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

May 5, 2009

John T. Conway
Senior Vice President and Chief Nuclear Officer
Pacific Gas and Electric Company
P.O. Box 3
Mail Code 104/6/601
Avila Beach, California 93424

Subject: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2009002 AND 05000323/2009002

Dear Mr. Conway:

On March 31, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Diablo Canyon Power Plant. The enclosed integrated inspection report documents the inspection findings, which were discussed on April 2, 2009, with Mr. James Becker, Site Vice President and other members of your staff.

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

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket: 50-275
50-323

License: DPR-80
DPR-82

Enclosure:
NRC Inspection Report 05000275/2009002 and 0500323/2009002
w/Attachment: Supplemental Information

cc w/Enclosure:
Sierra Club San Lucia Chapter
ATTN: Andrew Christie
P.O. Box 15755
San Luis Obispo, CA 93406

Nancy Culver
San Luis Obispo
Mothers for Peace
P.O. Box 164
Pismo Beach, CA 93448

Chairman
San Luis Obispo County
Board of Supervisors
1055 Monterey Street, Suite D430
San Luis Obispo, CA 93408

Truman Burns\Robert Kinosian
California Public Utilities Commission
505 Van Ness Ave., Rm. 4102
San Francisco, CA 94102

Diablo Canyon Independent Safety Committee
Attn: Robert R. Wellington, Esq.
Legal Counsel
857 Cass Street, Suite D
Monterey, CA 93940

Director, Radiological Health Branch
State Department of Health Services
P.O. Box 997414 (MS 7610)
Sacramento, CA 95899-7414

City Editor
The Tribune
3825 South Higuera Street
P.O. Box 112
San Luis Obispo, CA 93406-0112

James D. Boyd, Commissioner
California Energy Commission
1516 Ninth Street (MS 31)
Sacramento, CA 95814

James R. Becker, Site Vice President
Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

Jennifer Tang
Field Representative
United States Senator Barbara Boxer
1700 Montgomery Street, Suite 240
San Francisco, CA 94111

Chief, Radiological Emergency Preparedness Section
Chemical and Nuclear Preparedness and
Protection Division
Department of Homeland Security
1111 Broadway, Suite 1200
Oakland, CA 94607-4052

Chairperson, Radiological Assistance Committee
Federal Emergency Management Agency
Region IX
Department of Homeland Security
1111 Broadway, Suite 1200
Oakland, CA 94607-4052

Electronic distribution by RIV:
Regional Administrator (Elmo.Collins@nrc.gov)
Deputy Regional Administrator (Chuck.Casto@nrc.gov)
DRP Director (Dwight.Chamberlain@nrc.gov)
DRP Deputy Director (Anton.Vegel@nrc.gov)
DRS Director (Roy.Caniano@nrc.gov)
DRS Deputy Director (Troy.Pruett@nrc.gov)
Senior Resident Inspector (Michael.Peck@nrc.gov)
Resident Inspector (Tony.Brown@nrc.gov)
Branch Chief, DRP/B (Vincent.Gaddy@nrc.gov)
Senior Project Engineer, DRP/B (Rick.Deese@nrc.gov)
DC Site Secretary (Agnes.Chan@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Team Leader, DRP/TSS (Chuck.Paulk@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)

Only inspection reports to the following:
DRS STA (Dale.Powers@nrc.gov)
John Adams OEDO RIV Coordinator (John.Adams@nrc.gov)
ROPreports
W. A. Maier, RSLO (Bill.Maier@nrc.gov)
R. E. Kahler, NSIR (Robert.Kahler@nrc.gov)
NSIR/DPR/EP (Steve.LaVie@nrc.gov)

File located: R:_REACTORS_DC\2009\DC2009-02RP-mspADAMS.doc ML091250142

SUNSI Rev Compl.	☒ Yes ☐ No	ADAMS	☒ Yes ☐ No	Reviewer Initials	Vgg
Publicly Avail	☒ Yes ☐ No	Sensitive	☐ Yes ☒ No	Sens. Type Initials	Vgg

RIV:RI:DRP/B	SRI:DRP/B	C:DRS/EB1	C:DRS/PSB2	C:/DRS/PSB1
MABrown	MSPeck	TRFarnholtz	GEWerner	MShannon
/RA/-email	***/RA/-email***	***/RA/***	***/RA/***	***/RA***
04/30/2009	04/30/2009	04/24/2009	04/27/2009	04/27/2009
C:/DRS/EB2	C:DRS/OB	C:DRP/B		
NOkeefe	RELantz	VGGaddy		
/RA/	***/RA/***	***/RA/***		
04/28/2009	04/28/2009	05/4/2009		

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	05000275, 05000323
License:	DPR-80, DPR-82, SNM-2511
Report:	05000275/2009002 05000323/2009002
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:	January 1 through March 31, 2009
Inspectors:	M. Peck, Senior Resident Inspector M. Brown, Resident Inspector B. Henderson, Reactor Inspector G. George, Reactor Inspector M. Vasquez, Senior Health Physicist D. Stearns, Health Physicist C. Alldredge, NSPDP
Approved By:	V. G Gaddy, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000275/2009002, 05000323/2009002; 1/1/2009 – 3/31/2009; Diablo Canyon Power Plant, Integrated Resident and Regional Report; Refueling and Other Outage Activities and Access Control to Radiologically Significant Areas

The report covered a 3-month period of inspection by resident inspectors and announced baseline inspections by regional based inspectors. Two green noncited violations of very low safety 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 Technical Specification 5.4.1, "Procedures," after plant operators failed to stabilize reactor power and perform a comparison between the calorimetric heat balance calculation and the power range output prior to exceeding 30 percent power. The inspectors concluded several human performance factors contributed to the procedure violation, including less than adequate pre-job brief and poor operational command and control of the reactor power ascension.

  This finding is greater than minor because the failure to follow procedure is associated with the human performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone's 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 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," to analyze the significance of this finding. The inspectors concluded the finding is of low safety significance because the violation is not a design or qualification deficiency, did not represent a loss of a system safety function or risk significant equipment, and did not screen as potentially risk significant due to a seismic, flooding, or a severe weather initiating event. This finding has a crosscutting aspect in the area of human performance and the work practices component because the licensee failed to ensure adequate supervisory oversight of power ascension activities [H.4(c)]. (Section 1R20)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1 for failure to develop a procedure for removing the reactor head from the reactor pressure vessel and the subsequent filling of the reactor coolant system in a manner that would minimize the potential for airborne contamination. Specifically, on March 5, 2009, while lifting the reactor vessel head in preparation for reloading the reactor core, the licensee experienced airborne radioactivity as high as 4.8 derived air concentrations due to the delay in flooding the reactor refuel cavity. The delay allowed the radioactive contamination on the reactor upper internal structure to dry and subsequent air flow around the upper internal structure caused the contamination to become airborne. The licensee evacuated unnecessary personnel from the containment, initiated containment purge to reduce airborne contamination, and obtained air samples until airborne contamination levels were reduced to normal levels (less than 0.2 derived air concentrations). The licensee entered this item into the corrective actions program as Notification 50209442 and is conducting an apparent cause evaluation of the event.

