UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
1600 EAST LAMAR BLVD
ARLINGTON, TEXAS 76011-4511

August 3, 2012

Mr. Edward D. Halpin
Senior Vice President and
Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 56, Mail Code 104/6
Avila Beach, CA 93424

Subject: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2012003 AND 05000323/2012003

Dear Mr. Halpin:

On June 22, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Diablo Canyon Power Plant. The enclosed inspection report documents the inspection results, which were discussed on June 21, 2012, 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.

Five NRC-identified findings and one self-revealing finding of very low safety significance (Green) were identified during this inspection. Four of these findings were determined to involve violations of NRC requirements. Additionally, the NRC has determined that a traditional enforcement Severity Level IV violation occurred. This traditional enforcement violation was identified with an associated finding. Further, three licensee-identified violations which were determined to be of very low safety significance are listed in this report. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest these non-cited 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, Region IV; 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.

If you disagree with a cross-cutting aspect assigned in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your

disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the Diablo Canyon Power Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management 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/

Neil F. O'Keefe, Chief
Project Branch B
Division of Reactor Projects

Docket Nos.: 050000275, 050000323
License Nos.: DPR-80, DPR-82

Enclosure: Inspection Report 05000275/2012003 and 05000323/2012003
w/Attachments: Supplemental Information
RFI for Inservice Inspection
RFI for Occupational Radiation Safety Inspection

cc w/Enclosure: Electronic Distribution

Regional Administrator (Elmo.Collins@nrc.gov)
Deputy Regional Administrator (Art.Howell@nrc.gov)
DRP Director (Kriss.Kennedy@nrc.gov)
Acting DRP Deputy Director (Allen.Howe@nrc.gov)
Acting DRS Director (Tom.Blount@nrc.gov)
Acting DRS Deputy Director (Patrick.Louden@nrc.gov)
Senior Resident Inspector (Michael.Peck@nrc.gov)
Resident Inspector (Laura.Micewski@nrc.gov)
Branch Chief, DRP/B (Neil.OKeefe@nrc.gov)
Senior Project Engineer, DRP/B (Leonard.Willoughby@nrc.gov)
Project Engineer, DRP/B (Nestor.Makris@nrc.gov)
Public Affairs Officer (Victor.Dricks@nrc.gov)
Public Affairs Officer (Lara.Uselding@nrc.gov)
Project Manager (Joseph.Sebrosky@nrc.gov)
Branch Chief, DRS/TSB (Ray.Kellar@nrc.gov)
RITS Coordinator (Marisa.Herrera@nrc.gov)
Regional Counsel (Karla.Fuller@nrc.gov)
Congressional Affairs Officer (Jenny.Weil@nrc.gov)
Technical Support Assistant (Loretta.Williams@nrc.gov)
DRP Admin Assistants (R4DRP_AA@nrc.gov)
OEMail Resource
ROPreports
W. A. Maier, RSLO (Bill.Maier@nrc.gov)
R. E. Kahler, NSIR (Robert.Kahler@nrc.gov)
RIV/ETA: OEDO (Silas.Kennedy@nrc.gov)
DRS/TSB STA (Dale.Powers@nrc.gov)

File located: R:_REACTORS_DC\2012\DC2012-03RP-msp.docx ML 12216A105

SUNSI Rev Compl.	☑ Yes ☐ No	ADAMS	☑ Yes ☐ No	Reviewer Initials	NFO
Publicly Avail	☑ Yes ☐ No	Sensitive	☐ Yes ☑ No	Sens. Type Initials	NFO

RIV:SRI:DRP/B	RI:DRP/B	C:/DRS/TSB	AC:/DRS/PSB2	C:/DRS/EB1
MSPeck	LHMicewski	DPowers	JDrake	TRFarnholtz
/RA via E/	***/RA via E/***	***Not Available***	***/RA/***	***/RA/***
7/27/12	8/1/12	8/3/12	8/2/12	8/2/12
C:/DRS/EB2	C:/DRS/PSB1	C:/DRS/OB	C:/DRP/B	
GMiller	MHay	MHaire	NFOKeefe	
/TFarnholtz for/	***/RA/***	***/RA via T/***	***/RA/***	
8/2/12	8/2/12	7/27/12	8/3/12	

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
Report:	05000275/2012003 05000323/2012003
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	March 24 through June 22, 2012
Inspectors:	M. Peck, Senior Resident Inspector L. Micewski, Resident Inspector J. Drake, Senior Reactor Inspector N. Greene, Health Physicist N. Makris, Project Engineer L. Ricketson, Senior Health Physicist L. Willoughby, Senior Project Engineer
Approved By:	N. O'Keefe, Chief, Project Branch B Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000275/2012003, 05000323/2012003; 3/24/2012 – 6/22/2012; Diablo Canyon Power Plant, Integrated Resident and Regional Report; Adverse Weather Protection, Inservice Inspection Activities, Refueling and Other Outage Activities, Radiological Hazard Assessment and Exposure Controls

The report covered a 3-month period of inspection by resident inspectors and an announced baseline inspection by region-based inspectors. Four Green non-cited violations, one Severity Level IV non-cited violation, and one Green finding of significance were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." The cross-cutting aspect is determined using Inspection Manual Chapter 0310, "Components Within the Cross-Cutting Areas." 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: Initiating Events

- Green. Inspectors identified a non-cited violation of Technical Specification 5.4.1.e, for the failure to follow procedures that ensured hand files and wire brushes designated for stainless steel weld preparation were stored and maintained separately from hand files and wire brushes used on carbon steel. Specifically, the inspectors determined that the licensee was not segregating tools as required by Procedure MA1.ID12, "Control of Tools for Use on Stainless Steel," Revision 1, because inspectors observed rust deposits on stainless steel components in the plant. This indicated that carbon steel contaminated tools may have been used on these systems. The licensee took corrective actions to segregate the stainless steel tools that were mixed with tools used on carbon steel. The licensee established segregated locations in tool rooms for the separation of abrasive tools, trained tool room attendants to properly store and mark abrasive tools designated for use on stainless steel and evaluated the systems with indications of rust deposits. This issue was entered into the licensee's corrective action program as Notifications 50475217 and 50475779.

  Failure to assure that hand files and wire brushes designated for exclusive use on stainless steel were stored separately from tools used on other materials was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and, if left uncorrected, could become a more significant safety concern. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a

total loss of their safety function. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported [H.4(c)]. (Section 1R08.1.b(1))

- Green. The inspectors identified a finding for failure to follow applicable ASME Code requirements prior to returning the feedwater system to service after code repairs for flow accelerated corrosion. The licensee failed to recognize a rejectable indication in feedwater piping weld 2K16-550-30 FW 33 observable in the original acceptance radiography film. The licensee entered the issue into their corrective action program as Notifications 50473769 and 50475897 and re-examined the radiographic films for welds performed during Refueling Outage 2R16. A random re-examination of other radiographic films will be completed at a later date.

  This finding was more than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and directly affected the cornerstone objective of limiting events that challenge plant stability. Based on the results of the engineering evaluation that was performed when the flaw was recognized, the inspectors determined that the structural integrity of the feedwater piping was not affected. Based on the results of a significance determination process Phase 1 evaluation, the finding was determined to be of very low safety significance (Green) because it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or an internal/external flood. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure human error prevention techniques, such as self- and peer-checking were used so that work activities are performed safely [H.4(a)]. (Section 1R08.1.b.(2))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after plant engineers failed to adequately translate regulatory requirements and the design bases into the offsite power interface calculation on May 6, 2011. As a result, the licensee failed to demonstrate that the 230 kilo-Volt preferred offsite power source had adequate capacity and capability to supply the minimum required terminal voltage to plant engineering safety features following a limiting transmission system contingency. The licensee took corrective actions to limit the plant load that would automatically transfer to the preferred power source following a unit trip and entered the condition into the corrective action program as Notification 50492766.

  The failure to ensure that the 230 kV power system had adequate capability and capability as defined in the current licensing basis requirements was a performance deficiency. This performance deficiency was more than minor because it was associated with the modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded this

finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making [H.1.(b)]. (Section 1R01.b.(1))

- Green - Severity Level IV. The inspectors identified a non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," because the licensee failed to document an evaluation providing a basis that changes made to the facility and associated changes to Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," did not require prior NRC approval. When a 50.59 review was performed, the licensee incorrectly concluded that only a screening was needed. Plant operators use Procedure OP J-2:VIII to determine the operability of the preferred offsite power system for various transmission system configurations. This change accepted a reduction in the preferred offsite power capacity and capability, below the minimum specified by the current licensing basis, due to local service area load growth. This condition would have likely required prior NRC approval had a 50.59 evaluation been performed. The licensee entered this finding into the corrective action program as Notification 50492767.

  The failure to perform a 50.59 evaluation was also a performance deficiency. The inspectors concluded that this issue involved traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This performance deficiency is more than minor because it was associated with modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions to adopt the licensing basis requirement during decision making [H.1.(b)]. (Section 1R01.b.(2))

Cornerstone: Barrier Integrity

- Green. The inspectors identified a non-cited self-revealing violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," following the unplanned loss of the Unit 1 low temperature overpressure protection system during Mode 5 operations on June 7, 2012. One train of the low temperature overpressure protection system safety function was lost after a

maintenance technician mistakenly opened the breaker providing power to the functioning train performing troubleshooting activities on the other train. The licensee's corrective actions included promptly restoring power to temperature overpressure protection system and entering the condition into the corrective action program as Notification 50488636.

The failure of the plant technician to follow troubleshooting work instructions was a performance deficiency. This performance deficiency was more than minor because the performance deficiency is associated with the human performance attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors concluded that the finding is of very low safety significance (Green) because adequate mitigating equipment remained available and the finding did not constitute a loss of control, as defined in Appendix G, "Shutdown Operations Significance Determination Process." This finding has a cross-cutting aspect in the area of human performance associated with the work practices component because the licensee failed to use human error prevention techniques, such as self- and peer-checking, commensurate with the risk of the assigned task such that work activities were performed safely [H.4(a)]. (Section 1R20)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors reviewed a self-revealing non-cited violation of Technical Specification 5.7.2, which was the result of a worker entering a high radiation area with dose rates greater than 1 rem/hour without knowing of the dose rates in the area. In response, licensee representatives suspended fuel movement, posted the area as a locked high radiation area, documented the occurrence in the corrective action program as Notification 50478716 and evaluated the occurrence.

  Entering a high radiation area with dose rates greater than 1 rem/hour without knowing the dose rates in the area was a performance deficiency. The performance deficiency was more than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of program and process (exposure control) and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation because the failure exposed workers to high dose rates. Using the occupational radiation safety significance determination process, the inspectors determined the finding to be of very low safety significance because: (1) it was not an as low as is reasonably achievable finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This finding has a cross-cutting aspect in the human performance area, resources component, because the licensee did not have adequate facilities and equipment in the form of physical or visual barriers to preclude moving fuel into the vicinity of the spent fuel pool door with the transfer canal drained [H.2(d)]. (Section 2RS01)

B. Licensee-Identified Violations

Violations of very low safety significance that 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 associated corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, Pacific Gas and Electric (PG&E) Company was operating both units at full power.

On April 1, 2012, plant operators reduced Unit 1 to about 53 percent power after ocean debris fouled the condenser cooling water screens. On April 3, the source of debris abated and plant operators returned the unit to full power operation. On April 22 plant operators shutdown Unit 1 for a refueling and maintenance outage. On June 15 the licensee completed outage work, restarted the unit, and increased power output to 78 percent. On June 19 reactor operators rapidly reduced the reactor to 50 percent power after observing high vibration on a main feedwater pump. The licensee determined the high vibration was due to a faulty pump seal. The licensee repaired the main feedwater pump and returned the unit to full power on June 22.

On March 30, 2012, plant operators reduced Unit 2 power to 93 percent after a feedwater heating supply valve failed. The licensee repaired the valve and returned the unit to full power on March 31. On April 23, plant operators reduced Unit 2 to about 20 percent power after ocean debris fouled the main condenser cooling water screens. On April 25, the ocean debris increased, so plant operators conducted a reactor shutdown. On May 1, the source of the ocean debris abated and licensee personnel restarted the reactor, and returned Unit 2 to full power.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Summer Readiness for Offsite and Alternate-ac Power

a. Inspection Scope

The inspectors performed a review of preparations for summer weather for the 230 kV preferred offsite power system, including conditions that could lead to loss-of-offsite power and conditions that could result from high temperatures. The inspectors reviewed the procedures affecting these areas and the communications protocols between the transmission system operator and the plant to verify that the appropriate information was being exchanged when issues arose that could affect the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the transmission system operator and the plant's operations personnel during off-normal or emergency events
- The causes for the changed grid conditions
- The estimates of when the offsite power system would be returned to a normal state
- The notifications from the transmission system operator to the plant when the offsite power system was returned to normal

During the inspection, the inspectors focused on plant-specific design features and the procedures used by plant personnel to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Final Safety Analysis Report Update (FSARU) and performance requirements for 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 attachment 1. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures.

These activities constitute completion of one readiness for summer weather affect on offsite and alternate-ac power sample as defined in Inspection Procedure 71111.01-01.

b. Findings

(1) (Closed) Unresolved Item: 05000275;323/2009003-01, "Corrective Action Following Degraded Offsite Power System"

The inspectors identified Unresolved Item 05000275;323/2009003-01 related to the acceptability of the 230 kV preferred offsite power system to meet design basis requirements. Additional NRC review was needed to determine if the preferred offsite system has sufficient capacity and capability to supply the engineered safety features buses for all required accidents and transients. Based on the discussion and enforcement action below, this item is closed.

Introduction. The inspectors identified a green non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because plant engineers failed to adequately translate regulatory requirements and the design bases into the offsite power interface calculation. The calculation was inadequate to demonstrate that the 230 kV preferred offsite power source was capable of supplying the minimum required terminal voltage for engineering safety feature (ESF) equipment following the limiting single transmission system contingency.

