feedwater System Drawing	50
OP D-1:II	U2, Auxiliary Feedwater System Alignment Verification for Plant Startup	27
107723, Sheet 9	Auxiliary Building Ventilation System Drawing	63
OP H-1:II	U2, Auxiliary Building Safeguards Ventilation (ABVS) – Normal Operation	9

Section 1R06: Flood Protection Measures

PROCEDURES

NUMBER	TITLE	REVISION
OP O-28	Intake Management	11
DCM T-12	Pipe Break, Flooding and Missiles	17A
FSARU 9.3.3	Equipment and Floor Drainage Systems	19
DCM S-21	Diesel Engine System	23

NOTIFICATIONS

50203329	50360328	50203329

Section 1R07: Heat Sink Performance

PROCEDURES

NUMBER	TITLE	REVISION
CAP E-4	Auxiliary Saltwater Sampling	16
OP F-2:1	Component Cooling Water System – Make Available	33
PEP M-234	CCW Heat Exchanger Performance Test	11
STP I-1A	Routine Shift Checks Required by Licenses	117
STP I-1C	Routine Weekly Checks Required by Licenses	93
STP M-51	Routine Surveillance Test of Containment Fan Cooler Units	29
STP M-93A	Refueling Interval Surveillance – Containment Fan Cooler System	20
STP V-13A	CCW Flow Balancing	18

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
2002-S023-002	Diablo Canyon Power Plant Health Issue	August 17, 2010
2002-S023-003	Diablo Canyon Power Plant Health Issue	August 17, 2010
420DC.09.29	DCPP CCW 2-1 and 2-2 Heart Exchanger tests Pre 2R15	September 16, 2009
420DC-10.33	DCPP CCW 1-1 and 1-2 Heart Exchanger tests Pre 1R16	October 4, 2010
DCM No. S-14	Design Criteria Memorandum S-14 Component Cooling Water System	22
DCM No. S-23A	Design Criteria Memorandum S-23A Containment HVAC System	20
	PHC Presentation: Containment Fan Cooler Units Cooling Coils	June 10, 2008
	Project Review Committee Presentation: U1/U2: Replace CFCU Cooling Coils	April 29, 2009
663079	Containment Fan Cooler Cooling Coil Coilbank Assembly	5

	Drawing	
	Phase 1 Containment Fan Cooling Coil Study for DCPP	0

CALCULATIONS

NUMBER	TITLE	REVISION
M-962	CCW Maximum Allowable Differential Pressure – Performance Based	3
M-1017	Component Cooling Water System (CCW) – To determine flows in the CCW System	5
M-1019	Evaluate Heat Removal Capability of the CCW System Following a LOCA	0
M-1027	Maximum ASW Temperature with Two CCW HXs	3

NOTIFICATIONS/ACTION REQUESTS

50034303	50038009	50045336	50198737	50199694
50285172	50293492	50364991	A0371869	A0694722
A0695269	A0697097	A0719450	A0721872	A0721874
A0725518				

Section 1R08: Inservice Inspection Activities

NOTIFICATIONS

50346137	50352439	50341111	5026896	50346137
50269107	50214568	50339483	50350525	50269108
50269093	50346268	50350930	50352022	50346138
50214592	50351088	50346224	50044275	50346312
50351206	50259535	50259535	50194685	50347020
50351354	50351936	50346264	50214599	50351355
50352294	50346244	50347012	503468531	60018048
80347017				

CERTIFIED MATERIAL TEST REPORTS FOR DIABLO CANYON AGREEMENT 3500720877

001/2008	006/2008	011/2008	018/2008	002/2008
007/2008	013/2008	013/2008	021/2008	003/2008
008/2008	014/2008	022/2008	004/2008	009/2008
015/2008	005/2008	010/2008	016/2008	

CONTRACT VARIATION APPROVAL REQUESTS

87-9044795-000	87-9128865-000	87-9121324-000	87-9086792-000
87-9061641-000	87-9045826-000	87-9032552-000	87-9062179-000
87-9064078-000	87-9080500-000	87-9113362-001	

DRAWINGS

NUMBER	TITLE	REVISION
1C83374	Replacement Steam Generator Partition Plate and Partition Stub	1
1C83374, Sheet 1	Replacement Steam Generator Partition Plate and Partition Stub	1
2110-49739-004	Old Steam Generator Storage Facility: Bldg 403 Concrete floor Plan	3
6656E33, Sheets 1-4	Replacement Steam Generator Tube Plate Surfacing Weld and Machining	3
6656E33, Sheets 1-4	Replacement Steam Generator Tube Plate Surfacing Weld and Machining	4
6656E45, Sheets 1-5	Replacement Steam Generator Channel Head Weld Assembly and Final Machining	4
6656E45, Sheets 1-5	Replacement Steam Generator Channel Head Weld Assembly and Final Machining	4
6656E61	Replacement Steam Generator General Arrangement Diablo Canyon Units 1 and 2	2