The failure to develop and implement procedures for removing the reactor head and filling the reactor coolant system in a manner that minimized the potential for airborne radioactivity is a performance deficiency. The finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of the program and process and affected the cornerstone objective of exposure/contamination control in that failure to develop and implement adequate procedures for removing the reactor vessel head and fill the reactor coolant system resulted in workers' unplanned, unintended dose. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined this finding had very low safety significance because the finding involved as low as is reasonably achievable planning and work controls, and the licensee's 3-year rolling average collective dose is less than 135 person-rem per unit. Because the AMS-4 on the refuel floor in containment alarmed at an airborne concentration of greater than 0.5 derived air concentrations, the finding is self-revealing. Additionally, the finding had a crosscutting aspect in the area of human performance, work control component, because the licensee failed to plan and coordinate work activities by incorporating job site conditions which may impact radiological safety [H.3(a)]. (Section 2OS1)

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 are listed in Section 4OA7.

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, Diablo Canyon Units 1 and 2 were operating at full power. On January 25, 2009, the licensee shut down Unit 1 for refueling and steam generator replacement. On March 24, plant operators restarted Unit 1 and increased power to approximately 89 percent. On March 28, plant operators observed less than adequate flow from main feed pump 1-1 and reduced power to 55 percent and removed the feed pump from service. PG&E completed repairs to the feed pump and returned Unit 1 to full power on March 31. Both units remained at full power throughout the rest of the inspection period.

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

Since thunderstorms with high winds and heavy rains were forecast in the vicinity of the facility for the week of February 15, 2009, the inspectors reviewed the licensee's overall preparations and protection for the expected weather conditions. On February 17, 2009, the inspectors evaluated the licensee's preparations against the site procedures and determined that the actions taken were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report and performance requirements for 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.

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 of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Equipment Walk-downs

a. Inspection Scope

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

- Diesel Generator 2-1
- Auxiliary Feedwater Train 2-1
- Residual Heat Removal Train 2-1

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, Updated Final Safety Analysis Report, Technical Specification requirements, administrative Technical Specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in the attachment.

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

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 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 31, Unit 1 spent fuel pumps and heat exchanger rooms, February 10, 2009
- Fire Zone 3-R, Unit 1 spent fuel floor, February 10, 2009
- Fire Zones 1-A, 1 B and 1-C, Unit 1 containment, February 17, 2009
- Fire Zone 24-B, Unit 2 Bus G 4 kV Switchgear room, February 19, 2009

The inspectors reviewed areas to assess if licensee personnel had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant; effectively maintained fire detection and suppression capability; maintained passive fire protection features in good material condition; and had implemented adequate compensatory measures for out of service, degraded or inoperable fire protection equipment, systems, or features, in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to affect equipment that could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the attachment, 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. Specific documents reviewed during this inspection are listed in the attachment.

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

b. Findings

No findings of significance were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

On February 25, 2009, the inspectors observed a fire brigade activation for a simulated fire in the Unit 2 Turbine Building Ventilation Fan Room. The observation evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that the licensee staff identified deficiencies; openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient firefighting equipment brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other plant areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

These activities constitute completion of one annual fire-protection inspection sample as defined by Inspection Procedure 71111.05-05.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

02.01 Inspection Activities Other Than Steam Generator Tube Inspection, Pressurized Water Reactor Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The inspection procedure requires review of two or three types of nondestructive examination activities and, if performed, one to three welds on the reactor coolant system pressure boundary; and review one or two examinations with recordable indications that have been accepted by the licensee for continued service. Activities reviewed are listed below:

System	Identification	Exam Type	Result
Containment	Containment Liner	VT-Gen	No relevant indications
Reactor Pressure Vessel	CRDM #16, 18, 72	UT, ET	No relevant Indications

The inspectors reviewed records for the following nondestructive examinations:

System	Identification	Exam Type	Result
Reactor Pressure Vessel	CRDM #71	UT, ET	No relevant Indications
Reactor Coolant System	RPV Inlet Nozzle Safe-End Welds (4)	VT-2	No pressure boundary leakage identified
Reactor Coolant System	RPV Outlet Nozzle Safe-End Welds (4)	VT-2	No pressure boundary leakage identified
Chemical & Volume Control System	Charging Pump 1-1	VT, PT	Indications accepted for continued service
Main Steam	SG 1-1, Main Steam FW2	Radiography	No relevant Indications
Main Steam	SG 1-2, Main Steam FW1	Radiography	No relevant Indications
Main Steam	SG 1-2, Main Steam FW1R1	Radiography	No relevant Indications
Main Feedwater	SG 1-2, Feedwater FW9	Radiography	No relevant Indications
Reactor Coolant System	SG 1-2, RCS Hot Leg FW1	Radiography	No relevant Indications
Main Steam	SG 1-3, Main Steam FW1	Radiography	No relevant Indications
Main Feedwater	SG 1-3, Feedwater FW2	Radiography	No relevant Indications
Reactor Coolant System	SG 1-3,RCS Cold Leg FW2	Radiography	No relevant Indications
Reactor Coolant System	SG 1-3, RCS Hot Leg FW1	Radiography	No relevant Indications
Reactor Coolant System	SG 1-4, RCS Cold Leg FW2	Radiography	No relevant Indications
Reactor Coolant System	SG 1-4, RCS Hot Leg FW1	Radiography	No relevant Indications

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code requirements and applicable procedures. Indications were compared with previous examinations and dispositioned in accordance with American Society of Mechanical Engineers Code and approved procedures. The qualifications of all nondestructive examination technicians performing the inspections were verified to be current.

The inspectors reviewed nondestructive examination records of relevant indications on the inner pump casing of centrifugal charging pump 1-1. The augmented examination was a visual and dye penetrant examination of the inner surfaces of the stainless steel clad and carbon steel casing. The augmented examinations was a recommended practice from the pump vendor, Dresser Industries, to address operational experience with cracking of the stainless steel clad and exposure of the carbon steel casing.

Since 1995, the licensee has identified indications on the inner surface of the pump casings of centrifugal charging pump 1-1. The licensee has completed evaluations each refueling outage. Those evaluations determined that wall thinning has occurred but has not challenged the minimum wall thickness of the casing. The licensee plans to replace the casing with a stainless steel casing in Unit 1 Refuel Outage 16.

The following engineering evaluations were reviewed:

Number	Description
DN 50197238	Casing Indications on CCP1-1

The following calculations were reviewed:

Number	Title	Revision/Date
N-195	Verification of various wall thicknesses and structural integrity of the pump casing	August 25, 1995

Seven examples of welding on the reactor coolant system pressure boundary during steam generator replacement activities were examined through direct observation and/or record review as follows:

System	Component/Weld Identification
Reactor Coolant System	Steam Generator 1-1 Hot Leg
Reactor Coolant System	Steam Generator 1-1 Cold Leg
Reactor Coolant System	Steam Generator 1-2 Hot Leg
Reactor Coolant System	Steam Generator 1-2 Cold Leg

System	Component/Weld Identification
Reactor Coolant System	Steam Generator 1-3 Cold Leg
Main Steam System	Steam Generator 1-3 Main Steam Line
Main Steam System	Steam Generator 1-4 Main Steam Line

Welding procedures and nondestructive examination of the welding repair conformed to American Society of Mechanical Engineers Code requirements and licensee requirements.

The inspectors verified, by review, that the welding procedure specifications and the welders had been properly qualified in accordance with American Society of Mechanical Engineers Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the gas tungsten arc welding process (machine and manual) and the shielded metal arc welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications.