Description. In April 2009, the inspectors identified that the offsite power interface Calculation 359-DC, "Determine 230 kV Grid Interface Requirements as a DCPP Offsite Power Source," Revision 8, and the electrical dynamic loading calculation 357A-DC, "Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis," Revision 12, did not include the immediate load demand for both units. This issue was unresolved pending NRC review of the plant design and licensing bases. On December 14, 2009, the NRC concluded that the current licensing basis (CLB) required the preferred offsite power source to have adequate capacity and capability for the immediate load demand for both units, assuming an accident on one unit and a safe shutdown of the other (NRC Letter, Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Request for Technical Specification Interpretation of 230 Kilovolt System Operability, TAC Nos. ME0711 and ME0712, ADAMS Accession No. ML093130428). The inspectors dispositioned this issue as NCV 05000275; 323/2011002-01, "Inadequate Design Control for the Preferred Offsite Power System." The licensee entered this condition into the corrective action program as Notification 50335438 and concluded that the preferred offsite power source was operable after considering the additional load demand from the second unit. The inspectors reviewed the licensee's evaluation but were unable to verify that the preferred

offsite power source was operable because the licensee failed to include an offsite transmission system normal minus one (N-1) contingency in the analysis. The unresolved item remained open pending additional NRC review of the preferred offsite power system CLB requirements, as discussed in Section 1R01 of Integrated Inspection Report 05000275; 323/2011002.

The original offsite power licensing basis, as described in FSARU, Section 3.1.8.3.1, "Criterion 17, 1971 - Electric Power Systems," stated that the combination of the 230 kV and 500 kV circuits provided the independent sources of offsite power required by General Design Criterion 17, "Electric Power Systems." In NUREG-0675, "Safety Evaluation Report Related to the Operation of Diablo Canyon Nuclear Power Plant, Units 1 and 2," Section 8.2, "Off Site Power," the NRC stated that PG&E completed an electrical grid stability analysis which concluded that the loss of any single generator on the grid would not adversely affect the ability to provide offsite power to the Diablo Canyon Plant. The NRC concluded that a combination of a 230 kV circuit and a 500 kV circuit provided sufficient assurance that redundant and independent sources of offsite power were provided, as required by General Design Criterion 17, and that the design of the offsite power system was acceptable because General Design Criterion 17 was met.

The licensing basis was clarified in License Amendments 132 and 130 (NRC letter to PG&E dated April 29, 1999, "Issuance of Amendments for Diablo Canyon Nuclear Power Plant, Unit No. 1, TAC No. MA0743 and Unit 2, TAC No. MA0744," ADAMS Accession No. ML022390464). These amendments involved upgrades to the preferred offsite power system to offset removal of credit for certain local generation support. During the review of these amendments, PG&E confirmed that the preferred power source met the CLB requirement to have sufficient capacity and capability to supply the necessary voltage to safety system loads following the occurrence of the worst case network contingency (PG&E Letter DCL-98-076, "Response to NRC Request for Additional Information Regarding License Amendment Request (LAR) 98-01," ADAMS Legacy Library Accession No. ML9805280285). The licensee defined this contingency as the loss of the most heavily loaded transmission line, switchyard bus, capacitor bank, or generating unit connected to (or associated with) the transmission network. The licensee also submitted to the NRC the results of transmission network analysis demonstrating that the preferred power source met this CLB requirement. As described in the NRC safety evaluation report (ADAMS ML022390464), approval of the amendment was based, in part, on the preferred power source's ability to meet this CLB requirement. The NRC safety evaluation report stated:

> *For the normal offsite configuration, analysis results indicated that the 230 kV system will continue to remain operable in accordance with licensing bases requirements described above following offsite system contingencies or events. These analysis results satisfy staff review procedures/guidelines described in Section 8.2, Part III.1.(f) to the NRC's Standard Review Plan (NUREG 800) for meeting the requirements of Criterion 17 or 10 CFR Part 50, Appendix A. The results provide reasonable assurance that offsite power will be operable and thus available to safety system loads when needed following an accident. The failure of equipment has been included as single contingencies/events in the analysis.*
>
> *The NRC staff concludes that the component parts of the offsite system have the necessary reliability to assure the availability of offsite power when needed following a design basis event. The proposed offsite system changes meet the*

requirements of Criterion 17 of 10 CFR Part 50, Appendix A, and are considered acceptable.

Based on the above, the NRC staff concludes that there is reasonable assurance that the proposed new offsite system configuration will have sufficient capacity and capability to supply power when needed to safety system loads and other required equipment following a design basis event, that it meets Criterion 17 of 10 CFR Part 50, Appendix A, and that it is therefore acceptable.

The inspectors identified that the capacity and capability of the preferred offsite power source had degraded since NRC approval of License Amendments 132 and 130. This degradation resulted from uncompensated load growth in the Los Padres Service Area. By 2003, this load growth resulted in the preferred offsite power source no longer being capable of supplying the required voltage to plant safety loads following the most limiting transmission network contingency under the most limiting grid loading and voltage conditions. The inspectors concluded that this reduction in 230 kV system capacity and capability should have been recognized by the licensee as having an impact on CLB requirements when Calculation O-23, "General Operating Instructions for Reliable Transmission at DCPP," December 16, 2003, first indicated that the worst-case contingency criteria was not met.

The inspectors also identified that on May 6, 2011, the licensee revised Calculation 359-DC, Section 2.6, "Use of the Transmission Term "N-1" Contingency." This revision added a statement that the CLB requirement to include the limiting contingency when demonstrating that the preferred power source was capable of supplying the necessary terminal voltage to plant ESF equipment did not have to be met. This appears to be evidence of when the licensee first recognized that the 230 kV system no longer met the worst-case contingency requirement from the CLB. Through discussion with licensee personnel, the inspectors determine that the licensee had not understood the underlying regulatory requirement, and had then attempted to interpret it through the letters used to support License Amendments 132 and 130. On June 22, 2012, the licensee took immediate corrective action to block the automatic transfer of large non-essential loads to preferred power following a unit trip. By reducing plant power demand, this action restored the capability and capacity of the preferred offsite power source. The licensee also entered the finding into the corrective action program as Notification 5049228.

To assess the potential impact on the plant systems had they been called upon to mitigate a design basis accident, the inspectors reviewed the Los Padres Service Area electrical loading and voltage for the past three years. Based on the load history, the inspectors identified periods where the combination of transmission system loading and switchyard voltage resulted in insufficient capacity and capability to meet the CLB requirement. However, the duration of these periods were less than Technical Specification allowed out-of-service time for a single offsite power source.

The inspectors concluded the most significant contributor to the performance deficiency was the licensee's failure to demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making when deciding to exclude the worst-case single contingency.

Analysis. The inspectors concluded that the failure of the licensee to ensure that the 230 kV power source met CLB requirements was a performance deficiency. This

performance deficiency was more than minor because it was associated with the Mitigating Systems Cornerstone modification design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the finding in accordance with the Reactor Oversight Program using Management Chapter 0609.04 "Phase 1 - Initial Screening and Characterization of Findings." The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making [H.1.(b)].

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion III, "Design Control," requires the licensee to implement measures to assure that applicable regulatory requirements and the design bases are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the licensee failed to assure that all applicable regulatory requirements and the design bases were correctly translated into Calculation 359-DC, Revision 9. Specifically:

1. In December, 2003, the licensee failed to recognize that the changes to the grid reflected in Calculation O-23, December 16, 2003, affected the required capacity and capability required by General Design Criterion 17, "Offsite Power," and failed to address that change in Calculation 359-DC, "Determine 230 kV Grid Interface Requirements as a DCPP Offsite Power Source." As a result, the licensee failed to assure that applicable regulatory requirements were still being met.

2. On May 6, 2011, the licensee failed to assure that all applicable regulatory requirements and the design bases were correctly translated into Calculation 359-DC while making Revision 9. This revision specifically stated that the design basis requirement to be able to withstand the most limiting offsite transmission system contingency when assessing whether the preferred offsite power source had adequate capacity and capability Was no longer applicable.

Because this finding is of very low safety significance and was entered into the corrective action program as Notification 50492766, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275; 323/2012003-01, "Inadequate Preferred Offsite Power System Design Control."

(2) Failure to Perform a 50.59 Evaluation

Introduction. The inspectors identified a green finding and an associated Severity Level IV NCV of 10 CFR 50.59, "Changes, Tests, and Experiments," because the licensee failed to document an evaluation providing a basis that changes made to the facility and to Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," did not require prior NRC approval. When a 50.59 review was performed, the

licensee incorrectly concluded that only a screening was needed. The inspectors concluded that prior NRC approval would have likely been required had a 50.59 evaluation been performed.

Description. The PG&E power transmission network provider performed periodic evaluations of the capacity and capability of the offsite power transmission systems to provide power to the plant. The licensee used the results of these evaluations as input into the station dynamic loading analyses to demonstrate that each offsite power source could meet the electrical demands of plant ESFs. These transmission system evaluations were incorporated into plant Procedure OP J-2:VIII, “Guidelines for Reliable Transmission Service for DCPP.” Plant operators used Procedure OP J-2:VIII to determine the operability of the preferred offsite power system for various transmission system configurations.

The offsite power transmission system evaluation, completed on December 16, 2003 was incorporated into Revision 4 of Procedure OP J-2:VIII. This evaluation identified that the preferred offsite power system was no longer capable of supplying plant ESF equipment following the worst case single transmission network contingency. On April 4, 2011, the inspectors identified that the licensee had failed to perform a 50.59 review of the changes to the facility’s capacity and capability, as well as the specific changes made to Procedure OP J-2:VIII, Revision 4. The licensee entered this condition into the corrective action program as Notification 50390215. On September 6, 2011, the licensee completed corrective actions, including completing a 50.59 screening of the procedure revision. The licensee concluded that the change was “not adverse” and a 50.59 evaluation was not required.

The inspectors reviewed the changes to Procedure OP J-2:VIII using NEI 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” Revision 1. The inspectors used NEI 96-07 because Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” stated that the methods described in NEI 96-07, were acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. The inspectors concluded that the licensee’s assessment documented in the screening was inadequate to conclude that prior NRC approval was not required. As described in NEI 96-07, Section 4.1, the change should have been considered “adverse” because the reduction of preferred offsite power capacity and capability affected the system design function. NEI 96-07, Section 4.3, specified that the licensee must perform an evaluation of the “adverse” change.

The inspectors concluded that a 50.59 evaluation, if one had been performed, would have likely required the licensee to obtain prior NRC approval for the change. As described in NEI 96-07, Section 4.3.8, a change that resulted in a departure from a method of evaluation described in the CLB requires prior NRC approval. The new OP J-2:VIII transmission system evaluation yielded results that indicated that the result of having the worst-case contingency occur would no longer allow the licensee to demonstrate that the system had sufficient capacity and capability to meet the CLB requirements.

The inspectors concluded that the violation was also a finding because plant personnel failed to follow Procedure TS3.1D2, “Licensing Basis Impact Evaluations,” Revision 31. Procedure TS3.1D2, Section 9.4, also required the licensee to conclude that the change was “adverse” because the offsite power design function was affected.

Procedure TS3.1D2 also required a 50.59 evaluation to be performed for “adverse” changes.

In response to a previous NRC-identified adverse trend in maintaining CLB requirements, PG&E had established a Quality Review Board to review 50.59 evaluations. As described in Notification 50414294, the Quality Review Board rejected the initial 50.59 screening of Procedure OP J-2:VIII on July 2, 2011. The Quality Review Board concluded that Procedure TS3.1D2 required that the screening to identify the change as “adverse,” and that a 50.59 evaluation was required. However, the line organization decided not to perform the 50.59 evaluation because they concluded that the CLB was “ambiguous or arguable,” and that the changes had already been implemented. The licensee finalized the screening on September 6, 2011. The inspectors concluded that the most significant contributor to the finding was use of non-conservative assumptions in the decision not to perform a 50.59 evaluation.

Analysis. The failure to perform a 50.59 evaluation was determined to be a performance deficiency because it was caused by a failure to follow a station procedure. The inspectors concluded that this issue involved traditional enforcement because it had the potential for impacting the NRC’s ability to perform its regulatory function. However, if possible, the inspectors also evaluate the underlying technical issue under the significance determination process to determine the severity of the violation. In this case, the inspectors determined the finding could be evaluated under the significance determination process because the change resulted in plant procedures that accepted a reduction in the capacity and capability of the preferred offsite power source below that required by the CLB. This performance deficiency is more than minor because it was associated with modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The underlying finding was evaluated in accordance with the Reactor Oversight Program using Management Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings.” The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions to adopt the licensing basis requirement when making the decision to perform an 50.59 evaluation [H.1.(b)].

Enforcement. Title 10 of the Code of Federal Regulations 50.59, “Changes, Tests, and Experiments,” Section (d)(1) required, in part, that the licensee maintain records of changes in procedures made pursuant to 10 CFR 50.59(c) and that these records include a written evaluation which provides the basis for the determination that the change, test, or experiment does not require a license amendment. Contrary to this, the licensee changed the facility and made a change to a related procedure but did not maintain a record of the change that included a written evaluation. Specifically, the licensee failed to provide an evaluation that adequately documented that changes in capacity and capability of the 230 kV system and associated changes to

Procedure OP J-2:VIII, Revision 4, did not require prior NRC approval. In accordance with the NRC Enforcement Policy, this violation was classified as a Severity Level IV violation because the underlying technical issue was of very low risk significance. Because this violation was of very low safety significance, was not repetitive or willful, and was entered into the corrective action program as Notification 50492767, it is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy 05000275; 323/2012003-02, "Failure to Perform a 50.59 Evaluation."

.2 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

Since a strong cold front, with sustained high winds and capable of producing gale-force gusts, was forecast in the vicinity of the facility for May 23-25, 2012, the inspectors reviewed the plant personnel's overall preparations/protection for the expected weather conditions. On May 22, 2012, the inspectors walked down the 500 kV switchyard systems because their safety-related functions could be affected, or required, as a result of high wind-generated missiles or the loss of offsite power. The inspectors evaluated the plant staff's preparations against the site's procedures and determined that the staff's actions 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 also toured the plant grounds to look for any loose debris that could become missiles during a windstorm. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the FSARU and performance requirements for the systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. The inspectors also reviewed a sample of corrective action program items to verify that the licensee-identified adverse weather issues at an appropriate threshold and dispositioned them through the corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

No findings were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors performed a partial system walkdown of the following risk-significant system:

- Unit 2, Auxiliary saltwater system train 2-2, April 6, 2012

The inspectors selected this system based on its risk significance relative to the reactor safety cornerstone at the time it was inspected. The inspectors attempted to identify any discrepancies that could affect the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, FSARU, technical specification requirements, administrative technical specifications, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also inspected accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

No findings were identified.