INSTRUCTOR LESSON GUIDE

NUMBER	TITLE	REVISION / DATE
TU0932	DCPP Boric Acid Corrosion Control Program	0
TU09NL2	ESP Newsletter 2nd Quarter 2009	0
	Chemistry & Radiation Protection Technician Training	August 5, 2008
NBAC	DCPP Boric Acid Corrosion Control Program	0
R096C8	DCPP Boric Acid Corrosion Control Program	0A
EADM18	Intro to Engineering Programs	1

NDE EXAMINATION DATA SHEETS

NUMBER	TITLE	DATE
NDE VT 2-1	Report of Section XI System Pressure Test, Pressurizer 1-1 Manway	October 18, 2010
NDE VT 1-1	Visual Examination VT-1, Pressurizer Manway Bolts	October 18, 2010
NDE UT-4	Ultrasonic Examination Data Sheet, Weld ID FW-11.07.01, SG 1-1 Tubesheet to Shell	October 11, 2010
QV Report No. 10-107 Pre-Freeze Seal	Liquid Penetrant Exam Data Sheet, Component ID CCW-1-RV-47	October 14, 2010
QV Report No. 10-107 Pre-Freeze Seal	Liquid Penetrant Exam Data Sheet, Component ID CCW-1-RV-47	October 15, 2010
NDE VT 3-1	Report of Visual Examination of Support for Piping or Component - Section XI, Hanger # 10-31R, Drawing # 049308	October 8, 2010
NDE VT 3-1	Report of Visual Examination of Support for Piping or Component - Section XI, Hanger # 0181-29, Drawing # 6000460	October 8, 2010

PROCEDURES

NUMBER	TITLE	REVISION/DATE
AD4.ID2	Plant Leakage Evaluation	10
MP M-54.3	Freeze Sealing of Piping	18A March 21, 2010
NDE MT-1	Magnetic Particle Examination Procedure	13
NDE PT-1	Visible Dye Liquid Penetrant Examination Procedure	3
NDE RT-1	Radiographic Examination Procedure for Welds	13
STP R-8A	Reactor Coolant System Leakage Test	15
STP R-8C	Containment Walkdown for Evidence of Boric Acid Leakage	8A September 21, 2009
STP R-8C	Containment Walkdown for Evidence of Boric Acid Leakage	9
54-ISI-400-018	Nondestructive Examination Procedure Multi-frequency Eddy Current Examinations of Tubing	August 4, 2010
51-9118042-002	Diablo Canyon EPRI Appendix H Eddy Current Site Validation	

NDE ET-7	Eddy Current Examination of Steam Generator Tubing	13
Steam Generator ECT ETSS	1 Coil Rotating Probe Examination	1
Steam Generator ECT ETSS	Bobbin Probe Examination	2
Steam Generator ECT ETSS	3 Coil Rotating Probe Examination	1
U-1, DC-1-04-M-HX-SG3	SG 1-3 AVB Wear on Tube in 1R16	October 15, 2010

<u>RELIEF REQUESTS</u>

NUMBER	TITLE	DATE
PG&E Letter DCL-07-038	Diablo Canyon Unit 2 – ASME Section XI Inservice Inspection Program Relief Request REP-1 U2	March 28, 2007
TAC No's: MD8646 MD8647	Diablo Canyon Power Plant, Unit No's 1 and 2, - Approval of Relief Request NDE-SBR for Snubber Visual Examination and Functional Testing Related to the Third 10-Year Interval Inservice Inspection Program	February 25, 2009
TAC No. ME0200	Diablo Canyon Power Plant, Unit No. 1- Approval of Request for Relief from 10 CFR 50.55a(g)(6)(ii)(D)(3) Requirement for Demonstrated Volumetric Leak Path Assessment	April 8, 2009
PG&E Letter DCL-10-051	Diablo Canyon Power Plant (DCPP) Unit 1- ASME Section XI Inservice Inspection Program Relief Request NDE-RCS-SE-1R16 to Allow Use of Alternate Sizing Qualification Criteria	May 17, 2010
TAC No. ME3942	Diablo Canyon Power Plant, Unit No. 1 – Approval of Request for Relief NDE-RCS-SE-1R16 from Examination Requirements of ASME Code, Section XI, Appendix VIII, Supplement 10, Root Mean Square Error	July 23, 2010

