UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

February 13, 2007

John S. Keenan
Senior Vice President - Generation
and Chief Nuclear Officer
Pacific Gas and Electric Company
P.O. Box 770000
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SUBJECT: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2006005 AND 05000323/2006005

Dear Mr. Keenan:

On December 31, 2006, the U.S. Nuclear Regulatory Commission completed an inspection at your Diablo Canyon Power Plant, Units 1 and 2, facility. The enclosed integrated report documents the inspection findings that were discussed on January 10, 2007, with Ms. Donna Jacobs and members of your staff.

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

There were three NRC-identified findings and one self-revealing finding of very low safety significance (Green) identified in this report. These findings involved violations of NRC requirements. However, because of their very low risk significance and because they are entered into your corrective action program, the NRC is treating these four findings as noncited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest any NCV in this report, 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, DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Diablo Canyon Power Plant.

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

Sincerely,

/RA/

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

Dockets: 50-275
50-323
Licenses: DPR-80
DPR-82

Enclosure:
NRC Inspection Report 05000275/2006005
and 05000323/2006005
w/attachment: Supplemental Information

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SUNSI Review Completed: __yes___ ADAMS: ☐ Yes ☐ No Initials: __vgg___
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RIV:RI:DRP/B	RI:DRP/B	SRI:DRP/B	C:DRS/OB
TAMcConnell	MABrown	TWJackson	ATGody
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2/9/07	2/9/07	2/9/07	1/31/07

DRS:PSB	DRS:EB1	DRS:EB2	C:DRP/B
MPShannon	WBJones	LJSmith	VGGaddy
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2/1/07	1/31/07	1/30/07	2/13/07

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-275, 50-323
Licenses:	DPR-80, DPR-82
Report:	05000275/2006005 05000323/2006005
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:	October 1 through December 31, 2006
Inspectors:	T. Jackson, Senior Resident Inspector T. McConnell, Resident Inspector M. Brown, Resident Inspector M. Peck, Senior Resident Inspector - Callaway Plant J. Dodson, Regional Operations Officer J. Drake, Operation Engineer P. Goldberg, Reactor Inspector R. Kellar, Health Physicist R. Lantz, Senior Emergency Preparedness Inspector
Approved By:	V. G. Gaddy, Chief, Projects Branch B Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000275/2006-005, 05000323/2006-005; 10/1/06 - 12/31/06; Diablo Canyon Power Plant Units 1 and 2; Maintenance Risk Assessments and Emergent Work Control, Problem Identification and Resolution, Operability Evaluations, and Other Activities.

This report covered a 13-week period of inspection by resident inspectors and Region-based health physics and reactor inspectors. Three NRC-identified and one self-revealing, Green, noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609 “Significance Determination Process.” Findings for which the Significance Determination Process does not apply may be Green or be assigned a severity level after NRC management review. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for the failure to apply adequate design control measures regarding the installation of thimble tubes with chrome-plated bands. Specifically, Pacific Gas and Electric Company installed thimble tubes with chrome-plated bands at the fuel assembly bottom nozzle/lower core plate interface to address flow-induced vibration wear. Due to the failure of engineering personnel to account for the chrome-plated bands in the thimble tube relocation procedure, the chrome-plated band on Thimble Tube L-13 was removed from its designed location at the fuel assembly bottom nozzle, thereby increasing the potential for thimble tube through-wall wear. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Nonconformance Report N0002211.

  The finding is greater than minor because it is associated with the Initiating Events Cornerstone attribute of design control and affects the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because, assuming the worst-case degradation, the finding would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage or affect mitigating systems. Specifically, the inspectors verified the worst-case leakage, i.e., guillotine break, from a thimble tube at the fuel assembly bottom nozzle/lower core plate interface to be approximately 7 gpm versus the Technical Specification reactor coolant system identified leakage limit of 10 gpm. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because Pacific Gas and Electric Company removed a corrective action to prevent recurrence of significant thimble tube wear (Section 4OA5.5).

Cornerstone: Mitigating Systems

- Green. An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” was identified for the failure to promptly correct a condition adverse to quality. Specifically, on October 15, 2006, Pacific Gas and Electric Company implemented a temporary modification to Vital Battery 1-1 contrary to American National Standards Institute/Institute of Electrical and Electronics Engineers Standard 450-1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.” Additionally, surveillance tests to monitor the condition of the degraded battery cell were adversely affected by the installed temporary modification. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0678820.

  The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because it did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program in that engineering staff did not thoroughly assess the operability of the battery and correct a condition adverse to quality in a timely manner (Section 1R15).

- Green. An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of engineering personnel to apply adequate design control measures. Specifically, on February 9, 2006, engineering personnel changed the acceptance criteria in the auxiliary saltwater pump surveillance test from greater than zero packing leak-off to zero packing leak-off with packing gland temperature less than 120°F. The acceptance criteria change was based on engineering judgment, even though vendor documentation called for greater than zero packing leak-off to prevent packing and pump shaft damage. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0684631.

  The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to be of very low safety significance because it did not represent an actual loss of system safety function, did not represent an actual loss of a single train for greater than its Technical

Specification allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with resources because engineering personnel failed to provide up-to-date design documentation to support a design change in surveillance test acceptance criteria (Section 4OA2.2).

Cornerstone: Barrier Integrity

- Green. An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” was determined for the failure of engineering and operations personnel to promptly identify and correct a condition adverse to quality. On two occasions between September 29 and November 9, 2006, operations and engineering personnel: (1) failed to address operability when using manual actions in place of automatic actions associated with the auxiliary building ventilation system and (2) failed to fully address the impact of debris between the circuit card and the panel connections of the auxiliary building ventilation system. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0678429.

  The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of structure, system, component, and barrier performance and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding only represents a degradation of the radiological barrier function provided for the auxiliary building. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because operations and engineering personnel did not adequately evaluate operability of the auxiliary building ventilation system due to the failure to fully encompass all aspects of the degraded conditions and corresponding compensatory measures (Section 1R13).

REPORT DETAILS

Summary of Plant Status

Diablo Canyon Unit 1 began this inspection period at 100 percent power. On November 26, 2006, operators reduced power to 50 percent for circulating water tunnel cleaning. Upon completion of the maintenance, reactor power was returned to 100 percent on December 1 and maintained that power level for the remainder of the inspection period.

Diablo Canyon Unit 2 began this inspection period at 100 percent power. On December 10, operators reduced power and manually tripped the reactor due to indications of Reactor Coolant Pump (RCP) 2-2 high stator temperature. Following repair activities, operators restarted the Unit 2 reactor on December 11, entering Mode 2 (startup). On December 11, while reactor power was being restored to 100 percent, Circulating Water Pump (CWP) 2-1 experienced an electrical short at the motor terminal leads, resulting in a reactor trip. Following repair activities, operators restarted the Unit 2 reactor on December 12, entering Mode 2. Reactor power was returned to 100 percent on December 19 and remained at that power level for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors walked down the six below listed plant areas to assess the material condition of active and passive fire protection features and their operational lineup and readiness. The inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional and that access to manual actuators was unobstructed; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features and that the compensatory measures were commensurate with the significance of the deficiency; and (7) reviewed the Final Safety Analysis Report (FSAR) Update to determine if Pacific Gas and Electric Company (PG&E) identified and corrected fire protection problems.

- October 3, 2006, Unit 1, Control Room Cable Spreading Room
- October 3, 2006, Unit 2, Control Room Cable Spreading Room

- October 10, 2006, Unit 1, Diesel Engine Generator Rooms

- October 10, 2006, Unit 2, Diesel Engine Generator Rooms

- October 11, 2006, Units 1 and 2, 140' and 85' elevation fire equipment storage lockers

- December 28, 2006, Unit 1, 85' elevation turbine building

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R07 Biennial Heat Sink Performance (71111.07B)

.1 Performance of Testing, Maintenance and Inspection Activities

a. Inspection Scope

The inspectors selected three heat exchangers that were either directly or indirectly connected to the safety-related service water system. The inspectors reviewed PG&E's test and cleaning methodology for the following heat exchangers:

- Emergency Core Cooling System Pump Lube Oil Coolers
- Containment Fan Cooler Units
- Residual Heat Removal (RHR) Heat Exchangers

In addition, the inspectors reviewed test data and inspection and cleaning records for the heat exchangers and design and vendor-supplied information to ensure that the heat exchangers were performing within their design bases. The inspectors also reviewed chemical controls to avoid fouling, the heat exchanger test, and inspection and cleaning results. Specifically, the inspectors reviewed design conditions, appropriate use of test instrumentation, and appropriate accounting for instrument inaccuracies. Additionally, the inspectors reviewed inspection and cleaning results and trending results, if available. The inspectors reviewed the methods and results of heat exchanger inspection and cleaning to verify that the methods used to inspect and clean were consistent with industry standards. The results were found appropriately dispositioned such that the final conditions were acceptable.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Verification of Conditions and Operations Consistent with Design Bases

a. Inspection Scope

For the selected heat exchangers, the inspectors reviewed the documents listed in the attachment to verify that PG&E established heat sink and heat exchanger condition and that operation and test criteria were consistent with the design assumptions. Specifically, the inspectors reviewed the applicable calculations to ensure that the thermal performance test acceptance criteria for the heat exchangers were being applied consistently throughout the calculations. The inspectors also reviewed documents in order to verify that the appropriate acceptance values for fouling and tube plugging for the heat exchangers cooled by the component cooling water heat exchangers remained consistent with the values used in the design-basis calculations.

b. Findings

Introduction: An unresolved item (URI) was identified regarding inadequate design control measures for verifying the adequacy of the safety-related RHR system heat exchangers. PG&E stated that the RHR heat exchangers were not inspected and cleaned, due to as low as is reasonably achievable (ALARA) considerations, and the heat exchangers were not tested. Test Procedure STP V-13A, "CCW Flow Balancing," along with Calculation M-1017, "Component Cooling Water System," were used by PG&E to determine if the RHR heat exchanger would meet its design basis. The calculation only establishes the flow balance for the component cooling water (CCW) system based on PG&E's modeling assumptions. This issue is unresolved for both significance and enforcement, since additional technical review by NRC was needed to assess this issue.