.2 Complete System Walkdown Associated with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal and Containment Spray Systems"

a. Inspection Scope

On May 11, 2012, the inspectors conducted a walkdown of the Unit 1 safety injection system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d).

In addition, the inspectors verified that the licensee had isometric drawings that describe the safety injection system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings:

- High point vents were identified

- High points that do not have vents were acceptably recognizable

- Other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation

- Horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified

- All pipes and fittings were clearly shown

- The drawings were up-to-date with respect to recent hardware changes and any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution

The inspectors verified that piping and instrumentation diagrams accurately described the subject systems, that they were up-to-date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the piping and instrumentation diagrams were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.b). Specific documents reviewed during this inspection are listed in attachment 1.

This inspection completes TI 2515/177. Previous portions of this TI were documented in inspection reports 2011003 and 2011004. These activities also constitute completion of one complete system walkdown sample as defined in Inspection Procedure 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

Quarterly Fire Inspection Tours

a. Inspection Scope

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

- April 6, 2012, Units 1 and 2, Fire Zone 30-A-5, intake structure lower level

- April 11, 2012, Unit 1, Fire Zone 8-B-3, control room ventilation equipment room

- April 17, 2012, Unit 2, Fire Area 6-B-3, bus H battery and vital switchgear room

- April 18, 2012, Unit 1, Fire Area 28, main transformer switchyard

- May 9, 2012, Unit 2, Fire Areas 30-A-1 and 30-A-2, compensating measures to provide firewater to auxiliary saltwater pumps 2-1 and 2-2 during firewater outage at intake structure

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 attachment 1, 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 attachment 1.

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

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

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

- May 1 and 14, 2012, Unit 1, auxiliary saltwater pumps 1-1 and 1-2 control cabling underground vaults
- May 3, 2012, Unit 2, auxiliary saltwater pump vaults

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

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed licensee programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for the Unit 1 containment fan cooling units. The inspectors verified that performance tests performed during the refueling outage were satisfactorily conducted for heat

exchangers/heat sinks and reviewed for problems or errors; the licensee utilized the periodic maintenance method outlined in EPRI Report NP 7552, "Heat Exchanger Performance Monitoring Guidelines," the licensee properly utilized biofouling controls; the licensee's heat exchanger inspections adequately assessed the state of cleanliness of their tubes; and the heat exchanger was correctly categorized under 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

Completion of Sections .1 through .5, below, constitutes completion of one sample as defined in Inspection Procedure 71111.08-05.

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

a. Inspection Scope

The inspectors observed 10 nondestructive examination activities and reviewed 10 nondestructive examination packages that included four types of examinations.

The inspectors directly observed the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Main Steam	MS-1-1013 Weld 3	Visual Test 2
Feedwater	2K16-550-30 FW 33	Magnetic Particle Test
Residual Heat Removal	WIC-5	Penetrant Test
Main Steam	MS-1-1013 Weld 1	Penetrant Test
Main Steam	MS-1-1013 Weld 3	Penetrant Test
Residual Heat Removal	WIC 22	Ultrasonic Test
Residual Heat Removal	WIC-16D	Ultrasonic Test
Feedwater	2K16-550-30 FW 33	Ultrasonic Test
Residual Heat Removal	WIC-95	Ultrasonic Test
Residual Heat Removal	WIC-95	Phased Array Ultrasonic Test

The inspectors reviewed records for the following nondestructive examinations:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Feedwater	2K16-550-30 FW 33	Radiograph
Feedwater	2K16-550-30 FW 2	Radiograph
Feedwater	2K16-550-30 FW 28	Radiograph
Main Steam	MS-1-1013 Weld 3	Visual Test 2
Main Steam	MS-1-1013 Weld 1	Penetrant Test
Main Steam	MS-1-1013 Weld 3	Penetrant Test
Residual Heat Removal	WIC-5	Penetrant Test
Residual Heat Removal	WIC 22	Ultrasonic Test
Residual Heat Removal	WIC-16D	Ultrasonic Test
Residual Heat Removal	WIC-95	Ultrasonic Test

During the review and observation of each examination, the inspectors verified that activities were performed in accordance with the ASME Code requirements and applicable procedures. There were two relevant indications that were left in service. The inspectors observed the ultrasonic examination of these indications and compared the results to records obtained during previous examinations and verified that licensee personnel evaluated the indications in accordance with the ASME Code and approved procedures. The inspectors also verified that the qualifications of nondestructive examination technicians performing the inspections were current.

The inspectors observed one weld on a main steam system pressure boundary.

The inspectors reviewed records for the following welding activities:

SYSTEM	WELD IDENTIFICATION	EXAMINATION TYPE
Main Steam	MS-1-1013 Weld 3	Shielded Metal Arc Welding

The inspectors verified, by review, that the welding procedure specifications and the welder had been properly qualified in accordance with ASME Code, Section IX, requirements. The inspectors also verified, through observation and record review, that essential variables for the welding process were identified, recorded in the procedure qualification record, and formed the bases for qualification of the welding procedure specifications. Specific documents reviewed during this inspection are listed in attachment 1.

These actions constitute completion of the requirements for Section 02.01.

b. Findings

(1) Introduction. Inspectors identified a Green NCV of Technical Specification 5.4.1.e, for the failure to follow procedures that ensured hand files and wire brushes designated for stainless steel weld preparation were stored separately from hand files and wire brushes used on carbon steel, and that stainless steel components which had indications of carbon or soft metal contamination were cleaned and restored.

Description. During inspection of the tool storage areas in the welding shop, machine shop, and the tool issue room in the radiologically controlled area, inspectors identified that hand files and wire brushes designated for either stainless steel or carbon steel weld preparation and maintenance were not stored separately. The inspectors noted that more than 10 hand files marked for use on stainless steel, were rusty and, therefore, most likely had been used on carbon steel. In addition, during system walkdowns, the inspectors identified stainless steel piping and welds with light surface rust. This was an indication that the area may have been cleaned with wire brushes that had previously been used on carbon steel. Inspectors were concerned that the failure to separate tools used for stainless steel weld preparation from tools used for carbon steel preparation could result in the contamination of stainless steel welds and piping by carbon steel filings and affect the material integrity and corrosion resistance of these components.

Inspectors reviewed Procedure MA1.ID12, "Control of Tools for Use on Stainless Steel," Revision 1, and concluded that the licensee staff was not consistently following the procedure to ensure the segregation of abrasive tools designated for use on stainless steel from tools used on carbon steel. Step 5 stated, "The following activities and processes may leave iron-bearing contaminants, such as particles of carbon steel, low alloy steel, or cast iron, embedded in or on the surface. If this occurs, the new passive layer being formed may contain imperfections and rust stains and minor pitting may result." Also, Step 5.2.3 stated, "Workers shall identify and segregate the following types of tools if they are to be used on stainless steel:

- Wire brushes
- Files
- Power brushes and wire wheels
- Grinding wheels and flapper wheels

The licensee investigated the inspectors' concerns and concluded that the storage of files and wire brushes designated for use only on stainless steel in the auxiliary building tool room was not meeting the requirements established in Procedure MA1.ID12. In particular, there was no segregation of files or wire brushes, and there were files designated for use on stainless steel that were rusty and may have been used on carbon steel. The licensee took immediate action to remove the stainless steel designations from tools used on stainless steel that were mixed with tools used on carbon steel. Additionally, the licensee planned to conduct additional training with maintenance personnel regarding the requirements for the separation of abrasive tools that are designated for use on stainless steel from those used on other materials. The licensee also planned to reinforce the standards to the tool room attendants to properly store and mark abrasive tools designated for use on stainless steel and to question the requester of abrasive tools for the end use location so the appropriate tool could be provided.

The inspectors reviewed documentation from two instances in which contaminated/incompatible wire brushes had contributed to corrosion on stainless steel components, Notifications 50320388 and 50320045. The inspectors walked down various safety related and important to safety systems and identified corrosion deposits on stainless steel components that may have been caused by using contaminated stainless steel brushes. Procedure MA1.ID12, “Control of Tools for Use on Stainless Steel,” Revision 1, Section 5.3.1 required, in part, that workers shall perform the following steps if stainless steel surface contamination is suspected or caused by inappropriate use of improperly identified tools:

a. Mechanically clean all of the area contacted by the incorrect tool.

b. Chemically clean the affected area using clean, lint-free cloths with new or redistilled alcohol or acetone followed by demineralized water.

Section 5.3.2 required, in part, that workers restore the affected area by mechanically cleaning with the proper "SS" (stainless steel) identified tools, followed by chemical cleaning, if rust stains or minor pitting is observed on stainless steel surfaces. The inspectors determined that the licensee was not consistently following the procedure to clean stainless steel surfaces that may have been contaminated by inappropriate use of improperly identified tools. This issue was entered into the licensee’s corrective action program as Notifications 50473575 and 50475217.

Analysis. The failure to follow the requirements of Procedure MA1.ID12, “Control of Tools for Use on Stainless Steel,” Revision 1, to assure that hand files and wire brushes designated for exclusive use on stainless steel were stored separately from tools used on other materials was a performance deficiency. This finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and, if left uncorrected, would become a more significant safety concern since contamination of the stainless steel can result in accelerated corrosion rates and early failure of the system piping. Using Inspection Manual Chapter 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” this finding was determined to be of very low safety significance (Green) because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding had a cross-cutting aspect in the area of human performance work practices component in that the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. Specifically, the inspectors observed that the licensee personnel were aware of the requirement to segregate the tools, but contractor personnel were not properly supervised when returning tools to the storage locations [H.4(c)].

Enforcement. Technical Specification 5.4.1.e requires, in part, “Written procedures shall be established, implemented, and maintained covering all programs specified in Specification 5.5.” Technical Specification 5.5.2 requires the primary coolant sources outside containment program. This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The control of tools used on stainless steel was an activity affecting quality, and was implemented by Procedure MA1.ID12, “Control of Tools for Use on Stainless Steel,” Revision 1.

Step 5.2.3 required, in part, that tools marked for use only on stainless steel be stored in a designated location and tools designated for use on stainless steel have the markings removed if used on carbon steel. Steps 5.3.1 and 5.3.2 required that if stainless steel surface contamination is suspected to mechanically clean all of the area contacted by the incorrect tool. Contrary to the above, prior to April 23, 2012, the licensee failed to implement written procedures covering requirements of the Primary Coolant Sources Outside Containment Program. Specifically, the licensee failed to accomplish the separation and appropriate designation of tools used on stainless steel or to clean stainless steel components that had indications of contamination with carbon steel. This issue was entered into the licensee's corrective action program as Notifications 50475217 and 50475779. This finding was determined to be of very low safety significance and was entered into the license's corrective action program; this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275/2012003-03, "Failure to Follow Procedure for the Control of Tools for Use on Stainless Steel."

(2) Introduction. The inspectors identified a Green NRC-identified finding for a failure to follow applicable ASME/ANSI Code requirements prior to returning the feedwater system to service following repairs for flow accelerated corrosion.

Description. As part of the nondestructive examination review during the 2012 Inservice Inspection, the inspectors selected several radiograph films of welds that were completed during the previous outage on Unit 2. The welds were performed as part of the Refueling Outage 2R16 flow accelerated corrosion project piping replacements to the main feedwater system on Unit 2. During the licensee's re-review prior to giving the files to the inspectors for review, an artifact was identified on the digital radiograph image of field weld 33. The weld is located on U2 line 550, feedwater heaters outlet header, a 30-inch diameter carbon steel line located on the 104 foot elevation of the turbine building. The licensee performed a high temperature ultrasonic examination of the weld and determined that the artifact was a relevant flaw in the weld. The main feedwater system piping upstream of the main feedwater isolation valves is designed to meet ASME/ANSI B31.1 (FSARU 10.4.7.1). This code requires that all welds be free of significant flaws prior to use in operation. The comparison of the ultrasonic examination results with the radiographic film indicated that there had been no growth of the flaw during the inservice period. An immediate operability determination indicted the system was safe for continued operation, and a formal structural integrity evaluation was ordered. The licensee entered the issue into their corrective action program as Notifications 50473769 and 50475897 and re-examined the radiographic films for welds performed during Refueling Outage 2R16. A random selection of other radiographic films will be examined as part of the corrective actions.

Analysis. The performance deficiency associated with this finding was a failure to follow the requirements of ASME/ANSI Code B31.1 1992 Edition, Chapter VI Examination, Inspection, and Testing, item 136.4.5 Radiography, (A) Acceptance Standards, which states in part, "Welds that are shown by radiography to have any of the following types of discontinuities are unacceptable: (A. 1) any type of crack or zone of incomplete fusion or penetration." Contrary to this, the licensee returned main feedwater piping to service following Refueling Outage 2R16 with an unacceptable flaw in the weld. This finding was more than minor since it was associated with the human performance attribute of the Initiating Events Cornerstone and directly affected the cornerstone objective of limiting events that challenge plant stability. Based on the results of the engineering

evaluation that was performed when the flaw was recognized, the inspectors determined that the structural integrity of the feedwater piping was not affected. The finding was of very low safety significance (Green) since it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or internal/external flood. This finding had a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure human error prevention techniques such as self- and peer-checking were used so that work activities were performed safely [H.4(a)].

Enforcement. No violation of regulatory requirements occurred. The finding did not represent a regulatory noncompliance issue since it occurred on balance of plant equipment. This finding was entered into the licensee's corrective action program as Notifications 50473769 and 50475897 and was identified as Finding (FIN) 05000323/2012003-04, "Feedwater System Weld Flaw."

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

a. Inspection Scope

The licensee did not perform inspections of the vessel upper head penetrations. No inspections were performed because the vessel upper head and its assembly were replaced and inspected in a previous outage. Therefore, the inspectors determined this section of Inspection Procedure 71111.08 is not applicable.