<u>MISCELLANEOUS</u>

NUMBER	TITLE	REVISION/DATE
TSI.ID3	Steam Generator Management Program	11
	Steam Generator Degradation Assessment	October 13, 2010
ER1.ID2	Boric Acid Corrosion Control Program	4

DCL-88-143	Response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"	June 2, 1988
OM7.ID1	Problem Identification and Resolution	33
420DC-10.24	Inservice IWL Examination of the Containment Concrete for Unit 1	1
ENGISI7	Boric Acid Corrosion Evaluator	1
000021350	Training Improvement Proposal	4
NDE VT 3C-1	VT-3C Visual Examination of the Containment Concrete Shell	3
NDE UT-4	Ultrasonic Examination of Pressure Vessel Welds Other than Reactor Vessels	1

Section 1R11: Licensed Operator Requalification Program

PROCEDURES

NUMBER	TITLE	REVISION
R104-ST	Simulator Exam – Faulted Steam Generator	12
E2ECA21B	Simulator Exam – Faulted Steam Generator	12

Section 1R12: Maintenance Effectiveness

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	(a)(1) Goal Setting Summary Report	10/4/2010
	(a)(1) Goal Setting Summary Report	11/2/2010
	System 43A PPC Scoping Document	2

NOTIFICATIONS

50340508	50252762	50032946

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

NOTIFICATIONS/ACTION REQUESTS

50344855	50347313	50346952	A0678820	50346671
50345630	50345631	50343303	50340070	

Section 1R15: Operability Evaluations

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
Calculation N-100	Maximum Flow from ECCS	4
OM7.ID12	Operability Determinations	17
CN-CRA-10-45	Steam Generator Tube Rupture Margin to Overfill Analysis	0
OP O-2	Operation of Hagan Controllers	15

NOTIFICATIONS

50361413	50362695	50362966	50360530

Section 1R18: Plant Modifications

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE
DCP 1000000397	Recirculation Sump Flow Path Modification	0
DCN 2000000758	Replacement of RCDT Concrete Hatch Covers with Grating	10/29/2010

Section 1R19: Postmaintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION / DATE
STR R-8A	Reactor Coolant System Leakage Test	115
NS-FSI-08-11	DPRI System Data Cabinet Cable/Connector Replacement Diablo Canyon Unit (Order 68008462-0030)	Rev.15/Oct. 22, 2010
STP R-27	Thermocouple/ RCS RTD Cross Calibration	Oct. 29, 2010
NS-VICO-03-01	CRDM Cable and Connector Testing (Order 68008483-0010)	Oct. 25, 2010
PMT 07.32	Unit 1 Reactor Head Replacement Project Testing, (Order 68007794-020)	Oct. 30, 2010
PMT 07.32	Maintain temperature of CRDM coil stacks during operation. (Order 68007794-020)	Oct. 30, 2010
STP I-87B5	Reactor Vessel Level Indication System DP3 Normalization Procedure	7
NS-FSI-08-11	DPRI System Data Cabinet Cable /Connector Replacement Diablo Canyon Unit (Order 68008462-0030)	Oct. 22, 2010

STP R-27	Thermocouple/ RCS RTD Cross Calibration	Oct. 29, 2010
NS-VICO-03-01	CRDM Cable and Connector Testing (Order 68008483-0010)	Oct. 25, 2010
PMT 07.32	Unit 1 Reactor Head Replacement Project Testing, (Order 68007794-020)	Oct. 30, 2010
PMT 07.32	Maintain temperature of CRDM coil stacks during operation. (Order 68007794-020)	Oct. 30,2010
MP I-1.6-8	Rod Control DC Hold Test,	2
STP 1-7-Y700.B	Digital Metal Impact Monitoring System Channel Calibration	17
Areva 51-9122752	Applicable Documents Index for Post Outage	0
STP R-1B	Rod Drop Measurement	35
MP I 1.6-6	Rod Control Coil Regulation Verification	1
MP I 1.6-6	Rod Control Coil Regulation Verification	1
MP I 1.6-5	Rod Control Slave Cycle Order	1
STP V-2U2D	Exercising S/G No. 2 AFW Supply Valves LCV-107 and LCV-111	5A
STP V-3P6A	Exercising Valves LCV-110 and LCV-111 Auxiliary Feedwater Pump Discharge	15
MP I-3-L111	Steam Generator 1-2 Aux FW Supply Level Control Channel LCV-111 Calibration	15
TP TO-10021	1R16 Fill and Vent of AFW Lines for LCV-111 Repairs	0
STP M-9L	Diesel Generator Shutdown Lockout Relay Test	27
STP M-21-RTS.1	Return Diesel Engine to Service Following Outage Maintenance	8
STP M-21-A.1	Diesel Engine Analysis	7
STP M-9B	Overspeed Trip Test of Diesel Generators	25
STP M-9D1	Diesel Generator Full Load Rejection Test	17
Calc M-800	Establish Auxiliary Feedwater Turbine Overspeed and Operating Speed Ranges	3