Description: The inspectors reviewed Calculation M-1017, "Component Cooling Water System," Revision 3, and found that the purpose of the calculation was to compute the flow rates in the CCW system for input into the accident analysis. The inspectors noted that the calculation only established the flow balance of the CCW system based on PG&E's modeling assumption. The calculation did not address the heat transfer capability of any of the heat exchangers and the references listed did not appear to provide heat transfer capability. The inspectors noted that the assumptions used by PG&E could vary the results of the calculation. Some of the assumptions were: the manufacturer's pressure drop data for the CCW, RHR, and containment fan cooling unit (CFCU) heat exchangers was assumed to be accurate; the RHR heat exchanger throttle valve was assumed to be in the full open position without verification; and even though the CCW systems of Units 1 and 2 are slightly different, PG&E concluded that the results of Unit 1 were applicable for Unit 2. Any of these assumptions, if incorrect, could change the results.

The inspectors reviewed Surveillance Test Procedure STP-V13A, “CCW Flow Balancing,” Revision 15. The purpose of the test is to adjust the CCW flow to assure that the CCW system will supply sufficient flow to vital equipment without exceeding the CCW design temperature limitations during the worst-case accident conditions. The inspectors noted that the test was conducted with CCW flow to Vital Headers A and B. The flow is balanced using the CFCU CCW throttling valve. Flow to each CFCU is adjusted to establish a flow rate between the minimum and maximum flow limitations. Based on the known hydraulics of the CCW system, when the CCW flow to each CFCU is throttled to within the desired range, the overall header will be appropriately balanced. The inspectors noted that, during the test, CCW flow to the RHR heat exchangers was secured.

The inspectors reviewed Action Request (AR) A0588366, initiated on August 3, 2003, which was written to develop a heat exchanger program to test and monitor heat exchangers. Based on engineering judgement, PG&E decided that preventive maintenance on the RHR heat exchangers could not be justified based on ALARA concerns and dose considerations. The inspectors noted that performance monitoring and trending were recommended to ensure that heat exchangers do not fail to perform their safety function. However, the RHR heat exchangers were not recommended for monitoring and trending. The inspectors found that the RHR heat exchangers were not tested, not inspected and cleaned, and not monitored and trended. The inspectors determined that PG&E had not demonstrated that the RHR heat exchangers would meet their safety function.

PG&E stated that they would send additional material for the inspectors to review in order to demonstrate that the RHR heat exchangers would perform their safety function.

<u>Analysis</u>: At the time of writing, PG&E had not demonstrated that the RHR heat exchangers would meet their safety function. This issue is potentially more than minor because it could affect the Mitigating Systems Cornerstone objective by causing the safety-related RHR system to not transfer sufficient heat to the CCW system to support the safety-related systems.

<u>Enforcement</u>: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of the design, such as by the performance of design reviews, by the use of alternative or simplified calculation methods, or by performance of a suitable testing program. Additional review by NRC is needed to determine if the RHR heat exchangers would meet their design safety function. Therefore, this item will be treated as a URI pending additional review: URI 50-275; 323/06-05-01, Additional Review of Material to Determine if the RHR Heat Exchangers Will Meet Their Safety Function.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors verified that PG&E had entered significant heat exchanger/heat sink performance problems into the corrective action program (CAP). The inspectors reviewed 17 ARs.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On November 28, 2006, the inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a volume control tank rupture, an auxiliary saltwater pump trip, an earthquake, and an anticipated transient without scram. Documents reviewed by the inspectors included:

- Lesson FRS1-B, "ATWS," Revision 10
- Procedure EP G-1, "Emergency Classification and Emergency Plan Activation," Revision 34

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Maintenance Effectiveness Inspection

a. Inspection Scope

The inspectors reviewed the one below listed maintenance activity to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the Maintenance Rule, 10 CFR Part 50, Appendix B, and the Technical Specifications (TS).

- November 7, 2006, Unit 1, Auxiliary Feedwater System Discharge Valves FW-1-LCV-113/115

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessments and Management of Risk

a. Inspection Scope

The inspectors reviewed the three below listed assessment activities to verify: (1) performance of risk assessments when required by 10 CFR 50.65(a)(4) and PG&E procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that PG&E recognizes, and/or enters as applicable, the appropriate risk category according to the risk assessment results and PG&E procedures; and (4) PG&E identified and corrected problems related to maintenance risk assessments.

- October 4, 2006, Unit 1, Preventive maintenance on Auxiliary Saltwater Cross-tie Valve SW-0-FCV-601, RCP undervoltage/under-frequency relay testing, and Morro Bay - Diablo Canyon 230 kV line outage

- October 9, 2006, Unit 2, Planned maintenance outage of condensate booster pump

- November 17, 2006, Unit 1, Installation of cell jumper on Vital Battery 1-1

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

.2 Emergent Work

a. Inspection Scope

The inspectors: (1) verified that PG&E performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) verified that emergent work-related activities, such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) reviewed the FSAR Update to determine if PG&E identified and corrected risk assessment and emergent work control problems.

- September 26, 2006, Unit 1, Emergent failure of POV-1 ventilation logic cabinet
- October 9, 2006, Unit 1, Emergent failure of a reactor protection channel
- November 3, 2006, Units 1 and 2, 230 kV disconnect switch warm termination
- November 22, 2006, Units 1 and 2, Hydrochloric acid developed by auxiliary saltwater piping cathodic protection
- December 18, 2006, Unit 2, Indication of nuclear fuel leak

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five samples.

b. Findings

Introduction: A Green NRC-identified, noncited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” was determined for the failure of engineering and operations personnel to promptly identify and correct a condition adverse to quality. On two occasions regarding poor circuit card connections in the auxiliary building ventilation control panel, operations and engineering personnel: (1) failed to address operability when using manual actions in place of automatic actions and (2) failed to fully address the impact of debris between circuit card and panel connections.

Description: On September 27, 2006, the Unit 1 Auxiliary Building Ventilation System Train B control panel (POV-2) was de-energized for planned maintenance. Shortly thereafter, the Train A control panel (POV-1) ac power controller tripped. At the time, both cabinets were declared inoperable and TS 3.0.3 was entered. Control Panel POV-2 panel was returned to service and operators moved from TS 3.0.3 to TS 3.7.12 for having one train of the auxiliary building ventilation system inoperable. To facilitate troubleshooting and repairs, on September 29, maintenance personnel de-energized both Panels POV-1 and POV-2. De-energizing the control panels placed the system dampers in their safety-related position. Additionally, operators manually started one auxiliary building exhaust fan and planned to manually start the other exhaust fan if needed during a design basis event. The controls for the auxiliary building exhaust fans were located in the cable spreading room. Operators subsequently declared both trains of the ventilation system operable and exited TS 3.7.12.

The inspectors questioned the decision to declare the train with the standby exhaust fan fully operable by taking credit for manual actions to start the fan during a design basis event. The attachment to Regulatory Information Summary 2005-20, Part 9900 Technical Guidance "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality of Safety," Section C.5, states that substitution of manual actions for automatic actions requires an operability determination. However, an operability determination was not performed. In response to NRC questioning, operators declared Control Panel POV-1 inoperable and re-entered TS 3.7.12. The inspectors observed that operators would have been able to start the standby exhaust fan had it been necessary; therefore, they would have been able to produce an operability determination.

Subsequent troubleshooting revealed the cause of the Control Panel POV-1 power supply to trip was oxidation buildup and dust collection on the terminals. An interlock feature on the control panels de-energizes the panel if a circuit card becomes loose. Control Panel POV-2 was also inspected and several loose cards were found and cleaned. The extent of condition review by engineering personnel concluded that the Unit 2 control panels were not affected due to the interlock feature being disabled. Unit 2 Control Panel POV-1 was inspected on November 9, 2006, during a scheduled card replacement. Nine cards were found to have loose connections, and these cards were cleaned and reinstalled. Unit 2 Control Panel POV-2 was scheduled for inspection in April 2007.

The inspectors concluded that engineering personnel had failed to adequately address operability of the Units 1 and 2 auxiliary building ventilation system control panels. Engineering and maintenance staff were aware that debris was being blown onto the control panel circuit cards by the panel's ventilation fans. The debris would work its way between the circuit card and panel connection points and cause the poor connections. During a design basis event, particularly a seismic event, the connections may not provide an adequate circuit path. While engineering staff discussed the fact that the interlock feature would not de-energize the control panels if a loose card were detected, their assessment did not discuss the affect of the debris on other connection points and what the impact that would have on the auxiliary building ventilation system. The inspectors did observe that any loose connection would still provide an alarm to the control room operators and failure of the control panels would move the dampers to their safety-related position. Additionally, operators would be able to manually control the exhaust fans. Therefore, engineering staff had a basis for determining that the control panels were degraded but operable, but failed to provide this basis in a timely manner.