These actions constitute completion of the requirements for Section 02.02.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors evaluated the implementation of the licensee's boric acid corrosion control program for monitoring degradation of those systems that could be adversely affected by boric acid corrosion. The inspectors reviewed the documentation associated with the licensee's boric acid corrosion control walkdown as specified in Procedure ER1.ID2, Revision 6, "Boric Acid Corrosion Control Program." The inspectors also reviewed the visual records of the components and equipment. The inspectors verified that the visual inspections emphasized locations where boric acid leaks could cause degradation of safety-significant components. The inspectors also verified that the engineering evaluations for those components where boric acid was identified gave assurance that the ASME Code wall thickness limits were properly maintained. The inspectors confirmed that usually the corrective actions performed for evidence of boric acid leaks were consistent with requirements of the ASME Code. Specific documents reviewed during this inspection are listed in attachment 1.

These actions constitute completion of the requirements for Section 02.03.

b. Findings

No findings were identified.

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

a. Inspection Scope

The licensee did not perform inspections of the steam generator tube inspection analysis. No primary side inspections were performed because the steam generators were replaced and inspected in a previous outage and no inspections were required this outage. Therefore, the inspectors determined this section of Inspection Procedure 71111.08 is not applicable.

These actions constitute completion of the requirements for Section 02.04.

b. Findings

No findings were identified.

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

a. Inspection scope

The inspectors reviewed 15 condition reports which dealt with inservice inspection activities and found the corrective actions for inservice inspection issues were appropriate. From this review the inspectors concluded that the licensee has an appropriate threshold for entering inservice inspection issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also has an effective program for applying industry inservice inspection operating experience. Specific documents reviewed during this inspection are listed in attachment 1.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

1 Quarterly Review of Licensed Operator Requalification Program

a. Inspection Scope

On April 18, 2012, the inspectors observed a crew of licensed operators in the plant's simulator during training in preparation for the Unit 1 reactor shutdown. The inspectors assessed the following areas:

- Licensed operator performance
- The quality of training provided
- The modeling and performance of the control room simulator
- Follow-up actions taken by the licensee for identified discrepancies

Specific documents reviewed are listed in attachment 1.

These activities constitute completion of one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

.2 Quarterly Observation of Licensed Operator Performance

a. Inspection Scope

The inspectors observed the performance of on-shift licensed operators in the plant's main control room. At the time of the observations, the plant was in a period of heightened activity due to a Unit 1 refueling outage. The inspectors observed the operators' performance of the following activities:

- April 21, 2012, Unit 1 ramp from 100 percent to 20 percent power
- April 22, 2012, Unit 1 reactor shutdown
- April 28, 2012, Unit 2 reactor startup
- June 15, 2012, Unit 1 reactor startup

In addition, the inspectors assessed the operators' adherence to plant procedures, including Procedure OP1.DC10, "Conduct of Operations," and other operations' department policies. Specific documents reviewed are listed in attachment 1.

These activities constitute completion of four quarterly licensed-operator performance samples as defined in Inspection Procedure 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- Auxiliary building ventilation, Notification 50458813
- Control room ventilation, Notification 50458797

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 attachment 1.

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

b. <u>Findings</u>

No findings were identified.

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

a. <u>Inspection Scope</u>

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:

- Planned maintenance of the Unit 1 pressurizer power-operated relief valve, PORV 456, April 17, 2012
- Planned maintenance of the Unit 2 turbine-driven auxiliary feedwater pump 2-1, April 17, 2012
- Emergent failure synchronize offsite power systems while attempting to manually transfer loads, Unit 1, April 22, 2012
- Planned maintenance of the Unit 2 230 kV startup bus, May 22, 2012

- Mid-loop and reduced reactor coolant inventory operations, Unit 1, June 5, 2012

- Emergent failure of the low temperature over pressurization system during Mode 5 operations, Unit 1, June 7, 2012

- Risk assessment supporting Unit 1 transition to Mode 1 with the control room envelope boundary inoperable, June 7, 2012

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 attachment 1.

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

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following assessments:

- Unit 1, Notification 50360551, November 13, 2010, degraded steam admission valve for the turbine-driven auxiliary feedwater pump

- Units 1 and 2, Notification 50438661, November 3, 2011, control room envelope design vulnerability

- Unit 2, Notification 50470603, April 5, 2012, high residual heat removal pump 2-2 check valve backflow during residual heat removal pump 2-1 testing

- Unit 1, Notification 50475232, April 23, 2012, failure of 4 kV safety buses to transfer to startup power

- Units 1 and 2, Notification 50487115, June 2, 2012, diesel fuel oil transfer pump 0-2 flow indicator failed during surveillance testing

- Unit 2, Notification 50493218, June 21, 2012, auxiliary feedwater hand control controller temperature related failure

The inspectors selected these operability and functionality assessments based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure technical specification operability was properly justified and to verify the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and FSARU to the licensee'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. Additionally, the inspectors 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 attachment 1.

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

b. Findings - Current and Past Operability of the Control Room Habitability System

Introduction. The inspectors identified an unresolved item associated with current and past operability of the control room habitability system.

Description. Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)," required the licensee to maintain two independent and redundant control room ventilation trains and the control room envelope operable. The specified safety function of each control room ventilation train, in conjunction with the control room envelope, was to maintain operator dose below 5 rem equivalent. Technical Specification 5.5.19, "Control Room Envelope Habitability Program," and Surveillance Requirement 3.7.10.5 required the licensee to perform control room envelope in-leakage testing in accordance with Positions C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors." These positions required the in-leakage test to be performed in the most limiting configuration for operator dose, consistent with the plant design and licensing basis.

In November 2011, the licensee completed testing in accordance with Surveillance Requirement 3.7.10.5. During the test, the licensee concluded that control room envelope in-leakage was greater than the in-leakage assumed in the licensing basis accident analysis. The licensee declared the control room envelope inoperable and applied Technical Specification 3.7.10, Actions B.1, B.2, and B.3. Action B.3 allowed continued reactor operation up to 90 days provided that the licensee implemented mitigating actions per Actions B.1 and B.2 to ensure control room envelope occupant exposures would not exceed limits. PG&E conducted additional in-leakage testing using alternate system alignments. The licensee observed that unfiltered in-leakage was reduced to an acceptable value in an alternate alignment with one control room ventilation system train plus one control room ventilation system booster fan from the opposite train in operation. The licensee subsequently established mitigating actions/compensatory measures to maintain a control room ventilation system booster fan from the opposite train available and declared the control room envelope operable.

The inspectors were concerned that the licensee’s operability determination was inconsistent with NRC inspection guidance contained in Regulatory Issue Summary 2005-20, “Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety.” While the results from the alternate alignment in-leakage test demonstrated that control room occupant exposures would not exceed limits, satisfying the requirements for Technical Specification Action B.3, the system alignment for this testing did not appear to satisfy either Technical Specification 5.5.19 or Surveillance Requirement 3.7.10.5. These requirements were not satisfied because the testing was did not use the most limiting configuration for operator dose. Technical Specification Surveillance Requirement 3.0.1 stated that the failure to meet a surveillance requirement, whether the failure was experienced during the performance of the surveillance or between performances of the surveillance, was a failure to meet the Technical Specification Limiting Condition for Operation.

On January 3, 2011, PG&E submitted Licensee Event Report (LER) 1-2011-008-00 “Diablo Canyon Power Plant - Control Room Ventilation System Design Vulnerability.” The licensee reported the failure of the control room envelope as an unanalyzed condition. However, the licensee did not report the failed surveillance test as a condition prohibited by technical specifications. Title 10 CFR 50.73 required the licensee to make a 60 day report to the NRC following discovery of a condition prohibited by technical specifications. These issues are considered unresolved pending NRC review of current and past control room habitability system operability, Unresolved Item: 05000275; 323/2012003-05 “Control Room Habitability Operability Issues.”

1R18 Plant Modifications (71111.18)

Temporary Modifications

a. Inspection Scope

To verify that the safety functions of important safety systems were not degraded, the inspectors reviewed the temporary modification identified as Notification 50488628, disablement of the automatic isolation function of valve HCV-133, residual heat removal isolation letdown.

The inspectors reviewed the temporary modification and the associated safety-evaluation screening against the system design bases documentation, including the FSARU and the technical specifications, and verified that the modification did not adversely affect the system operability/availability. The inspectors also verified that the installation and restoration were consistent with the modification documents and that configuration control was adequate. Additionally, the inspectors verified that the temporary modification was 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.

These activities constitute completion of one sample for temporary plant modifications as defined in Inspection Procedure 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-maintenance Testing (71111.19)

a. Inspection Scope

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

- Unit 2, preventive maintenance of auxiliary feedwater pump 2-1, Work Orders 64074217 and 64066942, April 17, 2012
- Unit 1, corrective maintenance of protection set 1, steam generator 1-1 steam pressure channel, Work Order 64062360, April 19, 2012
- Unit 1, preventive maintenance of safety injection valve SI-8871,Work Orders 60033569, 6400935, and 64001364, April 26, 2012
- Unit 1, preventive maintenance of containment spray valve CS-1-9002B, Work Orders 64083503, 6403384, and 64008639, April 28, 2012
- Unit 1, corrective maintenance of 120 volt vital AC power breaker 1Y14, Work Order 600478565, June 7, 2012

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

- The effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed
- Acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate

The inspectors evaluated the activities against the technical specifications, the FSARU, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance 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 attachment 1.

These activities constitute completion of five post-maintenance testing inspection samples as defined in Inspection Procedure 71111.19-05.

b. Findings

No findings 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 April 22 through June 15, 2012, 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 containment integrity as required by the technical specifications
- Refueling activities, including fuel handling
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing
- Licensee identification and resolution of problems related to refueling outage activities

Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

Unplanned Loss of Low Temperature Overpressure Protection System Due to Failure to Follow Work Instructions

Introduction. The inspectors identified a Green self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after the failure of a maintenance technician to follow procedure resulted in the unplanned loss of the Unit 1 low temperature overpressure protection system safety function.

Description. On June 7, 2012, the safety function of the low temperature overpressure protection system was lost after a maintenance technician opened the power supply breaker for the operable low temperature overpressure protection system train. Power to the other low temperature overpressure protection train had been previously lost due to an electrical breaker failure. The technician mistakenly opened the breaker to the inservice train while performing troubleshooting work on the failed train. The technician used Procedure MA1.DC10, "Troubleshooting," Revision 12, and Maintenance Work Order 60047865-0010 to perform the troubleshooting activities. Procedure MA1.DC10 only allowed work specified on the approved work order, and the approved work order restricted the technician to only manipulate components on the failed train.

The reactor coolant system was in Mode 5 and at low temperature and pressure at the time of the event. Both power-operated relief valves in the low temperature overpressure protection system were rendered inoperable after the technician opened the incorrect breaker. As a result, the low temperature overpressure protection safety function to protect the reactor coolant pressure boundary was lost. The licensee restored one train of low temperature overpressure protection after about 9 minutes. The licensee entered this condition into the corrective action program as Notification 50488636. The inspectors concluded that failure to effectively use human error prevention techniques was the most important contributing cause for the event.

Analysis. The inspectors concluded that the failure of the maintenance technician to follow the troubleshooting work instruction was a performance deficiency. This performance deficiency was more than minor because the finding was associated with the human performance attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. The inspectors used NRC Inspection Manual 0609, Appendix G, "Shutdown Operations Significance Determination Process," to evaluate the significance of the finding. The finding did not require a quantitative assessment because adequate mitigating equipment (residual heat removal relief valves) remained available. The finding screened "Green" because the event did not constitute a loss of control, as defined in Appendix G. This finding had a cross-cutting aspect in the area of human performance associated with the work practices component because the licensee failed to use human error prevention techniques, such as self- and peer-checking, commensurate with the risk of the assigned task such that work activities were performed safely [H.4(a)].

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” required that activities affecting quality be accomplished in accordance with written procedures. Quality Procedure MA1.DC10, “Troubleshooting,” Revision 12, required that troubleshooting activities be conducted in accordance with an approved maintenance work order. Approved Work Order 60047865-0010, issued for troubleshooting activities, restricted the technician to only manipulate equipment affecting the failed low temperature overpressure system train (Pressure Control Valve 456). Contrary to the above, on June 7, 2012, an activity affecting quality was not accomplished in accordance with written procedures. Specifically, a technician using Approved Work Order 60047865-0010 for conducting troubleshooting manipulated equipment in the operable low temperature overpressure system train (affecting Pressure Control Valve 455C). Because the finding was of very low safety significance and has been entered into the licensee’s corrective action program as Notification 50488636, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275/2012003-06, “Failure to Follow Procedure Resulted in the Loss of Low Temperature Overpressure Protection System Safety Function.”

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- Preconditioning
- Evaluation of testing impact on the plant
- Acceptance criteria
- Test equipment
- Procedures
- Jumper/lifted lead controls
- Test data
- Testing frequency and method demonstrated technical specification operability
- Test equipment removal
- Restoration of plant systems

- Fulfillment of ASME Code requirements
- Updating of performance indicator data
- Engineering evaluations, root causes, and bases for returning tested systems, structures, and components not meeting the test acceptance criteria were correct
- Reference setting data
- Annunciators and alarms setpoints

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

- April 16, 2012, Unit 1, routine surveillance test and calibration of containment pressure channel PT-937
- April 17, 2012, Unit 2, inservice test of auxiliary feedwater valve level control valves
- April 17, 2012, Unit 1, local leak rate test of containment penetration 63
- April 18, 2012, Unit 1, local leak rate test of containment penetration 43E
- April 25, 2012, Unit 1, routine surveillance integrated test of engineered safeguards and diesel generators
- May 3, 2012, Unit 2, routine surveillance test of auxiliary saltwater pump 2-2
- May 5, 2012, Unit 1, inservice test of residual heat removal pump 1-1
- June 13, 2012, Unit 1, routine surveillance test and acceleration timing of turbine-driven auxiliary feedwater pump 1-1

Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

No findings were identified.

.2 Surveillance Testing Associated with TI 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems"

a. Inspection Scope

When reviewing Procedures STP M-89, "ECCS Venting," and PEP M-248, "Ultrasonic Testing of Emergency Core Cooling System Piping," the inspectors verified that the procedures were acceptable for testing of the Unit 1 containment spray system during power operation, maintenance, and void determination.