The inspectors completed one sample under Section 02.01.

b. Findings

No findings of significance were identified.

02.02 Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

The licensee performed bare metal visual inspection of all control rod drive mechanism penetrations using a robotic video system. The inspectors reviewed video taped records of this inspection for Penetrations 11, 23, 37, 43, 54, 58, 68, 70, 74, and 75. No evidence of boric acid leakage was observed.

The licensee performed nondestructive examination of 100 percent of reactor vessel upper head penetrations. The inspectors directly observed a sample consisting of the examinations listed below:

System	Component ID	Examination Method	Result
Vessel Upper Head Penetration	CRDM #16,18,72	UT, ET	No Relevant Indications

The inspectors, using stored electronic data, reviewed the following examinations in which indications were observed, evaluated and determined not to be relevant indications in accordance with American Society of Mechanical Engineers Code:

System	Component ID	Examination Method	Result
Reactor Pressure Vessel	CRDM 71	UT,ET	No Relevant Indications

The nondestructive examination inspections were performed in accordance with the requirements of NRC Order EA-03-009. Qualifications of nondestructive examination personnel were reviewed and verified to be current.

The inspectors completed one sample under Section 02.02.

b. Findings

No findings of significance were identified.

02.03 Boric Acid Corrosion Control Inspection Activities

a. Inspection Scope

The inspectors observed a sample of boric acid corrosion control activities and verified that visual inspections emphasized locations where boric acid leaks can cause degradation of safety significant components.

Also, the inspectors performed an independent walkdown of the residual heat removal system from the reactor water storage tank to the containment penetrations. From this walkdown, the inspectors determined that the licensee properly identified leakage and entered it into the boric acid control program.

The inspectors reviewed 11 boric acid leakage evaluations. These evaluations emphasized excessive boric acid leakage in the chemical volume and control, residual heat removal, and safety injection systems.

Number	Description
DN 50112587	Boric Acid Leak at RHR-1-8724B
DN 50112736	Boric Acid Leak at RHR-1-8726A
DN 50205657	RHR-1-8715A Dry Boric Acid Spot Packing
DN 50114850	SI-1-8925 Slight Boric Acid Body-Bonnet
DN 50035713	CVCS-1-547 (U-1 Blender Room) Leak CREAT
DN 50041704	Line 154; BA at Portable PP Blind Flange Co
DN 50041881	Boric Acid Accumulation at Insulation Joint
DN 50039338	PI-142F: Boric Acid Leakage From Gauge F
DN 50041712	FE-1440 Flange Has Boric Acid Accumulation
DN 50041622	SI-1-8918A Boric Acid Leak on Packing
A0660831	LDHE 1-1: Boric Acid on Channel Head Flange

The inspectors reviewed one corrective action performed for evidence of boric acid leakage on heat exchanger DC-1-08-M-HX-LDHE1, the letdown heat exchanger in the chemical and volume control system.

The condition of all the components was appropriately inspected, evaluated and entered into the licensee's corrective action program. Corrective actions taken were consistent with American Society of Mechanical Engineers code requirements.

The inspectors completed one sample under Section 02.03.

02.04 Steam Generator Tube Inspection Activities

a. Inspection Scope

Unit 1 steam generators were replaced during this outage and steam generator tubes were not inspected.

The inspectors reviewed baseline eddy current inspections of the new steam generator tubes that were performed at the manufacturer's facility. Only minor indications were identified and no tubes were plugged.

b. Findings

No findings of significance were identified.

02.05 Identification and Resolution of Problems

a. Inspection Scope

The inspection procedure requires review of a sample of problems associated with inservice inspections documented by the licensee in the corrective actions program for appropriateness of the corrective actions.

The inspectors reviewed 23 corrective action reports which dealt with inservice inspection activities and found the corrective actions were appropriate. Corrective action documents reviewed during this inspection are listed in the attachment. From this review, the inspectors concluded that the licensee has an appropriate threshold for entering issues into the corrective actions program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On March 18, 2009, 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 of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Unit 1, 4 kV Busses F, G, and H
- Units 1 and 2, Motor-driven auxiliary feedwater level control valves
- Units 1 and 2, Fire protection

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

- Implementing appropriate work practices
- Identifying and addressing common cause failures
- Scoping of systems in accordance with 10 CFR 50.65(b)
- Characterizing system reliability issues for performance
- Charging unavailability for performance
- Trending key parameters for condition monitoring
- Ensuring proper classification in accordance with 10 CFR 50.65(a)(1) or (a)(2)

- Verifying appropriate performance criteria for structures, systems, and components classified as having an adequate demonstration of performance through preventive maintenance, as described in 10 CFR 50.65(a)(2), or as requiring the establishment of appropriate and adequate goals and corrective actions for systems classified as not having adequate performance, as described in 10 CFR 50.65(a)(1)

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

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- TS Sheet 1-TS-09-0001, Unit 1, 230 kV offsite power, January 6, 2009

- TS Sheet 2-TS-09-0001, Unit 2, Startup Bus, January 7, 2009

- Actions taken to minimize adverse impact on Unit 2 and common systems during Unit 1 steam generator replacement, January 21, 2009

- Mid-Loop Operations, Unit 1, March 13 and 14, 2009

- Unit 1 Containment Integrated Leak Rate Test, March 16, 2009

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

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Notification 50197090, Unit 2, Failure of Control Rod P-08 to withdraw

- Notification 50086062, New line of earthquake epicenters discovered offshore, updated February 17, 2009

- Notification 50184499, Unit 2, Containment Structure Sump 2-2 reads 0% Level, March 10, 2009

- Notification 50214618, Unit 1, Steam Generator 1-3 inadequate load bearing surfaces, March 22, 2009

- Notification 50214299, Unit 1, Decrease in pressurizer loop seal temperature, March 22, 2009

- Degraded 230 kV offsite power, March 25, 2009

- Notification 50227312, Unit 1, Diesel Generator 1-1 disconnected turbo air start solenoid junction box for, March 30, 2009

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

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

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the two temporary modifications and associated safety evaluation screenings, against system design bases documentation, including the Updated Final Safety Analysis Report and the Technical Specifications, and verified that the modifications did not adversely affect the system operability/availability. The inspectors also verified that the installations and restorations were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modifications were identified on control room drawings, appropriate tags were placed on the affected equipment, and licensee personnel evaluated the combined effects on mitigating systems and the integrity of radiological barriers.

The inspectors reviewed the following temporary and permanent modifications to verify that the safety functions of important safety systems were not degraded:

- Unit 1, Steam generator replacement project temporary power inside containment, January 12, 2009

- Unit 1, Steam generator replacement project rigging and handling, January 21, 2009

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 modification listed below. 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; postmodification 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 appropriateness of modification design assumptions, and 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.