<u>Analysis</u>: The performance deficiency associated with this finding involved two examples where PG&E personnel failed to perform an adequate operability determination. The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of SSC and barrier performance and affects the

associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding only represents a degradation of the radiological barrier function provided for the auxiliary building. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the CAP because operations and engineering personnel did not adequately evaluate operability of the auxiliary building ventilation system regarding use of manual actions and the full impact of debris on circuit card connections.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. Contrary to this, on two occasions between September 29 and November 9, 2006, PG&E staff failed to promptly identify and correct a condition adverse to quality when they failed to adequately evaluate operability of the auxiliary building ventilation system. In the first example, operations personnel credited manual actions in place of automatic safety actions without evaluating the capability of those actions in an operability determination. In the second example, engineering personnel failed to fully address the effect of debris between circuit card connections and their impact on system operability. Because the finding is of very low safety significance and has been entered into the CAP as AR A0678429, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy:
NCV 50-275; 323/06-05-02, “Failure to Adequately Evaluate Operability of Auxiliary Building Ventilation Control Panels.”

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plant status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the FSAR Update and design bases documents to review the technical adequacy of the operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any TS; (6) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (5) verified that PG&E has identified and implemented appropriate corrective actions associated with degraded components.

- October 17, 2006, Unit 2, Valves FW-2-LCV-111 and FW-2-LCV-115 placed in manual due to 10 CFR Part 21 Notice

- October 30, 2006, Unit 1, Diesel Engine Generator 1-3 slow to reach rated speed during surveillance test

- November 13, 2006, Unit 1, Vital Battery 1-1 Cell 15 low voltage

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

Introduction: An NRC-identified, NCV of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” was identified for the failure to promptly correct a condition adverse to quality. Specifically, on October 15, 2006, PG&E implemented a temporary modification to Vital Battery 1-1, contrary to American National Standards Institute/Institute of Electrical and Electronics Engineers (ANSI/IEEE) Standard 450-1987, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.” Additionally, surveillance tests to monitor the condition of the degraded battery cell were adversely affected by the installed temporary modification.

Description: The vital batteries are required to provide adequate power to loads necessary for plant cooldown during a station blackout. The battery system consists of three, 60-cell, 125 V battery banks with five battery chargers. Two chargers are capable of charging Battery 1-1. However, in the event of a loss of vital 480 V Bus 1F, Battery Charger 121 could be manually aligned to restore power to direct current (dc) loads associated with Battery 1-1. In the event of a loss of vital 480 V Bus 1F, Battery 1-1 supplies the following risk significant loads: Diesel Engine Generator 1-2 engine panel emergency source, solid-state protection system Train A; main steam isolation valves and 10 percent steam dump circuitry, vital 120 V Inverter IY 1-1, and Diesel Engine Generator 1-3 normal gage panel power. Calculation 235A-DC, “Battery 11 Sizing, Load Flow, Voltage Drop, Short Circuit and Charger Sizing,” Revision 8, determined the minimum number of cells for each operable bank to be 59 cells.

On October 3, 2006, Battery 1-1, Cell 15 indicated a low voltage of 2.093 V during the performance of Surveillance Test Procedure STP M-11A, "Station Battery and Pilot Cell Condition Monitoring," Revision 19. The TS 3.8.4 limit for cell voltage is 2.07 V. The battery was subsequently placed on equalizing charge in an attempt to restore voltage. This attempt was not successful and an individual cell equalizing charge was commenced on October 11. On October 15, maintenance personnel installed an individual cell charger on Cell 15 as a temporary modification. The charger was adjusted to a float voltage charge of 2.25 V as a compensatory measure to ensure battery operability.

On October 16, the inspectors engaged PG&E staff regarding the operability of Battery 1-1, while an individual cell charger was installed to maintain Cell 15 at the battery average voltage. In the event of a station blackout, the degraded cell could adversely impact Battery 1-1 by becoming an additional load on the battery (i.e., reverse polarizing). Engineering staff assumed that, during a battery discharge, Cell 15 voltage would drop at the same rate as the other battery cells. Therefore, if the individual cell charger maintained Cell 15 voltage at the same voltage level as the overall battery, the cell would not become an additional load during battery discharge. However, the inspectors observed that engineering staff did not have analysis, test data, or other information to support the assumption. In discussions with the battery vendor (C&D Power Systems), engineering staff determined that it is not known at what point or how much, if any, that having a degraded cell would affect battery performance. According to ANSI/IEEE Standard 450-1987, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," there are specified methods to correct battery cell deficiencies. This industry standard stated that, if a cell is not recoverable through equalizing charges, it should be removed from service. The inspectors noted that the staff had not provided sufficient tests, analyses, or other evidence that supported the installation of a cell charger for battery operability.

On November 17, Cell 15 was isolated electrically from Battery 1-1 by installing jumper cables. This action restored Battery 1-1 to an analyzed, operable condition of 59 fully functional cells. PG&E waited approximately 33 days to install the jumper cables since they were waiting for a response on a license amendment request to extend the current TS 3.8.4 allowed outage time from 2 hours to 12 hours. On November 15, PG&E received a one-time license amendment to extend the allowed outage time to 4 hours. The evolution on November 17 took approximately 74 minutes, which was within the original allowed outage time of 120 minutes. The inspectors also noted that prior mock-ups of the jumper installation demonstrated that the evolution could be completed in less than the TS allowed outage time.

Analysis: The performance deficiency associated with this finding involved the failure of engineering personnel to properly implement measures to correct a condition adverse to quality. The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 worksheets, the inspectors determined that this finding is of very low safety significance because it did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the CAP in that engineering staff did not thoroughly assess the operability of the battery and correct a condition adverse to quality in a timely manner.

Enforcement: 10 CFR Part 50, Appendix B, Criterion XVI, requires, in part, that measures be established to assure that significant conditions adverse to quality, such as deficiencies, defective material and equipment, and nonconformances, are promptly identified and corrected. Contrary to this requirement, from October 15 to November 17, 2006, PG&E implemented a temporary modification that had not been adequately analyzed for its effect on battery operability and was contrary to committed industry standards. As a result, PG&E failed to correct a condition adverse to quality in a timely manner. This finding is of very low safety significance and has been entered into the CAP as AR A0678820. This violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 50-275/06-05-03, Inadequate Temporary Modification to a Vital Battery.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed key affected parameters associated with energy needs, materials/replacement components, timing, heat removal, control signals, equipment protection from hazards, operations, flowpaths, pressure boundary, ventilation boundary, structural, process medium properties, licensing basis, and failure modes for the one modification listed below. The inspectors verified that: (1) modification preparation, staging, and implementation did not impair emergency/abnormal operating procedure actions, key safety functions, or operator response to loss of key safety functions; (2) postmodification testing maintained the plant in a safe configuration during testing by verifying that unintended system interactions will not occur, SSC performance characteristics still meet the design basis, the appropriateness of modification design assumptions, and the modification test acceptance criteria has been met; and (3) PG&E has identified and implemented appropriate corrective actions associated with permanent plant modifications.

- September 29, 2006, Unit 1, Modification of Excess Letdown Heat Exchanger Outlet Valve Controller HCV-123 with a 60-ohm resistor

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the four below listed postmaintenance test activities of risk-significant systems or components. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly realigned, and deficiencies during testing were documented. The inspectors also reviewed the FSAR Update to determine if PG&E identified and corrected problems related to postmaintenance testing.

- September 26, 2006, Unit 1, Excess letdown Heat Exchanger Valve HCV-123
- October 3, 2006, Unit 1, Auxiliary Building Ventilation System, POV-1/POV-2

- October 4, 2006, Unit 1, Auxiliary Saltwater Unit Cross-tie Valve SW-0-FCV-601
- November 1, 2006, Unit 1, Safety Injection Pump 1-1

Documents reviewed by the inspectors included:

- Procedure LT 8-82, "CVCS Excess Letdown Heat Exchanger Outlet HCV-123 Calibration," Revision 1

- Procedure STP V-3F3, "Exercising Valve FCV-601, Units 1 and 2 ASW Cross-tie," Revision 16

- Procedure PEP 23-01, "Aux Building and Fuel Handling Building Ventilation Systems Fan Failure Tests and Miscellaneous POV Tests," Revision 8

- Procedure STP P-SIP-11, "Routine Surveillance Test of Safety Injection Pump 1-1," Revision 19

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the FSAR Update, procedure requirements, and TS to ensure that the two below listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumpers; (7) test data; (8) testing frequency and method demonstrated TS operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of American Society of Mechanical Engineers Code requirements; (12) updating of performance indicator data; (13) accuracy of engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria; (14) reference setting data; and (15) annunciators and alarm setpoints. The inspectors also verified that PG&E identified and implemented any needed corrective actions associated with the surveillance testing.

- November 1, 2006, Unit 1, Eagle-21 partial trip board activation test

- December 6, 2006, Unit 2, In-Service Testing of Auxiliary Feedwater Level Control Valve FW-2-LCV-113

Documents reviewed by the inspectors included:

- Procedure STP- I-36-S1EPT, “Protection Set 1 Eagle-21 Partial Trip Board Activation Test,” Revision 12

- Procedure STP V-3P6B, “Exercising Valves LCV-115 and 113 Auxiliary Feedwater Pump Discharge,” Revision 13

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the FSAR Update, plant drawings, procedure requirements, and TS to ensure that the one below listed temporary modification was properly implemented. The inspectors: (1) verified that the modifications did not have an affect on system operability/availability; (2) verified that the installation was consistent with modification documents; (3) ensured that postinstallation test results were satisfactory and that the impact of the temporary modifications on permanently installed SSCs were supported by the test; (4) verified that the modifications were identified on control room drawings and that appropriate identification tags were placed on the affected drawings; and (5) verified that appropriate safety evaluations were completed. The inspectors verified that PG&E identified and implemented any needed corrective actions associated with temporary modifications.