The inspectors reviewed procedures used for conducting surveillances and determination of void volumes to ensure that the void criteria was satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance (TI 2515/177, Section 04.03.a). Also, the inspectors reviewed procedures used for filling and venting following conditions which may have introduced voids into the subject systems to verify that the procedures acceptably addressed testing for such voids and provided acceptable processes for their reduction or elimination (TI 2515/177, Section 04.03.b). Specifically, the inspectors verified that:

- Gas intrusion prevention, refill, venting, monitoring, trending, evaluation, and void correction activities were acceptably controlled by approved operating procedures (TI 2515/177, Section 04.03.c.1)

- Procedures ensured the system did not contain voids that may jeopardize operability (TI 2515/177, Section 04.03.c.2)

- Procedures established that void criteria were satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance (TI 2515/177, Section 04.03.c.3)

- The licensee entered changes into the corrective action program as needed to ensure acceptable response to issues. In addition, the inspectors confirmed that a clear schedule for completion was included for the corrective action program (TI 2515/177, Section 04.03.c.5)

- Procedures included independent verification that critical steps were completed (TI 2515/177, Section 04.03.c.6)

The inspectors verified the following with respect to surveillance and void detection:

- Specified surveillance frequencies were consistent with technical specifications surveillance requirements (TI 2515/177, Section 04.03.d.1)

- Surveillance frequencies were stated or, when conducted more often than required by technical specifications, the process for their determination was described (TI 2515/177, Section 04.03.d.2)

- Surveillance methods were acceptably established to achieve the needed accuracy (TI 2515/177, Section 04.03.d.3)

- Surveillance procedures included up-to-date acceptance criteria (TI 2515/177, Section 04.03.d.4)

- Procedures included effective follow-up actions when acceptance criteria are exceeded or when trending indicates that criteria may be approached before the next scheduled surveillance (TI 2515/177, Section 04.03.d.5)

- Measured void volume uncertainty was considered when comparing test data to acceptance criteria (TI 2515/177, Section 04.03.d.6)

- Venting procedures and practices utilized criteria such as adequate venting durations and observing a steady stream of water (TI 2515/177, Section 04.03.d.7)

- An effective sequencing of void removal steps was followed to ensure that gas does not move into previously filled system volumes (TI 2515/177, Section 04.03.d.8)

- Qualitative void assessment methods included expectations that the void will be significantly less than allowed by acceptance criteria (TI 2515/177, Section 04.03.d.9)

- Venting results were trended periodically to confirm that the systems are sufficiently full of water and that the venting frequencies are adequate. The inspectors also verified that records on the quantity of gas at each location are maintained and trended as a means of preemptively identifying degrading gas accumulations (TI 2515/177, Section 04.03.d.10)

- Surveillances were conducted at any location where a void may form, including high points, dead legs, and locations under closed valves in vertical pipes (TI 2515/177, Section 04.03.d.11)

- The licensee ensured that systems were not pre-conditioned by other procedures that may cause a system to be filled, such as by testing, prior to the void surveillance (TI 2515/177, Section 04.03.d.12)

- Procedures included gas sampling for unexpected void increases if the source of the void is unknown and sampling is needed to assist in determining the source (TI 2515/177, Section 04.03.d.13)

The inspectors verified the following with respect to filling and venting:

- Revisions to fill and vent procedures to address new vents or different venting sequences were acceptably accomplished (TI 2515/177, Section 04.03.e.1)

- Fill and vent procedures provided instructions to modify restoration guidance to address changes in maintenance work scope or to reflect different boundaries from those assumed in the procedure (TI 2515/177, Section 04.03.e.2)

- Fill and vent procedures provided instructions to modify restoration guidance to address changes in maintenance work scope or to reflect different boundaries from those assumed in the procedure (TI 2515/177, Section 04.03.e.2)

The inspectors verified the following with respect to void control:

- Void removal methods were acceptably addressed by approved procedures (TI 2515/177, Section 04.03.f.1)

Specific documents reviewed during this inspection are listed in attachment 1. This inspection completes the final inspection elements of TI 2515/177. The other portions were documented in inspection reports 05000275;323/2011003 and 05000275;323/2011004..

These activities also constitute completion of one surveillance testing inspection sample as defined in Inspection Procedure 71111.22-05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational and Public Radiation Safety

2RS01 Radiological Hazard Assessment and Exposure Controls (71124.01)

a. Inspection Scope

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

The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspectors performed walkdowns of various portions of the plant, performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation reported by the licensee in the occupational radiation safety cornerstone

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

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

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

- Radiological hazards control and work coverage, including the adequacy of surveys, radiation protection job coverage, and contamination controls; the use of electronic dosimeters in high noise areas; dosimetry placement; airborne

radioactivity monitoring; controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools; and posting and physical controls for high radiation areas and very high radiation areas

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

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

Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

Introduction. The inspectors reviewed a self-revealing violation of Technical Specification 5.7.2, which was the result of a worker entering a high radiation area with dose rates greater than 1.0 rem/hour without knowing the dose rates in the area. The violation had very low safety significance.

Description. On May 2, 2012, licensee representatives moved fuel assemblies in the Unit 1 spent fuel pool to inspect them. They worked in accordance with Radiation Work Permit 12-1019-00, "1R17 Fuel Handling at the Spent Fuel Pool." Two fuel handlers were on the spent fuel pool bridge crane. A senior reactor operator observed from the side of the pool. At 2:20 a.m., while the fuel handlers were moving a fuel assembly in front of the door between the spent fuel pool and the fuel transfer canal, the radiation detectors mounted on the spent fuel pool bridge crane alarmed. The fuel handlers stopped briefly to assess the situation and saw the transfer canal was empty. Then, the fuel handlers moved the spent fuel pool bridge crane and the fuel assembly away from the spent fuel pool door and the radiation detectors stopped alarming. The fuel handling team did not stop work, but continued with the fuel inspection. Then, they returned the fuel assembly to its storage location, but they did not approach the spent fuel pool door on the return trip. The senior reactor operator placed the remaining inspection on hold. When the workers exited the radiologically controlled area, radiation protection technicians were alerted to a problem because the fuel handlers had exceeded the dose rate limits of their electronic alarming dosimeters. The electronic alarming dosimeter of one of the fuel handlers indicated a maximum dose rate of 2.4 rem/hour and a dose of 11 millirem. The electronic alarming dosimeter of the second fuel handler indicated a maximum dose rate of 290 millirem/hour and a dose of 3 millirem. The refueling crew said they had not heard their electronic alarming dosimeter dose rate alarms because of the alarms of the radiation detection instruments on the spent fuel pool bridge crane.

In response, licensee representatives suspended fuel movement, posted the area as a locked high radiation area, and evaluated the occurrence. They determined the fuel handlers moved the fuel assembly in front of the spent fuel pool door. Because there was no water in the fuel transfer canal to provide shielding, a stream of radiation caused dose rates to increase beyond what the workers were briefed on and allowed by technical specification to enter. They determined the apparent cause was that physical or visual barriers did not exist to preclude moving fuel into the vicinity of the spent fuel

pool door with the transfer canal drained. As corrective action, a physical barrier was installed to prevent moving a fuel element in front of the spent fuel pool door.

Licensee representatives also identified three contributing causes. The procedural controls were not sufficiently robust to prevent moving a fuel assembly in the vicinity of the spent fuel pool door with the transfer canal drained. As corrective action, procedural guidance will be enhanced. Procedures OP B-8H, “Spent Fuel Pool Work Instructions,” Revision 39, and OP B -7:IX, “Refueling Door to Spent Fuel Pool Door Operation,” Revision 7, will be revised to include instructions to erect a physical barrier when the door between the spent fuel pool and the transfer canal is closed and irradiated fuel assemblies are moved. A precaution and limitation will be added in Procedure OP B-8H detailing the specific coordinates of the exclusion zone and stating the exclusion zone should not be entered with irradiated fuel assemblies if the spent fuel pool door is closed. There was a lack of direct communication to the fuel handling workers of the changing water level in the fuel transfer canal and the associated potential for radiological streaming. As corrective action, each fuel handling crew member will be required to attend a pre-job brief for the work with a refueling senior reactor operator present rather than simply turning over with his or her relief. The workers’ mindset was on the repetition of moving west to south during fuel movement and inspection and they lost track of where they were relative to the spent fuel pool door. This contributing cause would also be addressed by the procedural changes discussed above.

Analysis. Entering a high radiation area with dose rates greater than 1.0 rem/hour without knowing the dose rates in the area is a performance deficiency. The performance deficiency is more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute of program and process (exposure control) and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation because the failure exposed workers to high dose rates. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined the finding had very low safety significance because: (1) it was not an as low as is reasonably achievable finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This finding had a cross-cutting aspect in the human performance area, resources component, because the licensee did not have adequate facilities and equipment in the form of physical or visual barriers to preclude moving fuel into the vicinity of the spent fuel pool door with the transfer canal drained [H.2(d)].

Enforcement. Technical Specification 5.7.2, “High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeter from the Radiation Source or from any Surface Penetrated by the Radiation,” allows entry into such areas only after dose rates in the area have been determined and entry personnel are knowledgeable of them. Contrary to the above, on May 2, 2012, a fuel handler on the spent fuel pool bridge crane entered an area with a dose rate greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation without the dose rates having been determined and without the fuel handler having been made knowledgeable of the dose rates.

Because entering a high radiation area with dose rates greater than 1.0 rem/hour without knowing the dose rates in the area is of very low safety significance and has been entered into the licensee’s corrective action program as Notification 50478716, this

violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy: NCV 05000275/2012003-07, "Entering a High Radiation Area with Dose Rates Greater than 1.0 Rem/Hour Without Knowing the Dose Rates in the Area."

2RS02 Occupational ALARA Planning and Controls (71124.02)

a. Inspection Scope

This area was inspected to assess performance with respect to maintaining occupational individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspectors used the requirements in 10 CFR Part 20, the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspectors interviewed licensee personnel and reviewed the following items:

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

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

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

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

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

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

Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the performance indicator data submitted by the licensee for the first quarter 2012 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, "Performance Indicator Program."

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

b. Findings

No findings were identified.

.2 Safety System Functional Failures (MS05)

a. Inspection Scope

The inspectors sampled licensee submittals for the safety system functional failures performance indicator for Units 1 and 2 for the period from the first quarter 2011 through the first quarter 2012. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports, and NRC integrated inspection reports for the period of January 2011 through March 2012, 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 safety system functional failures samples as defined in Inspection Procedure 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index - Emergency ac Power System (MS06)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - emergency ac power system performance indicator for Units 1 and 2 for the period from the first quarter 2011 through the first quarter 2012. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, mitigating systems performance index derivation reports, issue reports, event reports, and NRC integrated inspection reports for the period of January 2011 through March 2012, to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

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

b. Findings

No findings were identified.

.4 Mitigating Systems Performance Index - High Pressure Injection Systems (MS07)

a. Inspection Scope

The inspectors sampled licensee submittals for the mitigating systems performance index - high pressure injection systems performance indicator for Units 1 and 2 for the period from the first quarter 2011 through the first quarter 2012. To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6. The inspectors reviewed the licensee's operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports, and NRC integrated inspection reports for the period of January 2011 through March 2012, to validate the accuracy of the submittals. The inspectors reviewed the mitigating systems performance index component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

Specific documents reviewed during this inspection are listed in attachment 1.

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

b. Findings

No findings were identified.

.5 Occupational Exposure Control Effectiveness (OR01)

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2011 through the first quarter 2012. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

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

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

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed performance indicator data for the second quarter 2011 through the first quarter 2012. The objective of the inspection was to determine the accuracy and completeness of the performance indicator data reported during these periods. The inspectors used the definitions and clarifying notes contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, as criteria for determining whether the licensee was in compliance.

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

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

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors focused their review on repetitive equipment issues, but also considered the results of daily corrective action item screening discussed in Section 4OA2.2, above, licensee trending efforts, and licensee human performance results. The inspectors nominally considered the 6-month period of January 2012 through June 2012 although some examples expanded beyond those dates where the scope of the trend warranted.

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

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

b. Findings

No findings were identified.

.4 Selected Issue Follow-up Inspection

a. Inspection Scope

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

- Notification 50477779, main steam line steam trap alignment
- Notification 50360551, Unit 1, turbine-driven auxiliary feedwater pump casing warm

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

b. Findings

The inspectors reviewed one licensee-identified finding associated with the steam trap misalignment, as documented in Section 4OA7 of this report.

.4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) LER 05000275/2011-001-00 and -01: Mode Transition with Turbine-Driven Auxiliary Feedwater Pump 1-1 Inoperable

PG&E identified that the Unit 1 turbine-driven auxiliary feedwater pump was inoperable during the transition to Mode 3 on November 6, 2010. Maintenance personnel had replaced the turbine governor prior to Mode transition. Plant operators completed the post-maintenance test following the Mode transition. Plant staff reviewing the completed post-maintenance test identified that the as-found governor speed setting exceeded the acceptance criteria. The licensee concluded that the turbine was inoperable during the Mode change due to the improper speed setting. Plant technicians restored the governor speed set within the acceptance range during the post-maintenance test.

The inspectors previously dispositioned this issue as a licensee identified violation in Section 4OA7.4 of NRC Integrated Inspection Report 05000275/2010005 and 05000323/2010005. The inspectors identified no other concerns. No additional findings were identified during this review.

This LER is closed.

.2 (Closed) LER 05000275; 323/2011-003-00 and -01: Deviation from License Condition for Physical Protection Due to Tsunami Event

On March 11, 2011, PG&E declared an Unusual Event for Units 1 and 2 following a tsunami warning issued by the National Oceanic and Atmospheric Administration National Weather Service, Pacific Tsunami Warning Center for the California West Coast. Plant operators and staff implemented the requirements of Casualty Procedure CP M-5, "Response to Tsunami Warning." During the event, the licensee evacuated personnel from the intake structure for personnel safety. This evacuation was a deviation from Diablo Canyon Power Plant License Condition 2.E. The licensee redeployed personnel to the intake structure later that day, restoring compliance with the license condition. No damage or injuries were observed as a result of this tsunami event and there was no impact on the health and safety of the general public. No findings were identified during this review.

This LER is closed.

.3 (Closed) LER 05000275; 323/2011-006-00 and -01: Loss of Control Room Envelope Due to the Work Control Shift Foreman Incorrectly Authorizing Removal of a Blank Flange

On August 29, 2011, plant operators discovered that the Units 1 and 2 common control room envelope was inoperable due to a maintenance error. Earlier in the day, maintenance personnel had isolated the Unit 2 normal control room ventilation system outside air inlet with a blank flange. Workers had installed the flange to preserve the integrity of the control room envelope while performing maintenance on the inlet isolation dampers. A maintenance technician subsequently removed the blank flange prior to reestablishing operability of the isolation dampers. Removal of the flange resulted in an inoperable control room envelope and would have allowed greater outside air in-leakage into the control room envelope than assumed in the dose analysis. Maintenance personnel reestablished the control room envelope on August 30, 2011.