The inspectors reviewed the following permanent modification to verify that the safety functions of important safety systems were not degraded:

- Unit 1, Steam generator replacement project permanent modifications, January 26, 2009

These activities constitute completion of two samples for temporary plant modifications and one sample for permanent plant modifications as defined in Inspection Procedure 71111.18-05

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

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

- Postmaintenance Test 64006711-0150, Safety Injection Valves SI-8807A and B, February 9, 2009

- Postmaintenance Test 64003459-5001, Diesel Generator 1-3 Air Start System, February 8, 2009

- Postmaintenance Test 64002867, Unit 1, Source Range Monitor N-31 card and power supply replacement, March 21, 2009

- Postmaintenance Test 64004002, Containment fan cooling refurbishment, February 3, 2009

- Postmaintenance Tests 60013440, 64019379 & 64004155, Diesel Generator 1-2 overhaul, March 11, 2009

- Postmaintenance Test 600113627, Corrective maintenance of Containment Fan Cooler 1-5, March 19, 2009

The inspectors selected these activities based upon the structure, system, or component's ability to affect risk. The inspectors evaluated these activities for the following (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 Updated Final Safety Analysis Report, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with postmaintenance tests to determine whether the licensee was identifying problems and entering them in the corrective action program and that the problems were being corrected commensurate with their importance to safety. Specific documents reviewed during this inspection are listed in the attachment.

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

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan and Contingency Plans for the Unit 1 refueling outage, conducted from January 25 to March 25, 2009, 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, is 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 the drywell (primary 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 as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction. The inspectors identified a noncited violation of Technical Specification 5.4.1, "Procedures," after plant operators failed to stabilize reactor power and perform a comparison between the calorimetric heat balance calculation and the power range channel prior to exceeding 30 percent power.

Description. On March 24, 2009, during the Unit 1 startup following refueling, plant operators failed to suspend power accession prior to exceeding 30 percent power. Technical Specification Surveillance Requirement 3.3.1.2 required the licensee to compare the calorimetric heat balance to the power range channel output prior to exceeding 30 percent rated thermal power. This requirement was translated in Operating Procedures OP L-3, "Secondary Plant Startup," Step 6.4.8, and OP L-4, "Normal Operation at Power," Step 6.1.3.i.2. Plant operators increased reactor thermal power to 38 percent without performing the surveillance. The problem was identified after the shift outage manager observed reactor power on the plant data network and contacted the control room. Operators reduced power to below 30 percent and completed the surveillance.

The inspectors added value by identifying several human performance deficiencies that contributed to the procedure violation after independently interviewing plant personnel. The pre-job brief and simulator training were less than adequate. The training and brief focused on paralleling the main generator and did not adequately address the subsequent power ascension and power limits. Some of the operators responsible for power ascension did not attend the simulator training. Operational command and control of the reactor was weak. Two shift foremen directed operator action during the evolution. The shift foremen focused on the generator electrical output rather than the reactor thermal power. The foremen directed the operator to stabilize the plant at 331 Megawatts electrical output rather than below 30 percent reactor thermal power. The plant reached 38 percent power when the load increase stopped at the target of 331 Megawatts electrical. During the power ascension, the reactivity supervisor raised a concern that the power ramp would exceed the 30 percent power limit. However, the shift foremen inadequately evaluated the situation and continued with the power ascension.

Analysis. The inspectors concluded that the failure of plant operators to follow power ascension procedures was a performance deficiency. This finding is greater than minor because the failure to follow procedure was associated with the human performance attribute of the Mitigating Systems Cornerstone and affected the cornerstone's 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 0609, Attachment 4, "Phase 1 - Initial Screening and Characterization of Findings," to analyze the significance of this finding. The inspectors concluded the finding is of low safety significance because the condition was not a design or qualification deficiency, did not represent a loss of a system safety function, and did not screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of human performance, associated with the work practices component, because the licensee failed to ensure adequate supervisory oversight of power ascension activities to ensure the reactor power limit was not exceeded [H.4 (c)].

Enforcement. Technical Specification 5.4.1 required PG&E implement the written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 included general plant operating procedures.

General Plant Operating Procedures OP L-3 and OP L-4 required the licensee to compare results of the calorimetric heat balance calculation to the power range channel, and adjust the power range channel output if needed before exceeding 30 percent rated thermal power. Contrary to this, on March 24, 2009, PG&E exceeded 30 percent rated thermal power without comparing the results of the calorimetric heat balance calculation to power range channel output. Because this finding is of very low safety significance and was entered into the corrective actions program as Notification 50216014, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000275/2009002-01, Failure to Follow Power Ascension Procedures.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, procedure requirements, and Technical Specifications to ensure that the eleven 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 American Society of Mechanical Engineers 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 set points

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

- December 18, 2008, Unit 1, Routine Surveillance of Diesel Generator 1-2
- January 22, 2009, Unit 1, Containment isolation valve local leak-rate test of Penetration 56
- February 3, 2009, Unit 2, Inservice test of Safety Injection Pump 2-2
- February 8, 2009, Unit 1, Containment isolation valve local leak-rate test of Penetration 20
- February 9, 2009, Unit 1, Inservice test of Containment Isolation Valve FCV-678
- February 9, 2009, Unit 1, Containment isolation valve local leak-rate test of Penetration 19
- February 10, 2009, Unit 2, Operability verification testing of Diesel Generators 2-1, 2-2, and 2-3
- March 11, 2009, Unit 1, Routine integrated test of engineering safeguards and diesel generators
- March 19, 2009, Unit 1, Reactor coolant system water inventory balance as part of Unit 1 steam generator replacement inspection activities
- March 21, 2009, Unit 1, Reactor coolant system leakage test as part of Unit 1 steam generator replacement inspection activities
- March 23, 2009, Unit 1, Shutdown Margin Determination

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

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

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess licensee personnel's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. 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. During the inspection, the inspectors interviewed the Radiation Protection Manager, radiation protection supervisors, and radiation workers. The inspectors performed independent radiation dose rate measurements and reviewed the following items:

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

- Controls (surveys, posting, and barricades) of radiation, high radiation, or airborne radioactivity areas

- Radiation work permits, procedures, engineering controls, and air sampler locations

- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personnel dosimeter noticeably malfunctions or alarms

- Barrier integrity and performance of engineering controls in airborne radioactivity areas

- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools

- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection

- Corrective action documents related to access controls

- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies

- Radiation work permit briefings and worker instructions

- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance

- Dosimetry placement in high radiation work areas with significant dose rate gradients

- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

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

Either because the conditions did not exist or an event had not occurred, no opportunities were available to review the following items:

- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent

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

These activities constitute completion of 20 of the required 21 samples as defined in Inspection Procedure 71121.01-05.

b. Findings

Introduction. The inspectors reviewed a Green, self-revealing noncited violation of Technical Specification 5.4.1 for failure to develop a procedure for removing the reactor head from the reactor pressure vessel and the subsequent filling of the reactor coolant system in a manner that would minimize the potential for airborne contamination.

Description. At approximately 11:50 p.m. on March 4, 2009, with the Unit 1 reactor defueled, containment purge in progress, and the containment hatch open; in preparation for reloading fuel into the reactor vessel, the licensee commenced lifting the reactor vessel head out of the reactor cavity. Filling of the reactor coolant system began almost 40 minutes later at 12:29 a.m., using gravity fill from the refueling water storage tank. At this time, the head was being placed on its stand on the refueling floor of containment (140-foot level). The time lapse between lifting the head and commencing fill of the reactor coolant system resulted in drying of the upper reactor vessel internals. The water filling the reactor coolant system displaced the air in the reactor cavity and provided a motive force for contamination on the upper reactor vessel internals to become airborne.