NOTICATIONS

50358757	50346969	50232184	50207912	50359610

Section 1R20: Refueling and Other Outage Activities

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION / DATE

PTLR-1	PTLR for Diablo Canyon	10
Calc STA-249	LTOP Temperature Limits for 23 EFPY with RSG	0
Calc SDP 10-04	Evaluation of Disabling LTOP During 1R16	0
AD8.DC55	Outage Safety Scheduling	31
OP O-32	Control of Refueling Tags	3
OP L-5	U-1, Plant Cooldown From Minimum Load to Cold Shutdown	86
OP L-6	U-1, Cold Shutdown/Refueling	61
OP L-1	U-1, Plant Heatup From Hot Shutdown to Hot Standby	79
OP L-2	U-1 & U-2, Hot Standby to Startup Mode	39
OP AP SD-2	U-1 & U-2, Loss of RCS Inventory	18
OP AP SD-5	U-1 & U-2, Loss Residual Heat Removal	9A
OP A-2:II	Reactor Vessel – Draining the RCS to the Vessel Flange – With Fuel in Vessel	39
R102C3	Instructor Lesson Guide – Outage Safety Plan	1
STP M-45B	Containment Inspection when Containment Integrity is Established	17
STP M-45A	Containment Inspection Prior to Establishing Containment Integrity	28
OM7.ID1	Problem Identification and Resolution	35
STP M-45C	Outage Management Containment Inspections	9

NOTIFICATIONS

50347713	50341634	50357702	50357835	50357839
50357901	50357904	50357905	50357909	50357950
50357951				

Section 1R22: Surveillance Testing

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
STP R-8A	Reactor Coolant System Leakage Test	15
STP P-CCW-12	Routine Surveillance test of Component Cooling Water Pump 1-2	12
STP V-654	Penetration 54 Containment Isolation Valve Leak Testing	16
STP V-621	Penetration 21 Containment Isolation Valve Leak Testing	10
STP P-AFW-A11	Comprehensive Testing of Turbine-Driven Auxiliary Feedwater Pump 1-1	6

STP P-AFW-11	Routine Surveillance Test of Turbine-Driven Auxiliary Feedwater Pump 1-1	27
STP P-DFO-01	Routine Surveillance of the Diesel Generator Fuel Oil Transfer Pump	7
Calc STA-135	Auxiliary Feedwater Pump Acceptance Curves	2

NOTIFICATIONS

50361635	50361660	50361727	50360893	50361089

Section 1EP6: Drill Evaluation

PROCEDURES/DOCUMENTS

NUMBER	TITLE	REVISION
E2ECA21B	Simulator Lesson Plan	19

Section 2RS01: Radiological Hazard Assessment and Exposure Controls

PROCEDURES

NUMBER	TITLE	REVISION
RCP D-220	Control of Access to High, Locked High, and Very High Radiation Areas	37A
RCP D-240	Radiological Posting	20
RCP D-500	Routine and Job Coverage Surveys	31
RCP D-620	Control of Radioactive Sources	7

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
100610010	Radiation Protection Programs Audit	July 27, 2010

NOTIFICATIONS

50344892	50342631	50335303	50317135	50315106
50309721	50307717	50303388		

RADIATION SURVEY RECORDS

11622	11663	11669	11670	11673
12262	12294	12296	12316	12330

Section 2RS02: Occupational ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
RCP D-200	ALARA Planning and Controls	47
RCP D-201	Writing Radiation Work Permits	1
RCP D-500	Routine and Job Coverage Surveys	31
RP1.ID1	Requirements for the ALARA Program	6
RP1.ID2	Use and Control of Temporary Radiation Shielding	9

NOTIFICATIONS

50279263	50302254	50306356	50317135	50315091
60024013	50338595	50341825	50335405	50341634

RADIATION WORK PERMITS

NUMBER	TITLE
10-1020	1R16 Reactor Disassembly
10-1023	1R16 Fuel Movement and Underwater Work in Containment
10-1026	1R16 Lower Covity and Transfer Canal Work
10-1050	1R16 RCP Pump Maintenance
10-1067	SI Test Header Optimizing Project
10-1133	1R16 Reactor Head Project ORVCH Disassembly