- November 16, 2006, Unit 1, Installation of jumper in Vital Battery 1-1

Documents reviewed by the inspectors included:

- AR A0678820
- Work Order C0207140

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1EP1 Exercise Evaluation (71114.01)

Cornerstone: Emergency Preparedness

a. Inspection Scope

The inspectors reviewed the objectives and scenario for the 2006 biennial emergency plan exercise to determine if the exercise would acceptably test major elements of the emergency plan. The scenario simulated a large condenser tube leak, with a subsequent failure of the reactor protection system to complete a reactor scram. Multiple main steam isolation valve failures then resulted in initiation of a small release of radioactivity to the environment. An initially small reactor coolant leak in containment greatly increased, ultimately resulting in a loss of reactor vessel level, core uncovering and damage of reactor fuel, with a rapid increase in the offsite release of radioactivity to the environment.

The inspectors evaluated exercise performance by focusing on the risk-significant activities of event classification, offsite notification, recognition of offsite dose consequences, and development of protective action recommendations, in the simulator control room and the following dedicated emergency response facilities:

- Technical Support Center
- Operations Support Center
- Emergency Operations Facility

The inspectors also assessed recognition of and response to abnormal and emergency plant conditions, the transfer of decision making authority and emergency function responsibilities between facilities, onsite and offsite communications, protection of emergency workers, emergency repair evaluation and capability, and the overall implementation of the emergency plan to protect public health and safety and the environment. The inspectors reviewed the current revision of the facility emergency plan, and emergency plan implementing procedures associated with operation of the above facilities and performance of the associated emergency functions. These procedures are listed in the Attachment to this report.

The inspectors compared the observed exercise performance to the requirements in the facility emergency plan, 10 CFR 50.47(b), 10 CFR Part 50, Appendix E, and to the guidance in the emergency plan implementing procedures and other federal guidance.

The inspectors attended the post-exercise critiques in each of the above facilities to evaluate the initial licensee self-assessment of exercise performance. The inspectors also attended a subsequent formal presentation of critique items to plant management.

The inspectors completed one sample during this inspection.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed licensee evaluations for the three emergency preparedness cornerstone performance indicators of Drill and Exercise Performance, Emergency Response Organization Participation, and Alert and Notification System Reliability, for the period October 1, 2005, through September 30, 2006. The definitions and guidance of NEI 99-02, "Regulatory Assessment Indicator Guideline," Revisions 3 and 4, and the licensee Emergency Plan Instruction 18, "Emergency Preparedness NRC Performance Indicators," 6/15/2006, were used to verify the accuracy of the licensee's evaluations for each performance indicator reported during the assessment period.

The inspectors reviewed a sample of drill and exercise scenarios and licensed operator simulator training sessions, notification forms, and attendance and critique records associated with training sessions, drills, and exercises conducted during the verification period. The inspectors reviewed selected emergency responder qualification, training, and drill participation records. The inspectors reviewed alert and notification system testing procedures, maintenance records, and a 100 percent sample of siren test records. The inspectors also reviewed other documents listed in the Attachment to this report.

The inspectors completed three samples during the inspection.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a daily screening of items entered into PG&E's CAP. This assessment was accomplished by reviewing ARs and event trend reports and attending daily operational meetings. The inspectors: (1) verified that equipment, human performance, and program issues were being identified by PG&E at an appropriate threshold and that the issues were entered into the CAP; (2) verified that corrective actions were commensurate with the significance of the issue; and (3) identified conditions that might warrant additional follow-up through other baseline inspection procedures.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-Up Inspection

a. Inspection Scope

In addition to the routine review, the inspectors selected the one below listed issue for a more in-depth review. The inspectors considered the following during the review of PG&E's actions: (1) complete and accurate identification of the problem in a timely manner; (2) evaluation and disposition of operability/reportability issues; (3) consideration of extent of condition, generic implications, common cause, and previous occurrences; (4) classification and prioritization of the resolution of the problem; (5) identification of root and contributing causes of the problem; (6) identification of corrective actions; and (7) completion of corrective actions in a timely manner.

- November 22, 2006, Units 1 and 2, Auxiliary Saltwater (ASW) Pump Shaft Packing

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure of engineering personnel to apply adequate design control measures. Specifically, engineering personnel changed the acceptance criteria in the ASW pump surveillance test from greater than zero packing leak-off to zero packing leak-off, with packing gland temperature less than 120°F. However, the change was based on engineering judgment versus analysis or test data. If the engineering judgment were incorrect, operability of the ASW pump could be impacted and the pump may be unable to perform its safety function due to potential shaft damage.

Description: The ASW system's safety function is to remove heat from the component cooling water heat exchangers using the ultimate heat sink (Pacific Ocean). The system consists of two pumps that take suction from the ocean and supply water to one or both heat exchangers. The ASW pumps utilize packing to limit the amount of water escaping the pump/shaft interface. The type of packing utilized in the ASW pumps is a Garlock Type 1304 graphite-based packing. The stuffing box for the ASW pumps utilizes a packing injection line, where the injection flow enters the box between an upper packing and a lower packing. A portion of the injection flow travels upward, between the pump shaft and the upper packing, and exits as visible packing leak-off at the top of the packing gland. The rest of the injection flow travels downward, between the pump shaft and the lower packing, and exits into the pump (i.e., nonvisible packing leak-off).

Operators observe evidence of visible packing leak-off on a daily basis to ensure that there is adequate cooling for the packing. Inadequate cooling to the packing may result in pump shaft seizure or damage to the packing gland and/or pump shaft. For more than 5 years, the ASW pump packing has been susceptible to sand and silt accumulation from ocean water, resulting in zero packing leak-off. When operations personnel found zero packing leak-off, the pump was declared inoperable and the pump was repacked. In all cases where zero packing leak-off was observed, the pump had just started versus when it had been running for some time.

Engineering personnel believed the ASW pumps were still operable without packing leak-off since the packing gland was cool to touch in such cases. Therefore, on February 9, 2006, engineering personnel developed a corrective action in AR A0657460 to modify the acceptance criteria in the ASW routine surveillance test procedures to allow zero packing leak-off as long as the packing gland temperature is less than 120°F. For example, the following procedures stated under Section 6, "Acceptance Criteria," that "pump packing leakage is greater than zero drops per minute, or stuffing box temperature is less than 120°F."

- STP P-ASW-11, "Routine Surveillance Test of Auxiliary Saltwater Pump 1-1," Revision 23

- STP P-ASW-12, "Routine Surveillance Test of Auxiliary Saltwater Pump 1-2," Revision 20

- STP P-ASW-21, "Routine Surveillance Test of Auxiliary Saltwater Pump 2-1," Revision 22

- STP P-ASW-22, "Routine Surveillance Test of Auxiliary Saltwater Pump 2-2," Revision 18

Prior revisions of these procedures required packing leakage to be greater than zero to pass the surveillance test and for the pump to be considered operable.

The inspectors reviewed AR A0657460 and the bases for the procedure change. The inspectors observed that licensee personnel based the change on engineering judgment, since there was no industry or vendor-specific criteria for an allowable stuffing box temperature. The inspectors also observed that Vendor Document 663030, Sheet 17, "Bingham Centrifugal Pumps - Instructions for Installation, Operation, and Maintenance," Revision 18, stated that a slight amount of leakage was needed to provide lubrication between the packing and the rotating element. If the leakage was cut off too much, the heat generated by the friction between the packing and the rotating element would destroy the packing and damage the rotating element. Engineering personnel felt that, with the stuffing box temperature less than 120°F, neither the shaft or packing would degrade and fail. The inspectors were concerned that there was no data or experience to show that the packing would remain at a constant temperature over time, particularly when the packing was full of sand and silt. Furthermore, the procedure change did not require temperature surveillance of the stuffing box on a periodic frequency if there was zero packing leak-off. Engineering personnel had determined that shift walkdowns of the ASW pumps performed twice a day by operators would be sufficient for monitoring. Again, there was no test data or experience to demonstrate that the surveillance frequency would be sufficient.

Analysis: The performance deficiency associated with this finding involved engineering personnel failing to apply design control measures to a routine surveillance test acceptance criteria that were commensurate with those applied to the original design. The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to be of very low safety significance, because it did not represent an actual loss of system safety function, did not represent an actual loss of a single train for greater than its TS allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with resources because engineering personnel failed to provide up-to-date design documentation to support a design change in surveillance test acceptance criteria.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design control measures shall be applied to items such as acceptance criteria for inspections and tests. Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to this, on February 9, 2006, engineering personnel failed to apply design control measures to a change in the routine surveillance test acceptance criteria that were commensurate with the original design. Specifically, engineering personnel modified the ASW pump routine surveillance test procedure acceptance criteria to allow zero packing leak-off, although original vendor documentation required greater than zero packing leak-off. Engineering personnel used judgment to justify the design change rather than vendor information or test results, which would have been commensurate with the original design. The corrective actions to restore compliance included obtaining vendor documentation to justify zero packing leak-off from the pumps. Because the finding is of very low safety significance and has been entered into PG&E's CAP as AR A0684631, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 50-275; 323/06-05-04, Inadequate Change to Auxiliary Saltwater Pump Routine Surveillance Test Acceptance Criteria.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely-related issues documented in ARs to identify trends that might indicate the existence of more safety-significant issues. The inspectors' review consisted of the 6-month period of July 1 to December 31, 2006. When warranted, some of the samples expanded beyond those dates to fully assess the issue. The inspectors also reviewed corrective actions associated with Emergency Action Level (EAL) Alert 2 and emergency core cooling system (ECCS) piping voids. The inspectors compared and contrasted their results with the results contained in PG&E's quarterly trend reports. Corrective actions associated with a sample of the issues identified in PG&E's trend reports were reviewed for adequacy. Documents reviewed by the inspectors are listed in the attachment.

b. Findings

.1 EAL Alert 2

Procedure EP G-1, “Emergency Classification and Emergency Plan Activation,” Revision 34, EAL Alert 2, describes the identification of fuel damage as shown by: “Confirmed Reactor Coolant System (RCS) sample > 300 μCi/cc of equivalent I-131 specific activity OR equivalent fuel failure is measured by exposure rate from systems carrying reactor coolant per EP RB-14A.” During the critique of the emergency drill on September 20, 2006, a concern was raised regarding the ability to determine the extent of possible fuel damage using Procedure EP RB-14A, “Initial Detection of Core Damage,” Revision 0. AR A0678189 was written to capture this concern.