As filling of the reactor coolant system continued, at about 1:23 a.m., a continuous airborne radioactivity monitor, AMS-4, on the 140-foot elevation of containment indicated increasing levels of airborne contamination and alarmed when airborne concentrations exceeded 0.5 derived air concentrations. At approximately 1:30 a.m., the radiation protection foreman ordered the containment equipment hatch to be closed. At 2:05 a.m., the licensee aligned the residual heat removal pump to continue filling the reactor coolant system. This provided increased flow rates as compared with the gravity flow from the refueling water storage tank causing additional increases in airborne contamination from the vessel upper internals as indicated by the AMS-4 continuous airborne monitor. The licensee evacuated all non-essential personnel from containment at 2:30 a.m. Eleven radiation protection and decontamination personnel remained in containment taking airborne activity samples and performing decontamination activities. However, at about 3:00 a.m., airborne readings from the AMS-4 increased to 4.8 derived air concentrations resulting in the decision to evacuate all personnel from containment.

At 3:45 a.m., water in the reactor cavity was at the 136.5-foot level sufficiently covering reactor vessel internals. The residual heat removal system was placed on recirculation while waiting for containment access, and residual heat removal pump 1-1 was eventually secured. Containment was purged under discharge permit 2009-1-24, and at approximately 5:30 a.m., containment airborne activity samples verified that airborne activity on each floor had returned to normal (less than 0.2 derived air concentrations).

Diablo Canyon had experienced elevated airborne contamination levels during previous reactor head lifts, however airborne contamination levels were higher than expected this time. Licensee representatives stated that normally the plant raises water level in the reactor vessel as the head is being lifted in order to minimize drying on the upper vessel internals. Even though this was the normal practice, the inspectors determined it was not required by the licensee's procedures. The operations department used Procedure TP TO-0824, "Core Offload Window Systems Restoration during SGRP," Revision 1, for this activity and the maintenance department used Procedure MP M-7.1A, "Reactor Vessel Closure Head Removal," Revision 9. However, the procedures did not contain sufficient precautions and coordination to ensure the filling of the reactor cavity was performed in conjunction with raising the reactor head, thus minimizing drying of the vessel upper internals. Procedure TP TO-0824, "Core Offload Window Systems Restoration During SGRP," did not contain instructions that would have limited the time that upper reactor vessel internals would have dried, providing a source of removable contamination that went airborne as the reactor vessel was filled with water. Procedure, MP M-7.1A, required mechanical maintenance to notify the shift foreman when the reactor head was on the head stand and the cavity was clear and ready for flooding. This conflicted with the licensee's intent that the reactor cavity be filled as the head is being lifted in order to minimize the dry time of the upper vessel internals.

Analysis. The failure to develop and implement procedures for removing the reactor head and filling the reactor coolant system in a manner that minimized the potential for airborne radioactivity is a performance deficiency. This finding is greater than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of program and process and affected the cornerstone objective of exposure/contamination control in that failure to develop and implement procedures for removing the reactor vessel head and fill the reactor coolant system resulted in workers' unplanned, unintended dose. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined this finding had very low safety significance because the finding involved as low as is reasonably achievable planning and work controls, and the licensee's 3-year rolling average collective dose is less than 135 person-rem per unit. Because the AMS-4 on the refuel floor in containment alarmed at an airborne concentration of greater than 0.5 derived air concentrations, the finding is self-revealing. The cause of this finding was related to the human performance component of work control in that the licensee failed to plan and coordinate work activities by incorporating job site conditions which would impact radiological safety. [H.3(a)]

Enforcement. Technical Specifications 5.4.1 states, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 7 requires procedures for the control of radioactivity including contamination controls. Section 9 of Regulatory Guide 1.33 also requires that procedures that could be categorized either as maintenance or operating procedures shall be developed for removing the reactor head and for refilling the reactor vessel. Contrary to the above, on March 5, 2009, the licensee failed to establish and implement procedures for removal of the reactor vessel head and refill of the refueling cavity that would minimize the potential for and magnitude of airborne radioactive contamination. Specifically, Procedure TP TO-0824, "Core Offload Window Systems Restoration During SGRP," Revision 1, and Procedure MP M-7.1A, "Reactor Vessel Closure Head Removal." Revision 9, did not contain instructions that would have coordinated work activities between the operations and maintenance department and limited the time that upper reactor vessel internals would be exposed. The drying upper internal structure provided a source of removable contamination that became airborne as the reactor vessel was filled with water. Because the failure to establish adequate procedures for this evolution is of very low safety significance and has been entered into the licensee's corrective action program as Notification 50209442, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV05000275/2009002-02, Inadequate Procedure.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors assessed licensee personnel's performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable. The inspectors used the requirements in 10 CFR Part 20 and the licensee's procedures required by Technical Specifications as criteria for determining compliance. The inspectors interviewed licensee personnel and reviewed the following:

- Current 3-year rolling average collective exposure
- Five outage work activities scheduled during the inspection period and associated work activity exposure estimates which were likely to result in the highest personnel collective exposures
- Site-specific ALARA procedures
- Method for adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered
- Exposure tracking system
- Workers' use of the low dose waiting areas
- Exposures of individuals from selected work groups
- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas
- Corrective action documents related to the ALARA program and follow-up activities, such as initial problem identification, characterization, and tracking

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

These activities constitute completion of 5 of the required 15 samples and 4 of the optional samples as defined in Inspection Procedure 71121.02-05.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the fourth quarter 2008 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, “Performance Indicator Program.”

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

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator for Units 1 and 2 for the period from the first quarter 2008 through fourth quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of January 1, 2008, through December 31, 2008, 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.

These activities constitute completion of two unplanned scrams per 7000 critical hour’s samples as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.3 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications performance indicator for Units 1 and 2 for the period from the first quarter through fourth quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 5, was used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports and NRC Inspection reports for the period from the first quarter through fourth quarter 2008 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.

These activities constitute completion of two unplanned scrams with complications samples as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.4 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Power Changes per 7000 Critical Hours performance indicator for Units 1 and 2 for the period from the first quarter through fourth quarter 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC inspection reports for the period from January 1, 2008 to December 31, 2008 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.

These activities constitute completion of two unplanned power changes per 7000 critical hours samples as defined by Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

.5 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences performance indicator for the fourth quarter of 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's assessment of the performance indicator for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's performance indicator data collection and analyses, the inspectors discussed with radiation protection staff, the scope and breadth of its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas.

These activities constitute completion of the occupational radiological occurrences sample as defined in Inspection Procedure 71151-05.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual Radiological Effluent Occurrences performance indicator for the fourth quarter of 2008. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates during the fourth quarter of 2008 to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Additionally, the inspectors reviewed the licensee's historical 10 CFR 50.75(g) file and selectively reviewed the licensee's analysis for discharge pathways resulting from a spill, leak, or unexpected liquid discharge focusing on those incidents which occurred over the last few years.

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 of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up Inspection

a. Inspection Scope

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

- Notification 50197238, Corrosion of Centrifugal Charging Pump CCP-1-1 pump casing, January 30, 2009

- Identification and resolution of issues associated with the Unit 1 steam generator replacement project, March 25, 2009

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

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 (Discussed) NRC - Characterization of the Shoreline Fault and Evaluation of the Potential Impact on Plant Systems

On November 21, 2008, PG&E notified the NRC of the discovery of a geologic feature that may represent a new earthquake fault (Event Notification 44675). This feature is located about 0.6 miles offshore from the Diablo Canyon Power Plant. The licensee named this new geologic feature the Shoreline Fault. The licensee estimated the ground motion spectrum (acceleration and frequency) that could be generated by a Shoreline Fault earthquake. The licensee concluded that the postulated spectrum was bounded by the ground motion previously analyzed as part of the plant seismic design and licensing basis. PG&E subsequently developed an action plan to fully characterize the Shoreline Fault. This action plan and schedule has been entered into ADAMS as ML083540266, ML090720505, ML090720516, and ML083540261.