The failure to maintain the control room envelope in the design configuration was a violation of Technical Specification 3.7.10, "Control Room Ventilation System," and was documented as a licensee-identified violation in Section 4OA7 of NRC Integrated Inspection Report 05000275/2011004 and 05000323/2011004. No additional findings were identified during this review.

This LER is closed.

4OA6 Meetings

Exit Meeting Summary

On April 26, 2012, the inspectors presented the inspection results of the review of inservice inspection activities to Mr. E. Halpin, Senior Vice President and Chief Nuclear Officer, 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 May 10, 2012, the inspectors presented the results of the radiation safety inspections to Mr. E. Halpin, Senior Vice President and Chief Nuclear Officer, 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 June 21, 2012, 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 2.3.2 of the NRC Enforcement Policy for being dispositioned as NCVs.

- Title 10 CFR 50.55a(g)4 requires in part, that ASME Code Class 1, 2, and 3 components be inspected throughout the service life of the reactor. Contrary to the above, until November 2011, the licensee failed to enter the reactor vessel supports, a Class 1 component, into the inservice inspection program and failed to perform required code inspections of accessible portions of reactor vessel supports. The licensee entered this issue into their corrective action program and performed the nondestructive examinations required by ASME Code. This finding is more than minor because if left uncorrected it would become a more significant safety concern. The failure to enter required components into the inservice inspection program and perform required inspections of safety-related components could have allowed undetected flaws to remain in service. These undetected flaws could grow in size until failure of the component, degraded system reliability, or if sufficient general corrosion occurred, a gross failure of the component could occur. The finding was of very low safety significance because

the finding did not represent a loss of safety function and the nondestructive examination for the Unit 1 reactor vessel supports did not identify any relevant indications. The licensee has scheduled the examination for the Unit 2 reactor vessel supports for the next refueling outage. This issue was entered into the licensee's corrective action program as Notification 50433947.

- Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," required that activities affecting quality be accomplished in accordance with written procedures. Diablo Canyon Power Plant Procedure OP1 L-7, "Plant Stabilization Following Reactor Trip," Section 6.9.2 required that the bypass valves for the main steam isolation valves be opened to provide a drain path to the installed steam traps. Contrary to this, from April 25-27, 2012, the licensee failed to open the bypass valves while aligning the steam system following a reactor shutdown. On April 27, 2012, plant operators identified that the steam plant had not in been in the correct valve lineup. This condition could have resulted in the loss of turbine-driven auxiliary feedwater pump safety function due to accumulation of condensation in the steam supply line. This finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding or severe weather initiating event. Pacific Gas and Electric entered the issue into the corrective action program as Notification 50477779.

- Technical Specification 5.7.2 requires each entryway to an area with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and doors and gates shall remain locked except during periods of personnel or equipment entry or exit. Contrary to this requirement, on March 23, 2012, during routine walkdowns, the licensee identified the locked high radiation area door into the reactor coolant pump room 2-2 area, on the 115-foot elevation, was not secured. Although the mechanism locked, the door was ajar and opened when pulled. The licensee confirmed the locking mechanism operated properly. The condition had existed since the first week of June 2011. The licensee acknowledged dose rates in the area during operation were as high as 2 rem/hour because of the presence of nitrogen-16. Using the Occupational Radiation Safety Significance Determination Process, the inspectors determined the finding had very low safety significance because: (1) it was not an as low as is reasonably achievable finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The licensee documented the violation in the corrective action program as Notification 50468048.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

T. Baldwin, Manager, Regulatory Services
J. Becker, Site Vice President
S. David, Director, Site Services
E. Davidson, Foreman, Radiation Protection
J. Fledderman, Director, Strategic Projects
R. Gagne, Foreman, Radiation Protection
P. Gerfen, Manager, Operations
D. Gonzalez, Lead, Inservice Inspection
E. Halpin, Chief Nuclear Officer
M. Huszarik, ALARA Supervisor, Radiation Protection
T. Irving, Manager, Radiation Protection
C. Neary, Welding, Engineering Programs
J. Nimick, Director, Operations Services
P. Nugent, Manager, Technical Support
L. Padovan, Supervisor, Regulatory Services
D. Peterson, Director, Quality Verification
R. Rogers, Foreman, Radiation Protection
L. Sewell, Supervisor, Radiation Protections
J. Summy, Director, Engineering Services
L. Walters, Director, Training
J. Welsch, Station Director

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000275; 05000323/2012003-05	URI	Control Room Habitability Operability Issues (Section 1R15)

Opened and Closed

05000275; 05000323/2012003-01	NCV	Inadequate Preferred Offsite Power System Design Control (Section 1R01)
05000275; 05000323/2012003-02	SLIV	Failure to Perform a 50.59 Evaluation (Section 1R01)
05000323/201200-03	NCV	Failure to Follow Procedure for the Control of Tools for Use on Stainless Steel (Section 1R08)
05000323/2012003-04	FIN	Feedwater System Weld Flaw (Section 1R08)
05000275/2012003-06	NCV	Failure to Follow Procedure Resulted in the Loss of Low Temperature Overpressure Protection System Safety Function (Section 1R20)

Opened and Closed

05000275/2012003-07	NCV	Entering a High Radiation Area with Dose Rates Greater than 1.0 Rem/Hour Without Knowing the Dose Rates in the Area (Section 2RS01)

Closed

05000275; 05000323/2009003-01	URI	Corrective Action Following Degraded Offsite Power System (Section 1R01)
05000275/1-2011-001-00	LER	Mode Transition with Turbine-Driven Auxiliary Feedwater Pump 1-1 Inoperable (Section 4OA3.1)
05000275/1-2011-001-01	LER	Mode Transition with Turbine-Driven Auxiliary Feedwater Pump 1-1 Inoperable (Section 4OA3.1)
05000275; 05000323/1-2011-003-00	LER	Deviation From License Condition For Physical Protection Due to Tsunami Event (Section 4OA3.2)
05000275; 05000323/1-2011-003-01	LER	Deviation From License Condition For Physical Protection Due to Tsunami Event (Section 4OA3.2)
05000275; 05000323/1-2011-006-00	LER	Loss of Control Room Envelope Due to the Work Control Shift Foreman Incorrectly Authorizing Removal of a Blank Flange (Section 4OA3.3)
05000275; 05000323/1-2011-006-01	LER	Loss of Control Room Envelope Due to the Work Control Shift Foreman Incorrectly Authorizing Removal of a Blank Flange (Section 4OA3.3)

Discussed

05000275; 05000323/1-2011-008-00	LER	Diablo Canyon Power Plant - Control Room Ventilation System Design Vulnerability (Section 1R15)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

PROCEDURES

NUMBER	TITLE	REVISION
OP J-2:VIII	Guidelines for Reliable Transmission Service for DCPP	19
0-23	Operating Instructions for Reliable Transmission Service to Diablo Canyon Power Plant	1.10
359-DC	Determine 230 kV Grid Interface Requirements as a DCPP Offsite Power Source	8
357A-DC	Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis	12

Section 1R04: Equipment Alignments

PROCEDURES

NUMBER	TITLE	REVISION
DCM S-17B	Auxiliary Saltwater System	18
OP A-2:VII	Core Offload Window Systems Restoration	30

DRAWINGS

NUMBER	TITLE	REVISION
108017	Saltwater Systems, Sheet 3B	80
106709	Operating Valve Identification Diagram Safety Injection, Sheets 2, 3, and 4	68
102009	Safety Injection System	80
437989	Piping and Mechanical Design Review Isometric Safety Injection, Loop No. 1 & 2 Area “F” and “G” Containment	13
437990	Piping and Mechanical Design Review Isometric Safety Injection Loop Nos. 3 & 4	11
446546	Piping and Mechanical Design Review Isometric Safety Injection System Suction & Discharge	13

Section 1R05: Fire Protection

PROCEDURES

NUMBER	TITLE	REVISION
OP K-2C	Fire Protection Network Operation	34
OM8.ID1	Fire Loss Prevention	22
OM8.ID2	Fire System Impairment	16
OM8.ID4	Control of Flammable and Combustible Materials	19
STP M-70A	Inspection of Fire Barrier and HELB Penetration Seals	6
STP M-70D	Inspection of Fire Barriers, Rated Enclosures, Credited Cable Tray Fire Stops, and Equipment Hatches	13
ECG 18.7	Fire Rated Assemblies	7

DRAWINGS

NUMBER	TITLE	REVISION
111906	Fire Barriers for Units 1 & 2, Intake Structure, 18’ Elevation, Sheet 32	1

Section 1R07: Heat Sink Performance

PROCEDURES

NUMBER	TITLE	REVISION / DATE
STP M-93A, Attachment 9.1	CFCU Visual Inspection Data Sheet	July 20, 2009
STP M-93A	Refueling Interval Surveillance – Containment Fan Cooler System	28

OTHER DOCUMENTS

NCR 001627

Section 1RO8: Inservice Inspection Activities

PROCEDURES

NUMBER	TITLE	REVISION
NDE-UT-2	Ultrasonic Examination of Austenitic Piping	7/8
NDE-UT-3	Ultrasonic Through Wall Sizing in Pipe Welds	2
EPRI-PIPE-MPA-1	Manual Phased Array Ultrasonic Testing of Piping	0
NDE-PT-1	Visible Dye Liquid Penetrant Examination	4
NDE MT-1	Magnetic Particle Examination	14
NDE VT-2-1	Visual Examination During Section XI Pressure Test	2
NDE VT 2 1	Visual Examination During Section XI System Pressure Test	2
ER1.ID2	Boric Acid Corrosion Control Program	6
STP R 8C	Containment Walkdown for Evidence of Boric Acid Leakage	9
AD4.ID2	Plant Leakage Evaluation	10
ISI X-CRDM	Reactor Vessel Top and Bottom Head Visual Inspections	6
AD7.ID11	Fluid Leakage Management Program	1
STP R-8A	Reactor Coolant System Leakage Test	16

PROCEDURES

NUMBER	TITLE	REVISION
MA 1.ID13	ASME Section XI Repair/Replacement Program and Implementation	14
MA 1.ID12	Control of Tools for Use on Stainless Steel	1
TS1.ID3	Steam Generator Management Program	12
WPS 5	ASME/ANSI Weld Procedure Specification Welding of p1 Materials with GTAW and/or SMAW ASMEI, ASME III, ASME VIII, ANSI B31.1, and AWS 5,2	8

OTHER DOCUMENTS

NUMBER	TITLE	REVISION / DATE
TAC NO. ME7236	Diablo Canyon Power Plant, Unit No.1 -Approval of Request for Relief Nde-Rcs-Se-Lp1 Cl to Allow use of Alternate ASME Code Case N-770-1	February 24, 2012
PG&E Letter DCL-11 -101	ASME Section XI Inservice Inspection Program Request for Alternative NDE-RCS-SE-LP1 CL to Allow Use of Alternate ASME Code Case N-770-1 Baseline Exam	September 22, 2011
PG&E Letter DCL-12-007	Request for Approval of an Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Examination Requirements for Class 1 and 2 Piping Welds	January 22, 2012
PG&E Letter DCL-12-023	Diablo Canyon Power Plant Third Interval Snubber Program	February 28, 2012 Revision 1
	Pacific Gas and Electric, Diablo Canyon Unit 1 Refueling Outage 1R16 October 2010 Steam Generator Condition Monitoring and Operational Assessment Mode 4 Report	October 27, 2010
	Pacific Gas and Electric, Diablo Canyon Unit 1 Refueling Outage 1R17 April 2012, Steam Generator Degradation Assessment	April 12, 2012 Revision 0
	Excerpt from Safety Evaluation Report Related to the License Renewal of Diablo Canyon Nuclear Power Plant, Units 1 and 2	June 2011
60030642-0030	Weld Map for MS-1-1013 Piping Configuration	

OTHER DOCUMENTS

NUMBER	TITLE	REVISION / DATE
60030642	MS-1-1013 Body to Bonnet Leak/Valve Replacement	November 7, 2010

NOTIFICATIONS

50071625	50248506	50286663	50303631	50303632
50320045	50320045	50320388	50320388	50343034
50348286	50349063	50349556	50349705	50350525
50350631	50350930	50351088	50351111	50351206
50351282	50351283	50351354	50351355	50351373
50351811	50351916	50351936	50352002	50352022
50352216	50352290	50352315	50352437	50352872
50352961	50353290	50353720	50354177	50354558
50354566	50354573	50355261	50355291	50355772
50355773	50356370	50356719	50357183	50357624
50357702	50357835	50357839	50357901	50358110
50358305	50360397	50361762	50362400	50365637
50366115	50366154	50366946	50366955	50367401
50368189	50368229	50368991	50369159	50369162
50370945	50372290	50373585	50373586	50373937
50375966	50376221	50378349	50380552	50382327
50383654	50384365	50387492	50387927	50388529
50388860	50388907	50390428	50395754	50399371
50399702	50400789	50401975	50402620	50402721
50403209	50404220	50404222	50405716	50407057
50408357	50413075	50417399	50418728	50419411
50420765	50421820	50421972	50425592	50426155
50427009	50427100	50427139	50427962	50428045
50428215	50430212	50432736	50432958	50433947
50434021	50435069	50435250	50435336	50439015
50439858	50441474	50442014	50442386	50445108

NOTIFICATIONS

50445455	50446267	50446715	50446811	50447155
50447161	50447322	50450174	50450175	50450176
50450177	50450178	50450179	50450180	50450358
50451698	50451826	50452554	50455413	50455579
50456984	50457170	50461250	50465003	50466721
50466751	50467251	50467515	50467740	50473475
50473769	50473901	50473903	50474154	50475779

Section 1R11: Licensed Operator Requalification Program

PROCEDURES

NUMBER	TITLE	REVISION
OP1.DC10	Conduct of Operations	30
OP L-4	Normal Operations at Power, Instructions for Power Decrease from 100% to 50%	83
OP1.ID3	Planned Plant Evolution Reactivity Brief	10
OP L-2	Hot Standby to Startup Mode	39