The inspectors observed that several ARs had been written since 2004 (A0616739, A0670612, A0670613, and A0675097) regarding the ability to perform the requirements of EAL Alert 2 using Procedure EP RB-14A or through confirmation of an RCS sample > 300 μCi/cc of equivalent I-131 specific activity. Concerns included the high dose rates that would occur in the letdown heat exchanger room, where piping dose rates would be taken, and the lack of training for radiological protection personnel that would perform the procedure during an event. Other concerns involved the use of the postaccident sampling system for obtaining and analyzing RCS samples, since much of that system had been abandoned in place. Some ARs were still open and others had been closed without fully addressing the concerns.

PG&E stated that, in an event, Procedure RB-14A would be adequately briefed and preparations made for taking dose rates. Also, portions of the postaccident sampling system were maintained operable for taking RCS samples. However, it was not until the NRC began to research the issue that PG&E adequately addressed the concerns.

.2 ECCS Piping Voids

The inspectors reviewed Diablo Canyon Power Plant performance with respect to ECCS piping voids. Within the past 6 years, both units exhibited voids at ECCS suction piping that could adversely impact the ability of ECCS pumps to perform their safety function. A notable area of ECCS voids was the cross-over suction piping from the RHR system to the safety injection system. PG&E believed that the voiding was caused by hydrogen gas coming out of solution at the RCP seal injection system and migrating to this section of pipe. To eliminate concerns of ECCS pipe voids at this location, PG&E installed an approximate 40 gallon void chamber (i.e., tank) to that section of pipe for both units. Gas accumulating in that section of the pipe now travels up piping to the void chamber. During monthly ECCS void surveillances, operators vent the void chamber. Per design, the gas coming out of solution is removed from the piping and contained in the void chamber where it does not affect the performance of safety-related pumps. The inspectors reviewed the performance of the void chambers and found them to operate as designed, without any adverse impacts to the plant.

The inspectors reviewed other operating experience at Diablo Canyon Power Plant that may point to potential ECCS void concerns. The inspectors observed that a section of RHR discharge piping in both units have experienced voiding in the past. This section of discharge piping, Line 509, is located inside containment, but outside the bioshield. In April 2001, AR A0528837 documented the presence of a large void in Line 509 for Unit 1. The voiding was caused by leakage of nitrogen-saturated water from Accumulator 1-3 into the RHR discharge piping. Prior to this determination, operators noticed an increase in RHR discharge pressure by a factor of four and a declining level in Accumulator 1-3. To prevent the voiding from impacting operability of the RHR system at that time, PG&E maintained high pressure on the RHR discharge piping. Later, PG&E was able to correct the accumulator leakage and eliminate the void. In May 2006, while performing an extent of condition review for a water hammer that occurred on Accumulator 2-3 discharge piping, engineering personnel discovered small void pockets in Unit 1 Line 509. As documented in AR A0669488, the volume of the void was approximately 3 percent of the pipe area, which was less than the 20 percent limit as described in Calculation STA-108, "ECCS Pump Suction Void Evaluation," Revision 0. PG&E continued to monitor Line 509 for the next 6 months on Units 1 and 2 and were able to vent at a maximum of ½ cup of gas during that time period for both units.

The inspectors reviewed PG&E's actions in regard to the voiding found inside containment, including their monitoring efforts. The inspectors found that in both cases the voids would not be a water hammer concern or impede core cooling following a safety injection. The inspectors also found that PG&E's monitoring efforts were consistent with industry practices. Specifically, PG&E would verify the piping inside containment was full before restarting the reactor following a refueling outage. Additionally, PG&E monitored RCS leakage and accumulator levels and pressures to identify the potential for nitrogen or hydrogen gas coming out of solution in ECCS piping.

4OA3 Event Followup (71153)

.1 Unit 2 Reactor Shutdown Due to Indicated High Stator Temperature for RCP 2-2

a. Inspection Scope

On December 10, 2006, operators received alarms for Unit 2 RCP 2-2 high stator temperature. Using Procedures AR PK05-02, "RCP No. 22," Revision 20, and OP AP-28, "Reactor Coolant Pump Malfunction," Revision 2, operators responded to the alarm condition. Both procedures required a reactor trip if the RCP stator temperature exceeded 300°F. Operators determined that a reactor shutdown was necessary as indicated stator temperature was 249°F and increasing. Operators manually tripped the reactor when indicated stator temperature was 300°F. Just prior to the reactor trip, the reactor was subcritical with k_{eff} less than 0.99; however, operators had not been able to fully insert control rods as part of the reactor shutdown. Upon the manual reactor trip, all control rods inserted. PG&E investigated the cause of the high stator temperature alarm and identified a failed resistance temperature detector. An installed spare resistance temperature detector in the RCP 2-2 stator was selected and operators began actions to restart Unit 2 on December 11.

The inspectors responded at the time of the event and: (1) observed plant parameters and status, (2) evaluated performance of mitigating systems and operators, (3) confirmed that PG&E properly classified the event in accordance with EAL procedures and made timely notifications to the NRC and state/local governments, and (4) communicated the details of the events and conditions to NRC management as input to determining the need for additional inspection effort.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Trip Due to Electrical Fault Associated With CWP 2-1

a. Inspection Scope

On December 12, 2006, an electrical fault occurred in Unit 2 CWP 2-1. The apparent cause of the fault was the failure of an oil-filled surge capacitor located at the motor terminal leads. As a result of the electrical fault, voltage on 12 kV nonvital Bus D dropped to approximately 6.5 kV for less than a second. This bus supplies CWP 2-1, as well as RCPs 2-2 and 2-4. Relay devices on Bus D sensed the degraded voltage and opened the breakers for RCPs 2-2 and 2-4 per design. With two RCP breakers open, the coincidence logic in the reactor protection system was met for a reactor trip. Subsequently, the Unit 2 reactor tripped from 25 percent power.

The inspectors responded at the time of the event and: (1) observed plant parameters and status, (2) evaluated performance of mitigating systems and operators, (3) confirmed that PG&E properly classified the event in accordance with EAL procedures and made timely notifications to the NRC and state/local governments, and (4) communicated the details of the events and conditions to NRC management as input to determining the need for additional inspection effort.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 Onsite Fabrication of Components and Construction of an Independent Spent Fuel Storage Installation (ISFSI) (60853)

a. Inspection Scope

The inspectors witnessed portions of the ongoing ISFSI construction activities and evaluated PG&E's performance in accordance with the criteria contained in Inspection Procedure 60853. PG&E continued concrete placement activities for the cask transfer facility (CTF) and the second ISFSI pad. Concrete was placed for the CTF on August 9, 2006, and for the Transporter Seismic Anchors (TSAs) on September 7, 2006. Concrete placement activities were completed for the second ISFSI pad on August 23, 2006. The inspectors witnessed portions of the concrete placement and testing activities for the CTF, TSAs, and the second ISFSI pad.

Grouted anchors were installed as part of the TSA design on October 31, 2006, to securely anchor the transporter during a seismic event. The grouted anchors were tensioned on November 6, 2006. The inspectors witnessed portions of the anchor installation, grouting, and tensioning.

b. Findings

No findings of significance were identified.

.2 Temporary Instruction 2515/169, "Mitigating Systems Performance Index Verification (MSPI)"

a. Inspection Scope

The inspectors verified that PG&E correctly implemented the Mitigating Systems Performance Index (MSPI) guidance for reporting unavailability and unreliability of the monitored safety systems. Monitored safety systems included emergency diesel generators, RHR pumps, secondary heat removal pumps, cooling water pumps, and high head safety injection pumps. During the inspection, the inspectors assessed the following:

- Accurate documentation of the baseline planned unavailability hours for the MSPI systems
- Accurate documentation of the actual unavailability hours for the MSPI systems
- Accurate documentation of the actual unreliability information for each MSPI monitored component
- Significant errors in the reported data, which resulted in a change of the indicated index color

- Significant errors in the basis document which resulted in: (1) a change of the system boundary; (2) an addition of a monitored component; or (3) a change in the reported index color

This temporary instruction is complete for Units 1 and 2.

b. Findings

No findings of significance were identified.

.3 (Closed) URI 05000275, 323/2006012-01, Oil Found in the Vicinity of RHR Pumps

In response to inspectors identifying the presence of oil in the vicinity of the drain plugs of the motors for RHR Pumps 1-1, 2-1, and 2-2, PG&E monitored leakage from the motors during subsequent pump runs and concluded that the RHR pumps would have remained operable for their mission times. Additionally, PG&E staff evaluated the use of a shortened cure time on the RHR pump motor oil drain plugs and determined that the sealing capability of the plugs had not been adversely affected. These evaluations were reviewed by the inspectors, no findings of significance were identified, and no violations of NRC requirements were identified. PG&E documented the evaluations for presence of the oil and the shortened sealant cure time in ARs A0670760 and A0675763. This URI is closed.

.4 (Closed) URI 07200026/2006-01, Review the concrete compressive strength test results to confirm that the concrete compressive strength of the CTF basemat and initial ISFSI pad meet the specified compressive strength of 5,000 psi at 90 days.