In addition to ground motion, PG&E estimated that the Shoreline Fault could potentially generate up to 2 inches of secondary ground deformation at the Diablo Canyon facility. This secondary ground deformation could adversely affect ultimate heat sink (auxiliary salt water) piping buried in the shale, claystone, and siltstone strata located between the power block and inlet structure. Seismic induced secondary ground deformations have not been previously analyzed as part of the Diablo Canyon design basis. To evaluate the qualification and operability of the buried piping, the licensee is examining evidence of past ground deformation in exposed cliff faces near the plant. PG&E will input this information to an analytical model to predict the range of potential ground movement and motion and evaluate the structural capacity of the buried piping. This analysis will address the upper and lower bounding soil and pipe effects, including induced stress and capacity and stiffness of buried pipe flanges. The study will also consider the extent of ovalization under postulated conditions and develop nonlinear load-deformation capacity relationships for Category II concrete pipe collars and concrete vault walls that may be overloaded due to the imposed ground deformation.

.2 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.3 Temporary Instruction 2515/172, "Reactor Coolant system Dissimilar Metal Butt Welds"

Temporary Instruction TI 2515/172 was performed at Diablo Canyon during Refueling Outage 1R15 in February and March 2009.

03.01 Licensee's Implementation of the MRP-139 Baseline Inspections

All baseline inspections were completed and inspected during earlier outages. Unit 1 pressurizer does not have dissimilar metal welds. There are eight unmitigated, dissimilar metal butt welds on the reactor coolant system hot and cold leg nozzles.

03.02 Volumetric Examinations

a. No volumetric examinations were required or performed under the MRP-139 program during this outage. The inspectors reviewed the records of visual inspections of all eight unmitigated dissimilar metal butt welds.

   No relevant conditions or deficiencies were identified during the visual examinations of the hot and cold leg unmitigated dissimilar metal butt welds.

b. This item is not applicable because there are no weld overlays in Unit 1.

c. The visual examinations were performed by qualified personnel.

d. No deficiencies were identified.

03.03 Weld Overlays

This item is not applicable because there are no weld overlays in Unit 1.

03.04 Mechanical Stress Improvement

This item is not applicable because mechanical stress improvement was not employed at Unit 1.

03.05 Inservice inspection program

The licensee's MRP-139 inspections have been managed through the Action Request process to assure that inspection requirements of MRP-139 are completed as scheduled. The licensee provided assurance that MRP-139 inspections will be included in the plant's inservice inspection program in a timely fashion.

The inspectors' review determined that the hot leg and cold leg dissimilar metal butt welds are appropriately categorized in accordance with MRP-139 requirements.

4OA6 Meetings

Exit Meeting Summary

On March 12, 2009, the inspectors presented the results of this Inservice Inspection to Mr. J. Becker, Site Vice President, and other members of licensee management. Licensee management acknowledged the inspection findings. The inspectors returned proprietary material examined during the inspection.

On March 12, 2009, the inspectors presented the Occupational and Public Radiation Safety 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.

On April 2, 2009, the inspectors presented the inspection results to Mr. J. Becker, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following 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.

- Title 10 CFR 50, Appendix B, Criteria X, “Inspection,” required PG&E to perform examinations or measurements where necessary to assure quality. Contrary to this, Work Package 1-3055C, “Reinstall Lower Supports 1-3,” completed on March 16, 2009, did not include an examination of the gap between the seismic mounting plates and the load bearing surfaces for the Unit 1 replacement steam generators. As a result, Steam Generator 1-3 was placed in service on March 20, 2009 in an unanalyzed condition due to excessive gaps between two seismic mounting plates and corresponding support columns. On March 22, 2009, after establishing the reactor coolant system at normal operating temperature and pressure, PG&E identified the excessive gaps after a temporary worker raised the concern during an exit interview. The licensee declared the reactor coolant system inoperable and applied the provisions of Technical Specification 3.0.3. PG&E took corrective actions to repair seismic mounting plates. The licensee entered this condition into the correction action program as Notification 50214618, “SG 1-3 Column Bearing Surface Issue,” This finding is of very low safety significance because the condition did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.
- Technical Specification 5.4.1.a, “Procedures,” required that PG&E implement written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33 includes procedures for draining the reactor coolant system. Contrary to this, on March 13, 2009, PG&E failed to properly align the mid-loop level monitoring system in accordance with Procedure MP I-2.28, “Activation of the Reactor Vessel Refueling Level Indication System,” prior to start of reactor drain down for mid-loop operations. During the reactor coolant system drain down, a maintenance technician identified that the narrow range level transmitter isolation valve was closed when required to be opened. PG&E stopped the drain down operations and performed a system walk down. During the walk down, PG&E also identified that the flex hose used for the wide range level instrument vent was not connected as required. A maintenance technician and independent verifier had signed off both procedure steps as completed. The inspectors concluded that less than adequate pre-job brief, failure to maintain the procedures in-hand, inadequate use at place keeping and peer checking during the system alignment contributed to this violation. PG&E entered this issue into the corrective action program as Notification 50212379. This finding is of very low safety significance because PG&E did not lose all reactor vessel level indications during mid-loop operations.

- Technical Specification 5.7.2.b states, in part, that each entryway to an area with dose rates greater than 1 rem/hour at 30 centimeters from the source shall be conspicuously posted as a high radiation area. Access to and activities in such area shall be controlled by means of a radiation work permit or equivalent that includes specification of radiation dose rates in the immediate work area(s). Contrary to the above, at approximately 7:00 a.m. on March 12, 2009, an individual inadvertently crossed the boundary for the locked high radiation area at the entrance to the stairway to the cavity. The individual was not signed in on a radiation work permit that allowed access to locked high radiation areas; and therefore, did not get a briefing on the dose rates in the area. The violation was identified by the radiation protection technician who immediately informed the individual to exit the area. This issue has been documented as Notification 50211054. The finding was determined to be of very low safety significance because it did not involve as low as is reasonably achievable planning and controls, did not involve an overexposure, did not have a substantial potential for overexposure, and did not result in an impaired ability to assess dose.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

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

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000275/2009002-01	NCV	Failure to Follow Power Ascension Procedures (Section 1R20)
05000275/2009002-02	NCV	Inadequate Procedure (Section 2OS1)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
CP M-12	Stranded Plant	4
AD8.DC55, Attachment 7.11	Outage Safety Schedule Change Evaluation Form	2/15/2009

Section 1R04: Equipment Alignment

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
OP J-6B	Diesel Generators	
STP M-9A	Diesel Engine Generator Routine Surveillance Test	

Section 1R05: Fire Protection

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
Drawing 111906, Sheet 26	Unit 1 Containment Building 91”	1
Drawing 111906, Sheet 27	Unit 1 Containment Building 117’	1
Drawing 111906, Sheet 28	Unit 1 Containment Building 140’	1
CP M-6	Fire	31