OTHER DOCUMENTS

Unit 1 Cycle 17 Ramp Plan, April 12, 2012

Section 1R12: Maintenance Effectiveness

PROCEDURES

NUMBER	TITLE	REVISION
MA1.ID17	Maintenance Rule Monitoring Program	23

NOTIFICATIONS

50458813

OTHER DOCUMENTS

Maintenance Rule Expert Panel Meeting 185, March 22, 2012

Maintenance Rule Expert Panel Meeting 188 Minutes, May 17, 2012

Maintenance Rule Expert Panel Meeting 186, April 11, 2012

Maintenance Rule Expert Panel Meeting 187, April 19, 2012

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

PROCEDURES

NUMBER	TITLE	REVISION / DATE
MA1.ID17	Maintenance Rule Monitoring Program	24
AD7.DC6	On-Line Risk Management	19
MA1.DC11	Assessment of Risk	11
STP P-DFO-02	Routine Surveillance Test of Diesel Fuel Oil Transfer Pump 0-2	7
AD8.DC55	Outage Safety Scheduling	35
OP1.ID4, Attachment 2	IPTE Pre-Job Brief Guidance for 230 kV Startup Outage for Switchyard work	May 21, 2011
OP O-36, Attachment 4	Active Protected Equipment List	5
OP O-36, Attachment 6	SSC and Component List for U2	5
AD8.DC51, Attachment 8.3	Walkdown Checklist Unit 2 Auxiliary Bank in Service (Startup Bank Cleared)	15

OTHER DOCUMENTS	REVISION / DATE
Risk Assessment DCPP-STRIDE02-PRA	0
Probabilistic Risk Assessment PRA 12-05, Mode Transitions with Control Room Envelope Inoperable	0
LCOTR # 1-TS-12-0572, PY-14 De-Energized	June 7, 2012
LCOTR # 0-TS-12-0113, PY-14 De-Energized	June 7, 2012
LCOTR # 1-TS-12-0573, PY-13 De-Energized	June 7, 2012
LCOTR# 1-TS-12-024, Startup power inoperable	April 21, 2012
LCOTR# 2-TS-12-306, Startup power inoperable	April 21, 2012
Risk Assessment PRA12-05, Rev 0, Unit 1 Transition to Mode 1 with Control Room Envelope Boundary Inoperable	

Section 1R15: Operability Evaluations

PROCEDURES

NUMBER	TITLE	REVISION / DATE
OM7.ID12	Operability Determinations	22
STP P-RHR-21	Routine Surveillance Test of RHR Pump 2-1	23
OP1.DC10	Formal Communication – Drain TDAFWP	April 16, 2012

PROCEDURES

NUMBER	TITLE	REVISION / DATE
AD8.DC51	Outage Safety Management Control of Off-Site Power Supplies to Vital Buses	15
OP J-6:II	Transferring 4160 Volt Banks	14
STP P-DFO-02	Routine Surveillance Test of Diesel Fuel Oil Transfer Pump 0-2	7

NOTIFICATIONS

50470603 50360551

AMS Controller Failure Report – June 22, 2012

DRAWINGS

NUMBER	TITLE	REVISION
108010	Residual Heat Removal System, Sheet 3	28
106704	Auxiliary Feedwater Pump 1-1	88
107704	Auxiliary Feedwater Pump 2-1	75

Section 1R18: Plant Modifications

PROCEDURES

NUMBER	TITLE	REVISION
CF4.ID7	Temporary Alteration	23
TS3.ID2	Licensing Basis Impact Evaluations	31A

NOTIFICATIONS

50488628

Section 1R19: Post-maintenance Testing

PROCEDURES

NUMBER	TITLE	REVISION
STP P-AFW-21	Routine Surveillance Test of Turbine-Driven Auxiliary Feedwater Pump 2-1	25
STP V-3R6	Exercising Steam Supply to Auxiliary Feedwater Pump Turbine Isolation Valves, FCV-37 and FCV-38	11
STP V-3R5	Exercising Steam Supply to Auxiliary Feedwater Pump Turbine Stop Valves, FCV-95	20
STP V-651B	Penetration 51B Containment Isolation Valve leak Testing	21

PROCEDURES

NUMBER	TITLE	REVISION
STP V-630	Penetration 30 Containment Isolation Valve leak Testing	29
STP 1-33D.1	Transmitter Response Time Testing Using Noise Analysis Techniques	12A

Section 1R20: Refueling and Other Outage Activities

PROCEDURES

NUMBER	TITLE	REVISION
	1R17 Outage Safety Plan	0
AD8.DC51	Outage Safety Management Control of Off-Site Power Supplies to Vital Buses	15
PTLR-1	PTLR for Diablo Canyon	11
AD8.DC55	Outage Safety Scheduling	35
OP L-0	Mode Transition Checklists	72
OPA-2:IX	Reactor Vessel Vacuum Refill of the RCS	17A

Section 1R22: Surveillance Testing

PROCEDURES

NUMBER	TITLE	REVISION
STP M-15	Integrated Test of Engineered Safeguards and Diesel Generators	51
STP P-ASW-22	Routine Surveillance Test Auxiliary Saltwater Pump 2-2	23A
STP P-RHR-PS	Preservice Testing of Residual Heat Removal Pumps	10
STP M-74	Auto Start of the ASW Pumps at Low Pressure	8
STP V-3P5	Exercising Valves LCV-106, 107, 108, and 109 Auxiliary Feedwater Pump Discharge	20
STP P-DFO-02	Routine Surveillance Test of Diesel Fuel Oil Transfer Pump 0-2	7
STP V-663	Penetration 63 Containment Isolation Valve Leak Test	18
STP 1-12-P937	Containment Pressure Channel PT-937 Calibration	10
STP ASW-11	Routine Surveillance Testing of Turbine Driven Auxiliary Feedwater Pump 1-1	31
STP P23-C	Acceleration Timing of Turbine Driven Auxiliary Feedwater Pump	16

PROCEDURES

NUMBER	TITLE	REVISION
STP M-15	Integrated Test of Engineered Safeguards and Diesel Generators	51
STP P-ASW-22	Routine Surveillance Test Auxiliary Saltwater Pump 2-2	23A
STP P-RHR-PS	Preservice Testing of Residual Heat Removal Pumps	10
STP M-89	ECCS System Venting	56
PEP M-248	Ultrasonic testing of ECCS Piping	8
OP I-2:I	Containment Spray System – Make Available	22

NOTIFICATIONS

50490628	64084002	64038143	64033877

Section 2RS01: Radiological Hazard Assessment and Exposure Controls

PROCEDURES

NUMBER	TITLE	REVISION
RCP D-202	RWP Work Instructions	4
RCP D-220	Control of Access to High, Locked High, and Very High Radiation Areas	38
RCP D-310	RCA Access Control	24
RCP D-500	Routine and Job Coverage Surveys	33
RCP D-620	Control of Radioactive Sources	7
OP1.DC10	Conduct of Operations	31
OP B-7:IX	Refueling Door to Spent Fuel Pool Door Operation	7
OP B-8H	Spent Fuel Pool Work Instructions	39

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

TITLE	DATE
Quality Performance Report – 3rd Period 2011	November 6, 2011
Quality Performance Report – 2nd Period 2011	June 6, 2011

NOTIFICATIONS

50398358	50398585	50398668	50399284	50399479
50399560	50399682	50399685	50400939	50401477
50403959	50404079	50407065	50407168	50410002

50414802	50415292	50415336	50420194	50431048
50438521	50442103	50449534	50467450	50467490
50467632	50468565			

RADIATION WORK PERMITS

NUMBER	TITLE
12-1019-00	1R17 Fuel Handling at the Spent Fuel Pool
12-1053-01	1R17 RHR Pump 1-1 Motor Swap
12-1065-00	1R17 High Dose Valves
12-1081-00	1R17 Core Exit Thermocouple Replacement

RADIOACTIVE SOURCE LEAK TEST RECORDS

NUMBER	TITLE	DATE
283	Cesium-137 – 5 curies	November 22, 2011
710	Cesium-137 – 65 curies	November 22, 2011
825	Cesium-137 – 400 curies	November 22, 2011
826	Cesium-137 – 130 millicuries	November 22, 2011

Section 2RS02: Occupational ALARA Planning and Controls

PROCEDURES

NUMBER	TITLE	REVISION
RCP D-200	ALARA Planning and Controls	47
RCP D-201	Writing Radiation Work Permits	2
RCP D-202	RWP Work Instructions	4
RP1	Radiation Protection	7
RP1.ID1	Requirements for the ALARA Program	7
RP1.ID2	Use and Control of Temporary Radiation Shielding	10
RP1.ID9	Radiation Work Permits	11

NOTIFICATIONS

50399780	50399935	50400747	50413182	50419109
50419450	50421356	50422113	50423296	50428786
50428976	50430870	50435139	50437874	50439372
50439424	50440759	50440942	50445697	50461779

NOTIFICATIONS

50467673

RADIATION WORK PERMITS

NUMBER	TITLE
12-1019	1R17 Spent Fuel Pool Work
12-1030	1R17 NI and Excore Annulus Work
12-1081	1R17 CET Replacement

RADIATION WORK PERMITS CLOSURE PACKAGES

NUMBER	TITLE
11-0041-00	Filling and Solidification of Liners (Major RP Impact)
11-2002-00	2R16 Scaffolding in Containment
11-2027-01	2R16 Reactor Reassembly
11-2049-00	2R16 Steam Generator Platform Completion
11-2081-00	2R16 Core Exit Thermocouple Replacement

AUDITS, SELF-ASSESSMENTS, AND SURVEILLANCES

NUMBER	TITLE	DATE
	Quality Performance Assessment (QPAR) – Second Period 2011	June 6, 2011
	Quality Performance Assessment (QPAR) – Third Period 2011	November 6, 2011
50413182	Self-Assessment of IER-L2-11-1 Inadequate Collective Radiation Exposure Performance Improvements	September 29, 2011

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	DCPP 5 Year Dose Reduction Plan 2012-2017	0
	1R16 Post-Outage Dose Report	May 10, 2011
	2R16 Post-Outage Dose Chart	May 7, 2012
	2R15 Dose Estimates Comparison Chart	May 9, 2012
	1R17 RWP Limits for Active Permits	May 9, 2012
	1R17 Dose Graphs	May 9, 2012

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	REVISION / DATE
	DCPP 5 Year Dose Reduction Plan 2012-2017	0
	1R16 Post-Outage Dose Report	May 10, 2011
	2R16 Post-Outage Dose Chart	May 7, 2012
	1R17 Plan of the Day Report	May 8-10, 2012
TSR 12-109	Temporary Shielding Request: 91' Unit 1 CTMT Floor Loading	April 24, 2012
TSR 12-153	Temporary Shielding Request: CET Scaffold Shadow – 114' Unit 1 CTMT Cavity Floor Loading	May 4, 2012
TSR 12-154	Temporary Shielding Request: 140' Unit 1 CTMT Reactor Head Port 75	May 2, 2012

Section 4OA2: Identification and Resolution of Problems

PROCEDURES

NUMBER	TITLE	REVISION
OP L-7	Plant Stabilization Following Reactor Trip	19

NOTIFICATIONS

50477779	50360551	50468761

LIST OF ACRONYMS

ADAMS	Agencywide Document Access and Management System
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
CLB	Current Licensing Basis
CRVS	Control Room Ventilation System
EPRI	Electric Power Research Institute
ESF	Engineering Safety Features
FSARU	Final Safety Analysis Report Update
LER	Licensee Event Report
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PG&E	Pacific Gas and Electric
SSC	Structures, Systems and Components

The following items are requested for the
Inservice Inspection at

Diablo Canyon Nuclear Power Plant
April 16, 2012 through April 27, 2012
Integrated Report 2012003

Please provide the requested information. Thank you for your support.

NOTE: In an effort to keep the requested information organized, please submit this information to us using the same lettering system below. For example, all contacts and phone numbers for the above inspector should be in a file/folder titled 1- A, Applicable organization charts in file/folder 1- B, etc.

If you have any questions or comments, please contact me at (817) 200-1558 or e-mail me at James.Drake@nrc.gov.

PAPERWORK REDUCTION ACT STATEMENT
This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, control number 3150-0011.

INSERVICE INSPECTION DOCUMENT REQUEST

Inspection Dates: April 16, 2012 through April 27, 2012 (onsite dates)

Inspection Procedures: IP 71111.08 “Inservice Inspection (ISI) Activities”

Inspectors: James Drake, Senior Reactor Inspector (Lead Inspector - ISI)

A. Information Requested for the In-Office Preparation Week

The following information should be sent to the Region IV office in hard copy or electronic format (ims.certrec.com preferred), in care of James Drake, by April 1, 2012, to facilitate the selection of specific items that will be reviewed during the onsite inspection week. The inspector will select specific items from the information requested below and then request from your staff additional documents needed during the onsite inspection week (Section B of this enclosure). We ask that the specific items selected from the lists be available and ready for review on the first day of inspection. Please provide requested documentation electronically if possible. If requested documents are large and only hard copy formats are available, please inform the inspector(s), and provide subject documentation during the first day of the onsite inspection. If you have any questions regarding this information request, please call the inspector as soon as possible.

A.1 ISI/Welding Programs and Schedule Information

a) A detailed schedule (including preliminary dates) of:

i) Nondestructive examinations planned for Class 1 & 2 systems and containment, performed as part of your ASME Section XI, risk informed (if applicable), and augmented inservice inspection programs during the upcoming outage.