During the concrete placement activities for the Diablo Canyon ISFSI, several minor deviations were identified by the inspectors and those were not considered to be safety significant as documented in NRC Inspection Report 05000275/2006008; 05000323/2006008; 07200026/2006001. The determination that the deviations were not safety significant was based on the expectation that the concrete would meet or exceed the required minimum compressive strength of 5,000 psi at the specified cure period of 90 days. URI 07200026/2006-01 was opened to track and confirm that the minimum concrete compressive strength was achieved.

PG&E transmitted the 90-day concrete compressive strength test results for the initial ISFSI pad and the CTF basemat to the NRC on October 12, 2006. The 90-day concrete compressive test results varied from a low of 7,420 psi to a high of 9,060 psi. The arithmetic average of any three consecutive strength tests exceeded the required minimum concrete compressive strength of 5,000 psi as required by Section 5.6.2.3 of ACI 349, “Code Requirements for Nuclear Safety Related Concrete Structures.” There were no maximum concrete compressive strength limitations associated with the Diablo Canyon ISFSI. Based on satisfactory concrete compressive strength results for the initial ISFSI pad and the CTF basemat, URI 07200026/2006-01 is considered closed.

.5 (Closed) URI 05000323/2006004-03, Corrective Actions Regarding RCS Leakage Through In-core Thimble Tube

a. Inspection Scope

On August 31, 2006, an approximate 1.5 gpm RCS leak occurred in Unit 2. The location of the leakage was in the movable in-core detector system (MIDS) L-13 thimble tube. The inspectors responded to the RCS leakage at the time of the event and observed operator actions to identify, quantify, and mitigate the leakage. As discussed in NRC Inspection Report 05000275; 323/2006004, the inspectors found that PG&E staff took appropriate action in response to the leakage.

PG&E initiated Nonconformance Report (NCR) N0002211 to identify the root cause(s) of the leakage and ensure that appropriate corrective actions were taken. Following the RCS leakage on August 31, maintenance personnel began activities to repair the MIDS, which had been damaged by water. On September 6, maintenance personnel noticed leakage from the Path L-13 manual isolation valve at the threaded connection on the high pressure side. The leakage was determined to be 4 to 6 drops per minute. Maintenance personnel initially installed a freeze seal to isolate the leakage, and then tighten the threaded connection three flats to stop the leakage. PG&E subsequently continued with repair activities on the MIDS.

URI 05000323/2006004-03 was initiated in NRC Inspection Report 05000275; 323/2006004 in order to: (1) evaluate PG&E's root cause and corrective actions following the RCS leakage and (2) evaluate PG&E's response to the leak on the threaded connection on the high pressure side of the Path L-13 manual isolation valve. The inspectors have completed the necessary actions to evaluate these two aspects. Therefore, URI 05000323/2006004-03 is closed.

b. Findings

1. Failure to Preserve Corrective Action for Thimble Tube Wear

Introduction: A self-revealing, Green NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for the failure to apply adequate design control measures regarding the installment of thimble tubes with chrome-plated bands. Specifically, PG&E installed thimble tubes with chrome-plated bands at the fuel assembly bottom nozzle/lower core plate interface to address flow-induced vibration wear. Due to engineering personnel's failure to account for the chrome-plated bands in the thimble tube relocation procedure, the chrome-plated band on Thimble Tube L-13 was removed from its designed location at the fuel assembly bottom nozzle, thereby increasing the potential for thimble tube through-wall wear.

Background: Westinghouse pressurized-water reactors utilize MIDS to monitor nuclear power distribution within the reactor core. MIDS consists of a detector that travels inside the thimble tubes, with the thimble tubes inserted into select fuel assemblies. The thimble tubes are constructed of 316 stainless steel with a typical outer diameter of 0.3 inches and a wall thickness of 0.049 inches. The thimble tubes are flexible to allow withdrawal and re-insertion into fuel assemblies during reactor refueling activities. Inside the reactor vessel, the thimble tubes are supported by the fuel assembly instrument tube and the lower reactor internals guide columns. The thimble tubes enter the reactor vessel at the bottom of the vessel and are supported outside the reactor vessel by permanently-mounted thimble guide tubes. The permanently-mounted thimble guide tubes are welded at the bottom of the reactor vessel and stop at the in-core seal table, where a high pressure seal prevents reactor coolant from traveling between the thimble tube and thimble guide tube out of the RCS. The thimble tube connects to a selector device that directs the detector into the selected thimble tube. Since the thimble tube and thimble guide tube are subjected to RCS pressure, they are considered an extended portion of the RCS pressure boundary.

Description: On August 31, 2006, an approximate 1.3 gpm RCS leak occurred in the Unit 2 MIDS L-13 thimble tube. NCR N0002211 identified a presumed root cause and developed corrective actions to prevent recurrence. NCR N0002211 stated that the RCS leak was due to accelerated wear/wear-induced fatigue of the L-13 thimble tube due to flow-induced vibration in combination with multiple wear scars in a short length of tube, which was not prevented by the thimble tube wear management program. Until the thimble tube is removed for examination in the next refueling outage in early 2008, PG&E staff has classified the root cause as presumed and not definite.

The inspectors reviewed operating experience associated with thimble tubes at the Diablo Canyon Power Plant and in the nuclear industry. In 1987, NRC Information Notice 87-44, "Thimble Tube Thinning in Westinghouse Reactors," was issued to discuss thimble tube wall thinning due to flow-induced vibration on the exposed portion of thimble tubes at the bottom of fuel assemblies and the lower core plate. In 1988, NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," requested that licensees establish an inspection program to detect thimble tube wall thinning. For Unit 1, PG&E responded to NRC Bulletin 88-09 via PG&E Letter DCL-89-280, dated November 10, 1989. In the letter, PG&E noted that they found 20 of 58 thimble tubes exceeded the 60 percent through-wall degradation limit, with other tubes experiencing lesser degrees of damage. PG&E subsequently replaced 28 thimble tubes and repositioned the wear point on 17 other tubes by pulling up at least 1.5 inches on the seal table end of the tube and cutting off the excess tube at the high pressure seal. For Unit 2, PG&E responded in PG&E Letter DCL-90-094, dated April 4, 1990, with the results of the Unit 2 thimble tube inspection. PG&E found that 3 tubes, including Thimble Tube L-13, exceeded the acceptable wear limit, and they were capped for future replacement. In 4 other tubes, PG&E repositioned the wear points and kept the tubes in service, since the tube wall degradation did not exceed the acceptable wear limit.

The inspectors found that PG&E continued to inspect the thimble tubes each refueling outage using eddy current techniques. Thimble Tube L-13 remained capped and out of service until Refueling Outage 2R10 (May 2001), when it was replaced. The new thimble tube included a 16-inch long chrome band to protect the thimble tube from flow-induced vibration at the fuel assembly/lower core plate interface. In Refueling Outage 2R11 (February 2003), Thimble Tube L-13 showed 16 percent wear at the upper tie plate area (located in the reactor vessel lower internals). Since the wear was deemed acceptable per Procedure STP R-22, “Thimble Tube Inspection Program,” Revision 5, PG&E did not perform any corrective actions on the tube. In Refueling Outage 2R12 (November 2004), the wear on Thimble Tube L-13 indicated 46 percent at the upper tie plate area, and maintenance personnel repositioned the tube by 5 inches. In Refueling Outage 2R13 (May 2006), the wear on Thimble Tube L-13 again showed 46 percent through-wall, and maintenance personnel repositioned the tube by another 5 inches. Thimble Tube L-13 began leaking approximately 3 months later.

NCR N0002211 noted that, when maintenance personnel repositioned Thimble Tube L-13 the second time in Refueling Outage 2R13, the tube had been repositioned such that the chrome-plated band on the thimble tube was no longer in its designed location. As a result of thimble tube wall-thinning discovered in 1989 and 1990, NCR N0001325, Corrective Action to Prevent Recurrence 3, called for implementation of a design change to eliminate or greatly reduce flow-induced thimble tube wear rates. Westinghouse had developed thimble tubes with chrome-plated bands which would be more resistant to flow-induced thimble tube wear. PG&E installed these thimble tubes in Refueling Outage 1R6 for Unit 1 and Refueling Outage 2R10 for Unit 2. When the new thimble tube was installed at the L-13 position, the chrome-plated band was 16 inches long, with approximately 9 inches above the fuel assembly bottom nozzle and 7.5 inches below the lower core plate. The chrome-plated band was placed in this location since operating experience had shown the fuel assembly bottom nozzle/lower core plate interface to be a high wear-rate area. As discussed in AR A0205526, engineering personnel observed that the chrome-plated thimble tubes were an approved replacement part provided by Westinghouse. Therefore, engineering staff did not implement a design change for the new thimble tubes, but did have the drawings updated to reflect the chrome band on the thimble tubes. Subsequently, Procedure STP R-22 was not revised in order to restrict the amount of reposition for the chrome-plated thimble tubes. Therefore, when maintenance personnel repositioned Thimble Tube L-13 in Refueling Outage 2R13, the tube had been repositioned a total of 10 inches during its life. The final reposition moved a section of nonchrome-plated thimble tube area to the fuel assembly bottom nozzle/lower core plate interface.