Section 1R08: Inservice Inspection Activities

PROCEDURES

NUMBER	TITLE	REVISION / DATE
MRS-SSP-2284-PGE	Eddy Current Inspection of Pre-Service Heat Exchanger Tubing for Diablo Canyon Unit 1 PSI	0
MRS-SSP-2092-PGE/PEG	Pegasys Operating Procedure for Diablo Canyon (Units 1 & 2) Pre-Service Inspection	1
MRS-SSP-2293-PGE	Steam Generator Eddy Current Data Analysis Procedure for Pre-Service Inspection of Diablo Canyon Unit 1	0
TS1.ID9	Reactor Coolant System Alloy 600 Program	0
GT/8.8-1	ASME Section III, P-1 to P-8, manual GTAW process where PWHT and impact testing are not required	0
GTM/8.8-1	ASME Section III, P-8 to P-8, machine GTAW process where PWHT and impact testing are not required	0
ER1.ID2	Boric Acid Corrosion Control Program	3
AD4-ID2	Plant Leakage Evaluation	9
AWP SP-003	Oversight and Alignment of SGT CAP with the DCPP CAP	0
AD4.ID2	Plant Leakage Evaluation	9
ER1.ID2	Boric Acid Corrosion Control Program	3

NUMBER	TITLE	REVISION / DATE
QEP 20.03	ASME General Welding Requirements	1
QEP 20.01	Documentation and Control of Welding	4
QEP 20.02	Welding Procedure Specifications	2
QEP 20.04	Welder Performance Qualification	0
QEP 20.05	Welding Material Control	1
QEP 20.06	Preheat and Postweld Heat Treatment	0
QEP 20.07	Weld and Base Material Welds	0
QEP 20.08	AWS General Welding Requirements	3

CALCULATIONS

NUMBER	TITLE	REVISION / DATE
N-195	Verification of various wall thicknesses and structural integrity of the pump casing	8/25/1995

CORRECTIVE ACTION DOCUMENTS

A0660831	A0728934	A0735886	50039756	50205657
A0699061	A0728937	A0737609	50040061	50202747
A0721736	A0729037	A0740713	50042470	
A0728551	A0730004	50038147	50042807	
A0728932	A0737163	50038306	50077660	
A0728933	A0737310	50038698	50198261	

WELDING PROCEDURE SPECIFICATIONS AND THEIR SUPPORTING PROCEDURE QUALIFICATION RECORDS

WPS	TITLE	PQRS	REVISION
GT-SM/1.1-1	Manual Gas Tungsten Arc Welding/Manual Shielded Metal Arc Welding	1/17/2007	0
GT-SM/1.1-2	Manual Gas Tungsten Arc Welding/Manual Shielded Metal Arc Welding	1/17/2007	0
GT/8.8-1	ASME Section III, P-1 to P-8, manual GTAW process where PWHT and impact testing are not required	1/18/2007	0
GTM/8.8-1	ASME Section III, P-8 to P-8, machine GTAW process where PWHT and impact testing are not required	1/18/2007	0

MISCELLANEOUS

NUMBER	TITLE	REVISION / DATE
PG&E Letter DCL-09-002	Technical Justification for Deviation from EPRI MRP Bare Metal Visual Examination Schedule – NEI 03-08 Mandatory Work Product Element – For Information Only	0
PG&E Letter DCL-08-103	ASME Inservice Inspection Program Relief Request NDE-Leak Path for the Unit 1, Fifteenth Refueling Outage, Third Ten-Year Inservice Inspection Interval to Allow Use of the Rules of the NRC First Revised Order, EA-03-009 for Performance of Volumetric Leak Path Assessment of Reactor Head Penetration Nozzles	0
N/A	DCPP Alloy 600/82/182 Component Location and Strategy Table	0

Section 1R11: Licensed Operator Requalification Program

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
Lesson R086JIT6	1R15 Startup Training	3/18/2009
OP L-2	Hot Standby to Startup Mode	38

Section 1R12: Maintenance Effectiveness

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
TS5.ID1	System Engineering Program	14
OM7.ID1	Problem Identification and Resolution	30
OM4.ID17	Project review Committee	5A
OM4.ID16	Plant Health Committee	2

CORRECTIVE ACTION DOCUMENTS

NOTIFICATIONS

50033892	50037631	50086341	600003091	A0661486
50035983	50039729	50086342	A0616554	A0731222

Section 1R13: Maintenance Risk Assessment and Emergent Work Controls

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
50041476	Yellow/orange arcing on all 9 insulators on the high voltage side SU XFMR 1-1	9/25/2008
TP 22.0	Unit 1 & 2 SGRP Risk Management Task Plan	0

AD8.DC50	Outage Safety Management	1
AD8.DC54	Containment Closure	12

Section 1R15: Operability Evaluations

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
50184499	Containment structure 2-2 LI-61 drifting down	1/16/2009
50199435	2LR-60 Containment sump LR ribbon/pen mismatch	2/6/2009
Drawing 106719, Sheet 3	Containment Structure Sumps	98
OP A-4A:IV	Draining of Pressurizer Safety Valve Loop Seals	5
OP J-2:VIII	Guidelines for Reliable Transmission Service for DCPP	14
ML083540266	Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Shoreline Fault Study ACTION PLAN FOR THE STUDY OF THE SHORELINE FAULT	12/12/2008
ML090720505	Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Email from Pacific Gas and Electric, Seismic Action Plan, Email Encloses Shoreline Fault Slides (ME0174 and ME0175)	3/20/2009
ML090720516	Shoreline Fault Characterization Action plan for Shoreline fault study.doc	3/20/2009
ML083540261	Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Shoreline Fault Characterization Schedule	12/17/2008

Section 1R18: Plant Modifications

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
DCP E-050768	Unit 2 Temporary Power Inside Containment	3/29/2008
DCP E-049769	Unit 1 Temporary Power Inside Containment	8/14/2008
DCP C-049744	SGRP – Unit 1 Rigging and Handling Inside Containment	8/27/2008
DCP C-049746	SGRP – Unit 1 Rigging and Handling Outside Containment	8/28/2008
DCP C-049751	SGRP – Unit 1 Structural Interferences	9/10/2008
DCP J-049832	SGRP – Unit 1 Revise Eagle 21 Setpoints and Documentation	1/15/2008
DCP J-049834	SGRP – Unit 1 Revise DFWCS Setpoints and Documentation	1/14/2008
DCP M-049790	SGRP – Unit 1 Licensing Basis and Design Basis Evaluation for RSG Operation	7/24/2008
DCP M-049908	SGRP – Unit 1 GE/GW Ventilation System Operation	11/16/2007
DCP P-049763	SGRP – Unit 1 Provide OSG and RSG Supports	9/11/2008
38241-CLP-01	Critical Lift Evaluation Form	3
QEP 10.05	Rigging and Handling	3

Section 1R19: Postmaintenance Testing

CORRECTIVE ACTION DOCUMENTS

NOTIFICATIONS

50047982	50211723	50212047

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
STP V-3L3	Exercising Valves SI-8807A and SI-8807B Safety Injection Charging Pump Suction Crosstie	13
STP V-302	Diesel Starting Air receiver Leak Check and Check valve Exercising	12
PTLR	Pressure temperature limits Curve for Diablo Canyon	9
STP 1-370N31.A	Source Range N31 Channel Operational Test	1
STP 1-4B4	Determination of Source Range Detector characteristic Curves	17
STP R-28	High Flux at Shutdown Alarm trip and Reset Set points	62
MA1.DC10	Troubleshooting	10
STP I-37-N31.B	Source Range N31 Channel Calibration	1
Work Order 60013655	Unit 1 N31 Indication Change During ILRT	0
STP M-92A	Refueling Interval Surveillance – Containment Fan Cooler System	20
STP M-9A	Diesel Engine Generator Routine Surveillance Test	78
STP M-9X	Diesel Generator Operability Verification	21
STP M-9D1	Diesel Generator Full Load Rejection Test	13
STP V-302	Diesel Generator Starting Air Receiver Leak Check and Check Valve Exercising	12