Section 4OA2 Identification and Resolution of Problems

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
License Bases Impact Evaluation MMD M00085	POA Update for Unit 1, Cycle 17 Reload Evaluation	0
PGE-10-54	230 kV Degraded Voltage Evaluation – Engineering Report	1
	Diablo Canyon Unit Cycle 17 Reload Analysis	Aug. 2010
COLR 1	COLR for Diablo Canyon Unit 1	0
Order 64018978	4 kV Vital Bus UV Relay Cal	0
Filenet # 092530004	DCPP Pre-NIEP Self Assessment Report	1

Audit #100330019,	Audit Exit Meeting – Audit of 1R16 Design Changes	0
OM7.ID1	Problem Identification and Resolution	35
FileNet # 102730053	Observer Department Report for Quality Verification	0
FileNet # 102730052	Observer Department Report for Quality Verification	0
FileNet # 102730057	Observer Department Report for Quality Verification	0
FileNet # 102730055	Observer Department Report for Quality Verification	0

Section 4OA3 Event Follow-Up

PROCEDURES

NUMBER	TITLE	REVISION

Section 4OA5 Temporary Instruction 2515/179

PROCEDURES

NUMBER	TITLE	REVISION
RCP D-620	Control of Radioactive Sources	7

Section 4OA5 Other Activities, Reactor Vessel Head Replacement Inspection (71007)

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
12-5-HT/C-100	Postweld Heat Treatment of a PG&E RRVHC	1
12-5-HT-83	Preheat, Interpass and Post Weld Temperature Control for Commercial ASME Contracts Fabricated for AREVA-NP	2
08-9023468	Replacement Reactor Vessel Closure Head Forgings Pacific Gas & Electric Diablo Canyon Units 1 & 2	1
QP-0506-01	Diablo Canyon Unit 1 Replacement Reactor Vessel Closure Head Quality Plan	2
700-0168-52	Section I, Postweld Heat Treatment Unit 1	June 26, 2009

JQA-08-008	Replacement Reactor Vessel Closure Head	February 5, 2008
33-9116825	ASME Design Report for Diablo Canyon Unit 1 Replacement Reactor Vessel Closure Head	2
QAL-8-61	RRV Closure Head Final Stress Relief	1
08-9031646	Alloy 690 Material for Reactor Vessel Head Nozzles Pacific Gas & Electric, Diablo Units 1 & 2	0
N-7825-10	Technical Manufacturing Program for Forged Part for Closure Head	G

Section 4OA5 Other Activities, IP 92723 Follow Up Inspection for Three or More Severity Lever IV Traditional Enforcement Violations in the Same Area in a 12-Month Period

Documents Reviewed

Root Cause Evaluation Report, Adverse Trend in Thoroughness of Problem Evaluation, Revision, 06/07/2010

Common Cause Analysis: Impeding the Regulatory Process, Reference No. 50331845

Common Cause Analysis: Non-Conservative Assumptions-H.1 (b), Reference No. 50322060

2008 Corrective Program Audit, dated August 14, 2008, File No. EDMS # 1290001

DCPP, Units 1&2, UFSAR, Chapter 17, Quality Assurance, Rev. 5/19/2010

Nuclear Industry Evaluation Program, Audit of DCPP Quality Organization, Dated 07/28/2008

Nuclear Industry Evaluation Program, Audit of DCPP Quality Organization, Dated 03/26/2010

Quality Verification Audit of Design Modifications, Dated 0/16/2008, File # 080910025

DCPP Internal & External Audit Schedule, 2010

Technical Specification And testing Audit, dated 10/23/2009, File # 091480003

DCPP Quality Performance Assessment Report, First Period, Revision 1, Dated 5/21/2009\

DCPP Quality Performance Assessment Report, Second Period, Dated 9/10/2009

Procedures Reviewed

DCPP Procedure OM7.ID3, “Root Cause Investigation- Root Cause Team,” Rev.22

DCPP Procedure OM7.ID1, “Problem Identification and Resolution,” Rev. 34

DCPP Procedure AD7.DC8, “Work Control,” Revision 33

DCPP Procedure OM4.ID13, “Nuclear Power Generation Internal Auditing,” Rev. 15

DCPP Procedure OM4.NQ5, “Quality Verification Internal Auditing,” Rev. 13A

DCPP Procedure OM4.NQ2, “Quality Verification Assessments,” Rev. 8

DCPP Procedure OM4.NQ1, “Self-Assessments of the Quality Assurance Organization,” Rev. 5

DCPP Procedure OM5.ID2, “Stop work Authority,” Rev. 5

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