<u>Analysis</u>: The performance deficiency associated with this finding involved engineering personnel failing to apply adequate design control measures, which would have included necessary procedure changes to reflect the use of thimble tubes with chrome bands. The finding is greater than minor because it is associated with the Initiating Events Cornerstone attribute of design control and affects the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown

as well as power operations. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because, assuming the worst-case degradation, the finding would not result in exceeding the TS limit for identified RCS leakage or affect mitigating systems. Specifically, the inspectors verified the worst-case leakage, i.e. guillotine break, from a thimble tube at the fuel assembly bottom nozzle/lower core plate interface to be approximately 7 gpm versus the TS RCS identified leakage limit of 10 gpm. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the CAP because PG&E disabled a corrective action to prevent recurrence of significant thimble tube wear.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design. Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to this, on April 24, 2006, engineering personnel removed a corrective action to prevent significant thimble tube wear when they repositioned Thimble Tube L-13 such that the tube was no longer in its intended location. Specifically, Thimble Tube L-13 was repositioned to the extent that its chrome-plated band was no longer effectively covering the fuel assembly bottom nozzle and lower core plate interface zone. The cause of the violation was the failure of engineering personnel to update applicable procedures with information that would prevent the chrome-plated bands from being removed from their designed location during repositioning of thimble tubes. The corrective actions to restore compliance included actions to update the applicable procedures to provide repositioning limits on thimble tubes with chrome-plated bands and to utilize thimble tubes with chrome-plated bands for the entire length of tube inside the reactor vessel. Because the finding is of very low safety significance and has been entered into PG&E’s CAP as NCR N0002211, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 50-323/06-05-05, Failure to Update Relocation Procedure for Thimble Tube Chrome Band.

4OA6 Management Meetings

Exit Meeting Summary

On October 27, 2006, the lead inspector presented the results of the biennial emergency preparedness exercise inspection to Ms. D. Jacobs, Vice President, Nuclear Services, and other members of her staff, who acknowledged the findings. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On December 20, 2006, the inspectors presented the findings of the inspection on biennial heat sink performance to Mr. L. Parker and other members of PG&E management. PG&E acknowledged the inspection findings.

The resident inspection results were presented on January 10, 2007, to Ms. D. Jacobs, Vice President, Nuclear Services, Diablo Canyon and other members of PG&E management. PG&E acknowledged the findings presented.

The inspectors asked PG&E whether any materials examined during the inspection should be considered proprietary. Proprietary information was reviewed by the inspectors and left with PG&E at the end of the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

PG&E Personnel

J. Becker, Vice President - Diablo Canyon Operations and Station Director
D. Burns, Operations Training Supervisor
C. Dougherty, Regulatory Services Senior Engineer
S. David, Operations Director
D. Fried, Emergency Planning Coordinator
M. Ginn, Emergency Planning Coordinator
J. Haynes, Training manager
J. Haynes, Licensing Services Manager
R. Hite, Manager, Radiation Protection
D. Jacobs, Vice President - Nuclear Services
S. Ketelsen, Manager, Regulatory Services
K. Langdon, Director, Operations Services
M. Lemke, Emergency Planning Principal
M. Meko, Director, Site Services
C. Over, Regulatory Services Supervisor
L. Parker, Regulatory Services Supervisor
K. Peters, Director, Engineering Services
J. Purkis, Director, Maintenance Services
P. Roller, Director, Performance Improvement
D. Taggart, Manager, Quality Verification
B. Terrell, Emergency Planning Supervisor
R. Waltos, Manager, Emergency Preparedness
M. Zawalick, Emergency Services, Senior Coordinator
S. Zawalick, Regulatory Services Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-275; 323/06-05-01	URI	Additional Review of Material to Determine if the RHR Heat Exchangers Will Meet Their Safety Function (Section 1R07.2)

Opened and Closed

50-275; 323/06-05-02	NCV	Failure to Adequately Evaluate Operability of Auxiliary Building Ventilation Control Panels (Section 1R13)
50-275/06-05-03	NCV	Inadequate Temporary Modification to a Vital Battery (Section 1R15)

50-275; 323/06-05-04	NCV	Inadequate Change to Auxiliary Saltwater Pump Routine Surveillance Test Acceptance Criteria (Section 4OA2.2)
50-323/06-05-05	NCV	Failure to Preserve Corrective Action for Thimble Tube Wear (Section 4OA5.5)

Closed

50-275; 323/06-12-01	URI	Oil Found in the Vicinity of Residual Heat Removal Pumps (Section 4OA5.3)
72-026/06-01-01	URI	Review the Concrete Compressive Strength Test Results to Confirm that the Concrete Compressive Strength of the Cask Transfer Facility Basemat and Initial Independent Spent Fuel Storage Installation Pad Meet the Specified Compressive Strength of 5,000 psi at 90 days (Section 4OA5.4)
50-323/06-04-03	URI	Corrective Actions Regarding RCS Leakage Through In-core Thimble Tube (Section 4OA5.5)

LIST OF DOCUMENTS REVIEWED

Section 1R05: Fire Protection

Procedures

Number	Title	Revision
A-6	Equipment Inspections	5
OM8.ID4	Control of Flammable and Combustible Materials	14
STP M-69A	Monthly Fire Extinguisher Inspection	36
STP M-69B	Monthly CO2 Hose Reel and Deluge Valve Inspection	14
STP M-70C	Inspection/Maintenance of Doors	10

Section 1R07: Biennial Heat Sink Performance (71111.07B)

Action Requests

A0341604	A0556717	A0600918	A0608887	A0617092	A0648128
A0391927	A0588366	A0608886	A0608888	A0640406	A0653252
A0380732	A0592857				

Calculations

Number	Title	Revision
M-938	CCW Data Input for 1993 Containment Analysis Program	3
M-1017	Determine Flows in the CCW System	3

Procedures

Number	Title	Revision
STP M-93A	Refueling Interval Surveillance - Containment Fan Cooler	18
STP M-13A	CCW Flow Balancing	15
BIO D-5	Microfouling Sample Collections in Component Cooling Water Heat Exchangers	0A
CAP O-6	Chemical Additions to the Closed Cooling Water Systems	16A
OP F-5:III	Chemistry Control Limits and Action Guidelines for the Plant Support Systems	18
PEP M-200	Flow Balancing CCW to Equipment on CCP Pump Skid	0

Miscellaneous Documents

Title	Date/Revision
DCM No. S-23A, “Design Criteria Memorandum Containment HVAC System	18C
DCM No. S-10, “Design Criteria Memorandum S-10 Residual Heat Removal System	13D

Title	Date/Revision
DCM No. S-14, “Design Criteria Memorandum Component Cooling Water System	15E
Nonconformance Report N0002194, “Void at CCP - SIP Suction Cross Tie	0
PG&E letter DCL 90-027	Jan. 26, 1990
Westinghouse Letter PGE 96-605	Sept. 3, 1996

Work Orders

C0193402	R0234221	R0258879	R0259792	R0259790	R0259627
R0259787	R0269129	R0269164	R0234221	R0244369	R0244367

Section 1R12: Maintenance Effectiveness (71111.12)

Action Requests

A0678838	A0681428	A0681464

Procedures

Number	Title	Revision
MP E-53.7	Maintenance of ITT, General Controls Hydramotor Actuator	9A
AD13.ID4	Post-Maintenance Testing	14

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

Action Requests

A0610558	A0636037	A0637526	A0642790	A0660739	A0678429
A0678436	A0681559	A0683008	A0684812		

Procedures

Number	Title	Revision
A-29	Protected Train Restrictions	3

Number	Title	Revision
AD7.DC6	On-line Maintenance Risk Management	9
MA1.DC10	Troubleshooting	9
MA1.DC11	Risk Assessment	7
OP J-2:VIII	Guidelines for Reliable Transmission Service for DCPP	10
OP1.DC17	Control of Equipment Required by the Plant Technical Specifications or Other Designated Programs	11
OM7.ID12	Operability Determination	9
OP H-1:I	Auxiliary Building Ventilation System - Make Available and System Operation	9

Work Requests

C0202499 C0206797 C0208219

Section 1R15: Operability Evaluations (71111.15)

Action Requests

A0678820 A0680025 A0681006 A0682008

Calculations

Number	Title	Revision
235A-DC	Battery 11 Sizing, Load Flow, Voltage Drop, Short Circuit and Charger Sizing	8
369-DC	Vital 125 VDC System Calculation for PRA System Analysis (Station Blackout)	1

Drawings

Number	Title	Revision
437546	Single Line Meter and Relay Diagram 125 Volt DC System	39
445075	Single Line Meter and Relay Diagram 125 Volt DC System	156

445076	Single Line Meter and Relay Diagram 125 Volt DC System	15
050024	List of Electrical Devices for Protection and Control Circuits	47
D-INV-4-88598	Discharge Characteristics LCUN-33	1

Procedures

Number	Title	Revision
MP E-67.6	Station Battery Preventive Maintenance	6
OP O-2	Operation of Hagan Controllers	10
STP M-11A	Station Battery and Pilot Cell Condition Monitoring	20
STP M-11B	Station Battery Condition Monitoring	26
STP M-21A	Vital Station Battery Modified Performance Test	12

Work Requests

R0233847 R0213290

Miscellaneous Documents

Title	Date/Revision
Amendment to Facility Operating License, Amendment No. 190, License DPR-80	Nov. 15, 2006
ASCO Letter, “Potential Non-Conformances of Plunger Tubes Used In Certain NH Series Hydramotor Pump and pump Kits”	Sept. 15, 2006
BCT-2000 Battery Load Test Report, “Modified Performance Test Vital Battery 1-1"	Oct. 27, 2005
BCT-2000 Battery Load Test Report, “Modified Performance Test Non-Vital Battery 2-5"	May 12, 2006
DCM No. T-42, “Station Blackout”	9

Title	Date/Revision
IEEE Standard 535-1986, “Qualification of Class 1E Lead Storage Batteries”	
Operations Shift Orders	Oct. 17, 2006
Regulatory Guide 1.155, “Station Blackout”	Aug. 1988
PG&E Letter DCL-06-120, “Exigent License Amendment Request 06-08 Revision to Technical Specification 3.8.4, “DC Sources-Operating,” Condition B	Oct. 18, 2006
PG&E Letter DCL-06-126, “Supplement to Exigent License Amendment Request 06-08 Revision to Technical Specification 3.8.4, “DC Sources-Operating,” Condition B	Oct. 18, 2006