Section 1R20: Refueling and Other Outage Activities

CORRECTIVE ACTION DOCUMENTS

NOTIFICATIONS

50214258	50213411	5022611	50198023	50199120
50167655	50197238	50226116	50207697	

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	1R15 Outage Safety Plan	0
STP M-55	Recording of Cyclic fatigue or Transient	11
PTLR	Pressure temperature limits Curve for Diablo Canyon	9
OP A-2:1X	Reactor Vessel Vacuum Refill of the RCS	12
OP B-2:VI	RHR Draining the Refuel Cavity	5

Section 1R22: Surveillance Testing

CORRECTIVE ACTION DOCUMENTS

NOTIFICATIONS

50214224	50214226	50214227	50214315	50214317
50214597	50214611	50210705		

DOCUMENTS

NUMBER	TITLE	REVISION / DATE
STP P-SIP-22	Routine Surveillance Test of Safety Injection Pump 2-2	28
STP V-203	Inside Containment Sample Isolation Valves	6
STP V-619	Penetration 19 Containment valve Leak Testing	19
STP V-620	Penetration 20 Containment valve Leak Testing	9

NUMBER	TITLE	REVISION / DATE
STP V-656	Penetration 156 Containment valve Leak Testing	19
STP M-9X	Diesel Generator Operability Verification	21
50200836	DG 2-3 Lube Oil Leak Approx .25 - .5 GPM	2/11/2009
STP M-5	Routine Integrated Test of Engineering Safeguards and Diesel Generators	42
STP R-10C	Reactor Coolant System Water Inventory Balance	40
STP R-8A	Reactor Coolant System Leakage Test	13
STP R-19	Shutdown Margin Determination	21

Section 2OS1: Access Controls to Radiologically Significant Areas

PROCEDURES

NUMBER	TITLE	REVISION / DATE
RP1	Radiation Protection	5
RCP D-220	Control of Access to High, Locked High, and Very High Radiation Areas	36
RCP D-240	Radiological Posting	18
RCP D-250	Radiological Occurrence Reports	11
RCP D-500	Routine and Job Coverage Surveys	29
RCP D-610	Controls of Radioactive Materials	14
RCP D-370	Evaluation of Internal Deposition of Radioactive Material	10
TP TO-0824	Core Offload Window Systems Restoration During SGRP	1
MP M-7.1A	Reactor Vessel Closure Head Removal	9

AUDITS, SELF-ASSESSMENTS AND SURVEILLANCES

2008 Radiation Protection Program and Solid Radioactive Waste Management (Process Control) and Transportation Program

CORRECTIVE ACTION DOCUMENTS (NOTIFICATIONS)

50085008	50200344	50202932	50202408	50210881
50203789	50200344	50198337	50201288	50202662
50202321	50203125	50201599	50200361	50200359
50199537	50199328	50199321	50202638	50201738
50204158	50203412	50202663	50201506	50202609

RADIATION WORK PERMITS

RWP#	DESCRIPTION
09-1148	1R15 SGRP Support Structure Modifications
09-1145	1R15 SGRP Secondary Side Activities (non RCS)
09-1022	1R15 SGRP Movement of RX Head and Upper Internals
09-1012	1R15 Remove and Install Insulation in Containment
09-1004	1R15 Radiation Protection in Containment
09-1104	1R15 SGRP Radiation Protection in Containment

SAMPLE RESULTS AND SURVEYS

SURVEY#	DESCRIPTION
1864	Steam generator 1-1 insulation removal
1167	Verify dose rates and contamination levels inside 115' steam generator/RCP cubicles, HRA areas
1162	Insulation removal from steam generator channel head area
1239	Pre-job survey for steam generator work
1998	Steam generator 1-1 cold leg severance survey
2268	Between 3 & 4 RCPs inside bioshield dose rate update
3234	AMS-4 Alarm at Refueling Control Point

3236	Verify Conditions after Containment Evacuation
3237	Air Samples after Head Set in Stand
3243	Depost from Airborne After Head Lift

AIR SAMPLES

3234-1-P	3234-2-P	3234-4-P	3234-6-P	3234-7-P
3234-8-P	3234-9-P	3236-1-P	3236-2-P	3237-2-P
3237-3-P	3243-1-P	3250-1-P		

Section 2OS2: ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION / DATE
RCP D-200	Writing Radiation Work Permits and ALARA Planning	44
RP1.ID1	Requirements for the ALARA Program	4
RP1.ID9	Radiation Work Permits	9

CORRECTIVE ACTION DOCUMENTS (NOTIFICATIONS)

50201062	50083710	50204043	50201062	50199592
50203564	50203702			

Section 4OA1: Performance Indicator Verification

Occupational Radiation Safety

PROCEDURES

NUMBER	TITLE	REVISION / DATE
XI1.DC1	Collection and Submittal of NRC Performance Indicators	9
AWP O-003	NRC Performance Indicators: Occupational Exposure Control Effectiveness	5

Section 4OA2: Identification and Resolution of Problems

DOCUMENTS

DESIGN CHANGE PACKAGES

NUMBER	TITLE	REVISION / DATE
	DCPP OBSERVATION PROGRAM REPORT, Date Range: Jan 8, 09 To Jan 22, 09	
	DCPP OBSERVATION PROGRAM REPORT, Date Range: Jan 22, 09 To Jan 27, 09	
	Corrective Action Board Agenda February 26, 2009	
NCR N0002206	Adverse Trend in FME Events	
SGT 1-3055C	Reinstall Lower Supports 1-3	3/8/2009
AWP SP-003	Oversight and Alignment of SGT CAP with the DCPP CAP	0

NOTIFICATIONS

60008228	50209397	50228353	50227527	50227765
50211723	50209699	50227781	50214966	50221709
50206879	50264145	50211343	50212958	50213003
50206937	50194899	50198089	50198134	50214618
50214224	50210543	50210576	50210760	50209493
50210199	50210166	50210335	50209552	50209081
50209271	50209346	50209541	50209615	50209058
50209103	50210043	50210009	50070516	50209397
50207617	50207822	50207948	50208016	50203677
50203116	50203131	50203515	50203607	50203628
50203713	50203754	50203809	50203841	50203918
50203918	50203128	50203701	50203800	50204073
50204240	50204309	50204438	50205199	50203014
50156645	50166843	50198354	50197097	50194781
50192932	50181146	50195154	50180796	50194876
50196140	50196112	50202747	50201503	50201505
50201506	50201698	50201738	50201867	50202280
50202321	50202321	50202293	50202485	50202701
50201386	50202759	50202856	50202904	50202941
50206804	50206330	50206403	50206804	50207418
50214226	50214631	50213042	50213213	50210187
50214954	50214952	50205652	50199415	50199416
50199622	50179082			