Provide a status summary of the nondestructive examination inspection activities vs. the required inspection period percentages for this interval by category per ASME Section XI, IWX-2400. Do not provide separately if other documentation requested contains this information.

ii) Reactor pressure vessel head examinations planned for the upcoming outage.

iii) Examinations planned for Alloy 82/182/600 components that are not included in the Section XI scope (If applicable).

iv) Examinations planned as part of your boric acid corrosion control program (Mode 3 walkdowns, bolted connection walkdowns, etc.).

v) Welding activities that are scheduled to be completed during the upcoming outage (ASME Class 1, 2, or 3 structures, systems, or components).

b) A copy of ASME Section XI Code Relief Requests and associated NRC safety evaluations applicable to the examinations identified above.

c) A list of nondestructive examination reports (ultrasonic, radiography, magnetic particle, dye penetrant, Visual VT-1, VT-2, and VT-3), which have identified relevant conditions on Code Class 1 & 2 systems since the beginning of the last refueling outage. This should include the previous Section XI pressure test(s) conducted during start up and any evaluations associated with the results of the pressure tests. Also, include in the list the nondestructive examination reports with relevant conditions in the reactor pressure vessel head penetration nozzles that have been accepted for continued service. The list of nondestructive examination reports should include a brief description of the structures, systems, or components where the relevant condition was identified.

d) A list with a brief description (e.g., system, material, pipe size, weld number, and nondestructive examinations performed) of the welds in Code Class 1 and 2 systems which have been fabricated due to component repair/replacement activities since the beginning of the last refueling outage, or are planned to be fabricated this refueling outage.

e) If reactor vessel weld examinations required by the ASME Code are scheduled to occur during the upcoming outage, provide a detailed description of the welds to be examined and the extent of the planned examination. Please also provide reference numbers for applicable procedures that will be used to conduct these examinations.

f) Copy of any 10 CFR Part 21 reports applicable to your structures, systems, or components within the scope of Section XI of the ASME Code that have been identified since the beginning of the last refueling outage.

g) A list of any temporary noncode repairs in service (e.g., pinhole leaks).

h) Please provide copies of the most recent self-assessments for the inservice inspection, welding, and Alloy 600 programs.

A.2 <u>Reactor Pressure Vessel Head</u>

a) Provide the detailed scope of the planned nondestructive examinations of the reactor vessel head which identifies the types of nondestructive examination methods to be used on each specific part of the vessel head to fulfill commitments made in response to NRC Bulletin 2002-02 and NRC Order EA-03-009. Also, include examination scope expansion criteria and planned expansion sample sizes if relevant conditions are identified. (If applicable)

b) A list of the standards and/or requirements that will be used to evaluate indications identified during nondestructive examination of the reactor vessel head (e.g., the specific industry or procedural standards which will be used to evaluate potential leakage and/or flaw indications).

A.3 <u>Boric Acid Corrosion Control Program</u>

a) Copy of the procedures that govern the scope, equipment and implementation of the inspections required to identify boric acid leakage and the procedures for boric acid leakage/corrosion evaluation.

b) Please provide a list of leaks (including Code class of the components) that have been identified since the last refueling outage and associated corrective action documentation. If during the last cycle, the unit was shutdown, please provide documentation of containment walkdown inspections performed as part of the boric acid corrosion control program.

c) Please provide a copy of the most recent self-assessment performed for the boric acid corrosion control program.

A.4 Steam Generator Tube Inspections

a) A detailed schedule of:

i) Steam generator tube inspection, data analyses, and repair activities for the upcoming outage (If occurring)

ii) Steam generator secondary side inspection activities for the upcoming outage. (If occurring)

b) Please provide a copy of your steam generator inservice inspection program and plan. Please include a copy of the operational assessment from last outage and a copy of the following documents as they become available:

i) Degradation assessment

ii) Condition monitoring assessment

c) If you are planning on modifying your Technical Specifications such that they are consistent with Technical Specification Task Force Traveler TSTF-449, “Steam Generator Tube Integrity,” please provide copies of your correspondence with the NRC regarding deviations from the standard technical specifications.

d) Copy of steam generator history documentation given to vendors performing eddy current testing of the steam generators during the upcoming outage.

e) Copy of steam generator eddy current data analyst guidelines and site validated eddy current technique specification sheets. Additionally, please provide a copy of EPRI Appendix H, “Examination Technique Specification Sheets,” qualification records.

f) Identify and quantify any steam generator tube leakage experienced during the previous operating cycle. Also provide documentation identifying which steam generator was leaking and corrective actions completed or planned for this condition (If applicable).

g) Provide past history of the condition and issues pertaining to the secondary side of the steam generators (including items such as loose parts, fouling, top of tube sheet condition, crud removal amounts, etc.)

h) Provide copies of your most recent self assessments of the steam generator monitoring, loose parts monitoring, and secondary side water chemistry control programs.

i) Indicate where the primary, secondary, and resolution analyses are scheduled to take place.

j) Provide a summary of the scope of the steam generator tube examinations, including examination methods such as Bobbin, Rotating Pancake, or Plus Point, and the percentage of tubes to be examined. Do not provide these documents separately if already included in other information requested.

A.5 Additional Information Related to all Inservice Inspection Activities

a) A list with a brief description of inservice inspection, boric acid corrosion control program, and steam generator tube inspection related issues (e.g., condition reports) entered into your corrective action program since the beginning of the last refueling outage (for Unit 1). For example, a list based upon data base searches using key words related to piping or steam generator tube degradation such as: inservice inspection, ASME Code, Section XI, NDE, cracks, wear, thinning, leakage, rust, corrosion, boric acid, or errors in piping/steam generator tube examinations.

b) Please provide names and phone numbers for the following program leads:

Inservice inspection (examination, planning)

Containment exams

Reactor pressure vessel head exams

Snubbers and supports

Repair and replacement program

Licensing

Site welding engineer

Boric acid corrosion control program

Steam generator inspection activities (site lead and vendor contact)

B. Information to be Provided Onsite to the Inspector(s) at the Entrance Meeting (April 16, 2012):

B.1 Inservice Inspection / Welding Programs and Schedule Information

a) Updated schedules for inservice inspection/nondestructive examination activities, including steam generator tube inspections, planned welding activities, and schedule showing contingency repair plans, if available.

b) For ASME Code Class 1 and 2 welds selected by the inspector from the lists provided from section A of this enclosure, please provide copies of the following documentation for each subject weld:

   i) Weld data sheet (traveler)

   ii) Weld configuration and system location

   iii) Applicable Code Edition and Addenda for weldment

   iv) Applicable Code Edition and Addenda for welding procedures

   v) Applicable weld procedures used to fabricate the welds

   vi) Copies of procedure qualification records supporting the weld procedures from B.1.b.v

   vii) Copies of mechanical test reports identified in the procedure qualification records above

   viii) Copies of the nonconformance reports for the selected welds (If applicable)

   ix) Radiographs of the selected welds and access to equipment to allow viewing radiographs (If radiographic testing was performed)

   x) Copies of the preservice examination records for the selected welds

   xi) Copies of welder performance qualifications records applicable to the selected welds, including documentation that welder maintained proficiency in the applicable welding processes specified in the weld procedures (at least 6 months prior to the date of subject work)

   xii) Copies of nondestructive examination personnel qualifications (Visual inspection, penetrant testing, ultrasonic testing, radiographic testing), as applicable

c) For the inservice inspection related corrective action issues selected by the inspectors from section A of this enclosure, provide a copy of the corrective actions and supporting documentation.

d) For the nondestructive examination reports with relevant conditions on Code Class 1 and 2 systems selected by the inspectors from Section A above, provide a copy of the examination records, examiner qualification records, and associated corrective action documents.

e) A copy of (or ready access to) most current revision of the inservice inspection program manual and plan for the current Interval.

f) For the nondestructive examinations selected by the inspectors from section A of this enclosure, provide a copy of the nondestructive examination procedures used to perform the examinations (including calibration and flaw characterization/sizing procedures). For ultrasonic examination procedures

qualified in accordance with ASME Section XI, Appendix VIII, provide documentation supporting the procedure qualification (e.g., the EPRI performance demonstration qualification summary sheets). Also, include qualification documentation of the specific equipment to be used (e.g., ultrasonic unit, cables, and transducers including serial numbers) and nondestructive examination personnel qualification records.

B.2 Reactor Pressure Vessel Head

a) Provide the nondestructive personnel qualification records for the examiners who will perform examinations of the reactor pressure vessel head.

b) Provide drawings showing the following: (If a visual examination is planned for the upcoming refueling outage)

i) Reactor pressure vessel head and control rod drive mechanism nozzle configurations

ii) Reactor pressure vessel head insulation configuration

Note: The drawings listed above should include fabrication drawings for the nozzle attachment welds as applicable.

c) Copy of nondestructive examination reports from the last reactor pressure vessel head examination.

d) Copy of evaluation or calculation demonstrating that the scope of the visual examination of the upper head will meet the 95 percent minimum coverage required by NRC Order EA-03-009 (If a visual examination is planned for the upcoming refueling outage).

e) Provide a copy of the procedures that will be used to identify the source of any boric acid deposits identified on the reactor pressure vessel head. If no explicit procedures exist which govern this activity, provide a description of the process to be followed including personnel responsibilities and expectations.

f) Provide a copy of the updated calculation of effective degradation years for the reactor pressure vessel head susceptibility ranking.

g) Provide copy of the vendor qualification report(s) that demonstrates the detection capability of the nondestructive examination equipment used for the reactor pressure vessel head examinations. Also, identify any changes in equipment configurations used for the reactor pressure vessel head examinations which differ from that used in the vendor qualification report(s).

B.3 Boric Acid Corrosion Control Program

a) Please provide boric acid walkdown inspection results, an updated list of boric acid leaks identified so far this outage, associated corrective action documentation, and overall status of planned boric acid inspections.

b) Please provide any engineering evaluations completed for boric acid leaks identified since the end of the last refueling outage. Please include a status of corrective actions to repair and/or clean these boric acid leaks. Please identify specifically which known leaks, if any, have remained in service or will remain in service as active leaks.

B.4 Steam Generator Tube Inspections

a) Copies of the Examination Technique Specification Sheets and associated justification for any revisions.

b) Copy of the guidance to be followed if a loose part or foreign material is identified in the steam generators.

c) Please provide a copy of the eddy current testing procedures used to perform the steam generator tube inspections (specifically calibration and flaw characterization/sizing procedures, etc.). Also include documentation for the specific equipment to be used.

d) Please provide copies of your responses to NRC and industry operating experience communications such as Generic Letters, Information Notices, etc. (as applicable to steam generator tube inspections) Do not provide these documents separately if already included in other information requested such as the degradation assessment.

e) List of corrective action documents generated by the vendor and/or site with respect to steam generator inspection activities.

B.5 Codes and Standards

a) Ready access to (i.e., copies provided to the inspector(s) for use during the inspection at the onsite inspection location, or room number and location where available):

i) Applicable Editions of the ASME Code (Sections V, IX, and XI) for the inservice inspection program and the repair/replacement program.

ii) EPRI and industry standards referenced in the procedures used to perform the steam generator tube eddy current examination.

Inspector Contact Information:
James Drake
Senior Reactor Inspector
817-276-6558
James.Drake@nrc.gov

Mailing Address:
US NRC Region IV
Attn: James Drake
1600. Lamar Blvd,
Arlington, TX 76011

PAPERWORK REDUCTION ACT STATEMENT This letter does not contain new or amended information collection requirements subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). Existing information collection requirements were approved by the Office of Management and Budget, control number 3150-0011.

The following items are requested for the

Occupational Radiation Safety Inspection

at Diablo Canyon Power Plant

May 7-11, 2012

Integrated Report 2012003

If you have any questions or comments, contact Larry Ricketson at (817) 200-1165 (e-mail – Larry.Ricketson@nrc.gov) or Natasha Greene at (817) 200-1154 (Natasha.Greene@nrc.gov).

1. Radiological Hazard Assessment and Exposure Controls (71124.01)

NOTE: Please submit this information using the same lettering system as below. For example, all contacts and phone numbers for the above inspector should be in a file/folder titled 1- A, Applicable organization charts in file/folder 1- B, etc.

Please provide the requested information for regional inspector review by April 6, 2012.

A List of contacts and telephone numbers for the radiation protection staff

B Applicable organization charts

C Audits and self assessments written since May 9, 2011, related to radiation protection operations

D Radiation Protection Procedure Index

E Specific procedures, if used, related to the following areas: (Additional procedures may be requested by number after the inspector reviews the procedure index.)

1. Radiation Protection Program Description
2. Radiation Protection Conduct of Operations
3. Posting of Radiological Areas
4. RCA Access Controls and Radworker Instructions
5. Conduct of Radiological Surveys

6. High Radiation Area Controls
7. High Radiation Area Key Control
8. Control of Highly Radioactive Items in Pools
9. Radioactive Source Inventory and Control
10. Dosimetry Placement and Multi-Badging

F List of corrective action documents, including corporate and subtiered systems, assigned to or written by the radiation protection operations staff since May 9, 2011

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide this information in a "searchable" format, so the inspector can perform keyword searches.

G A summary of corrective action documents since May 9, 2011, involving unmonitored releases, unplanned releases, or releases in which any dose limit or administrative dose limit was exceeded (to verify the Public Radiation Safety Performance Indicator, in accordance with IP 71151)

H List of radiologically significant work activities scheduled to be conducted during the inspection week(s)

I List of radiation work permits and projected doses, if the plant is conducting a refueling outage

J Radioactive source inventory

2. Occupational ALARA Planning and Controls (71124.02)

NOTE: In an effort to keep the requested information organized, please submit this information to us using the same lettering system below. For example, all contacts and phone numbers for the above inspector should be in a file/folder titled 1- A, Applicable organization charts in file/folder 1- B, etc.

A. List of contacts and telephone numbers for the following areas:
[If different than Part 1]

1 ALARA Planning

2 Radiation protection organization

B. Applicable organization charts

[If different than Part 1]

C. Copies of audits, self-assessments, and LERs, written since May 9 2011, related to:

1 ALARA
2 Electronic dosimeter alarms
3 Teledosimetry
[If different than Part 1]

D. Procedure index for:

1 ALARA Program

[If different than Part 1]

E. Please provide specific procedures related to the following areas. Additional Specific Procedures will be requested by number after the inspector reviews the procedure indexes.

1 RP Program Description
2 ALARA Program
3 ALARA Committee
4 Radiation Work Permit Preparation

F List of corrective action documents, including corporate and subtiered systems, assigned to or written by the radiation protection operations staff since May 9, 2011

NOTE: The lists should indicate the significance level of each issue and the search criteria used. Please provide this information in a "searchable" format, so the inspector can perform keyword searches.

[If different than Part 1]

G Site dose totals and 3-year rolling averages for the past 3 years (based on dose of record)

H Most recent refuel outage report

I List of work activities, greater than 1 rem, since May 9, 2011

Include original dose estimate and actual dose. (Include this item if it was not included in the outage report or if no outage report was published.)

J List of current outage radiation work permits and dose estimates

K Outline of source term reduction strategy