Section 1R17: Permanent Plant Modifications (71111.17)

Action Requests
A0678267

Documents

IB-129-154, Westinghouse Instruction bulletin, “Remote/Manual Setpoint Station,” dated November 1968

Drawings

Number	Title	Revision
102032	Excess Letdown Heat Exchanger Outlet Loop Block Diagram, Sheet 26	148
102032	Excess Letdown Heat Exchanger Outlet Loop Block Diagram, Sheet 31	116
106725	Unit 1 Containment	109
448928	Electrical Diagram Connections Unit 1 Vertical Control Board 1VB2	31
437930	Electrical Diagram Of Connections Penetration No. 24E	16

Work Requests

C0206720

Section 1EP1: Emergency Plan Implementing Procedures (71114.01)

EP G-1	Revision 34	“Emergency Classification and Emergency Plan Activation”
EP G-2	Revision 31	“Interim Emergency Response Organization”
EP G-3	Revision 47	“Emergency Notification of Off-Site Personnel”
EP G-4	Revision 22	“Assembly and Accountability”
EP G-5	Revision 9A	“Evacuation of Nonessential Personnel”
EP OR-3	Revision 6B	“Emergency Recovery”
EP RB-1	Revision 5B	“Personnel Dosimetry”
EP RB-2	Revision 5	“Emergency Exposure Guides”
EP RB-3	Revision 5	“Stable Iodine Thyroid Blocking”
EP RB-4	Revision 4A	“Access to and Establishment of Controlled Areas Under Emergency Conditions”
EP RB-5	Revision 6	“Alternate Personnel Decontamination Facilities”
EP RB-8	Revision 19	“Instructions to Field Monitoring Teams”
EP RB-9	Revision 11A	“Calculation Release Rate”
EP RB-10	Revision 12	“Protective Action Recommendations”
EP RB-11	Revision 12	“Emergency Offsite Dose Calculations”
EP RB-12	Revision 6	“Plant Vent Iodine and Particulate Sampling During Accident Conditions”
EP RB-14	Revision 8A	“Core Damage Assessment Procedure”
EP RB-14A	Revision 0	“Initial Detection of Core Damage”
EP RB-15	Revision 11	“Post Accident Sampling System”
EP RB-16	Revision 0	“Operating Instructions for the EARS Computer Program”
EP R-2	Revision 23	“Release of Airborne Radioactive Materials Initial Assessment”
EP R-3	Revision 8C	“Release Of Radioactive Liquids”
EP R-7	Revision 15A	“Off-Site Transportation Accidents”
EP EF-1	Revision 33A	“Activation and Operation of the Technical Support Center”
EP EF-2	Revision 28	“Activation and Operation of the Operations Support Center”
EP EF-3	Revision 26A	“Activation and Operation of the Emergency Operations Facility”
EP EF-4	Revision 15	“Activation of the Off-Site Emergency Laboratory”
EP EF-9	Revision 10	“Backup Emergency Response Facilities”
EP EF-10	Revision 8	“Activation and Operation of the Joint Media Center”
TQ1	Revision 3	“Personnel Training and Qualification”
TQ1.ID3	Revision 5	“Non-accredited Training Program Management”
OM10.ID4	Revision 7	“ERO Management”
OM10.DC2	Revision 4	“ERO On-Call”

Exercise Evaluation Reports:

Drill Reports for 2004 and 2002 Biennial NRC Evaluated Exercise
All full-scale Exercise reports conducted in 2005-2006

Summary List of Drill/Exercise Evaluation related Condition Reports, July 2004 through September 2006

Emergency Plan, Revision 45

Section 4OA2: Problem Identification and Resolution (71152)

Action Requests

A0528837	A0631582	A0657460	A0669488	A0680274	A0685913
A0535731	A0636681	A0661341	A0672126	A0680773	
A0558389	A0636984	A0664245	A0679011	A0684631	

Procedures

Number	Title	Revision
STP M-89 (Unit 1)	ECCS System Venting	46
STP P-ASW-11	Routine Surveillance Test of Auxiliary Saltwater Pump 1-1	23
STP P-ASW-12	Routine Surveillance Test of Auxiliary Saltwater Pump 1-2	20
STP P-ASW-21	Routine Surveillance Test of Auxiliary Saltwater Pump 2-1	22
STP P-ASW-22	Routine Surveillance Test of Auxiliary Saltwater Pump 2-2	18

Calculations

Number	Title	Revision
STA-108	ECCS Pump Suction Void Evaluation	0

Miscellaneous Documents

Title	Date
Licensing Position - Definition of Full and Accessible for Performance of Technical Specification (TS) Surveillance Requirement (SR) 3.5.2.3 (Revision 1)	Sept. 4, 2001

Title	Date
NRC Information Notice 97-40, "Potential Nitrogen Accumulation Resulting From Backleakage From Safety Injection Tanks"	June 26, 1997
Vendor Document DC 663030-17-8, Page 69, Auxiliary Saltwater Pumps	Dec. 24, 1969

Section 4OA1: Emergency Implementing Procedures (71151)

OM10.DC1, "Emergency Preparedness Drills and Exercises," Revision2A

AWP EP-001, "Emergency Preparedness Performance Indicators," Revision 5

EP G-3, "Emergency Notification of Off-Site Agencies," Revision 43, Attachment 6.1, "Instructions for the DCPP Emergency Notification Form"

EP R-2, "Release of Airborne Radioactive Materials Initial Assessment," Revision 23

EP RB-10, "Protective Action Recommendations," Revision 11

Section 4OA3: Event Followup (71153)

Action Requests

A0132116	A0165736	A0342551	A0665509	A0684192	A0685056
A0164337	A0205526	A0623606	A0684170	A0684536	A0686189

Drawings

Number	Title	Revision
441227	Single Line Meter & Relay Diagram 12 kV System Bus Section "D" & "E"	21
441284	Schematic Diagram - Circulating Water Pumps	31
441338	Schematic Diagram - Bus Potential and Synchronizing 12 kV System	16
441350	Schematic Diagram - 12 kV Bus Sections D & E Automatic Transfer	8

Procedures

Number	Title	Revision
AR PK05-02	RCP No. 22	20
EP G-1	Emergency Classification and Emergency Plan Activation	34
EP G-3	Emergency Notification of Off-Site Agencies	47
NDE ET-2	Eddy Current Examination of Heat Exchanger Tubing	5
OP AP-28	Reactor Coolant Pump Malfunction	2
STP R-22	Thimble Tube Inspection Program	5/6

Miscellaneous Documents

Title	Date
DC-663102-24-2, “Westinghouse Installation - Operation - Maintenance Instructions for Types SSV-T and SSC-T Relays for Class 1E Application”	Jan. 4, 1985
Event Investigation Report 2006-001, “Unit 2 Reactor Trip and Unusual Event Due to CWP 2-1 Fault”	Dec. 13, 2006
Event Notification 42822	Aug. 31, 2006
Event Notification 43042	Dec. 10, 2006
Event Notification 43047	Dec. 12, 2006
Licensee Event Report 93-007-00, “Manual Reactor Trip Initiated Due to High Stator Temperature on Reactor Coolant Pump”	June 26, 1993
Nonconformance Report (NCR) DC1-89-TN-N096/2, “Incore Thimble Tubes” (also titled NCR N0001325)	Jan. 30, 1991

Title	Date
NCR N0002211, “Root Cause Analysis Report - RCS Leak Through MIDS Thimble Tube”	Oct. 18, 2006
NRC Bulletin 88-09, “Thimble Tube Thinning in Westinghouse Reactors”	July 26, 1988
NRC Information Notice 87-44, “Thimble Tube Thinning in Westinghouse Reactors”	Sept. 16, 1987
PG&E Letter DCL-88-208, “Thimble Tube Thinning in Westinghouse Reactors”	Aug. 26, 1988
PG&E Letter DCL-89-280, “Thimble Tube Thinning in Westinghouse Reactors”	Nov. 10, 1989
PG&E Letter DCL-89-292, “Thimble Tube Thinning Due to Flow-Induced Vibration”	Nov. 20, 1989
PG&E Letter DCL-90-094, “Thimble Tube Thinning”	April 4, 1990
PG&E Memorandum File No. 96, “Licensing Position - TS 3.4.13 Application for Threaded Connection Leakage”	Oct. 5, 2006
WCAP-12866, “Bottom-Mounted Instrumentation Flux Thimble Wear”	Jan. 1991
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LIST OF ACRONYMS

ac	alternating current
ADAMS	agency document and management system
ALARA	As Low As is Reasonably Achievable
ANSI/IEEE	American National Standards Institute/Institute of Electrical and Electronics Engineers
AR	action request
ASW	auxiliary saltwater
CAP	corrective action program
CCW	component cooling water
CFR	*Code of Federal Regulations*
CFCU	containment fan cooling unit
CTF	cask transfer facility
CWP	circulating water pump
dc	direct current
EAL	emergency action level
ECCS	emergency core cooling system
FSAR	Final Safety Analysis Report
ISFSI	Independent Spent Fuel Storage Installation
LER	licensee event report
MIDS	movable in-core detector system
MSPI	Mitigating Systems Performance Index
NCR	nonconformance report
NCV	noncited violation

NRC	Nuclear Regulatory Commission
PG&E	Pacific Gas and Electric Company
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal
SSC	structure, system, and component
TS	Technical Specifications
TSA	transporter seismic anchor
URI	unresolved item