UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
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February 5, 2008

John S. Keenan
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and Chief Nuclear Officer
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SUBJECT: DIABLO CANYON POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000275/2007005; 05000323/2007005 AND 07200026/2007001

Dear Mr. Keenan:

On December 31, 2007, 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 9, 2008, with John Conway 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 license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

There was one NRC-identified finding of very low safety significance (Green) and one Severity Level IV violation 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 two 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
72-026
Licenses: DPR-80
DPR-82
SNM-2511

Enclosure:
NRC Inspection Report 05000275/2007005, 05000323/2007005,
and 07200026/2007001
w/attachment: Supplemental Information

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SUNSI Review Completed: __VGG__ ADAMS: ☒ Yes ☐ No Initials: _VGG_
☒ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive
R:\REACTORS_DC\2007\DC2007-05-MSP-VGG.wpd ADAMS ML080360630

RIV:RI:DRP/B	SRI:DRP/B	C:DRS/EB1	C:DRP/PS	C:DRS/OB
MABrown	MSPeck	RLBywater	MPShannon	RELantz
/RA VGG for/	/RA VGG for/	*/RA/*	*/RA/*	*/RA/*
01/ 31/08	01/31/08	01/17/08	01/22/08	01/17/08
C:DRS/EB2	DNMS	DNMS	C:DRP/B	
LJSmith	RLKellar	BSpitzberg	VGGaddy	
/RA NFO for/	/RA DBS for/	/RA/	/RA/	
01/31/08	01/29/08	01/29/08	02/5/08	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-275, 50-323, 72-026
Licenses:	DPR-80, DPR-82, SNM-2511
Report:	05000275/2007005 05000323/2007005 07200026/2007001
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, 2007
Inspectors:	M. Peck, Senior Resident Inspector M. Brown, Resident Inspector K. Clayton, Senior Operations Engineer D. Stearns, Health Physicist, Plant Support Branch R. Kellar, Health Physicist
Approved By:	V. G. Gaddy, Chief, Projects Branch B Division of Reactor Projects

CONTENTS

SUMMARY OF FINDINGS

IR 05000275/2007-005, 05000323/2007-005; 10/1/07 - 12/31/07; Diablo Canyon Power Plant Unit 1; Surveillance Testing, Identification and Resolution of Problems.

This report covered a 13-week period of inspection by resident inspectors and announced inspections on operator licensing and radiation protection. One NRC-identified, Green, noncited violation and one NRC-identified Severity Level IV noncited violation 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: Mitigating Systems

- Green. The inspectors identified a noncited violation of 10 CFR 50, Appendix B, “Corrective Action,” after Pacific Gas and Electric failed to identify a degraded emergency diesel generator. On October 15, 2007, the inspectors identified a buildup of black soot on the Emergency Diesel Generator 1-1 exhaust manifold. Licensee personnel subsequently identified that one of the four fasteners connecting the exhaust manifold to the turbo charger was missing. The licensee declared the diesel generator inoperable based on the potential reduction of electrical power output due to exhaust gas bypassing the turbo charger and the adverse affect of the missing fastener on seismic qualification. Plant operators determined that overall plant risk was significantly degraded (Orange) due to the combination of the unavailable diesel generator and other plant equipment removed from service at the time. The licensee had prior opportunity to identify the degraded diesel generator during operator rounds between September 23 and October 15, 2007.

  This finding is greater than minor because, if left uncorrected, continued failure to perform adequate operator rounds would become a more significant safety concern. This finding affected the mitigating systems cornerstone because the issue involved an emergency diesel generator. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 worksheet, this finding was determined to have very low safety significance because it did not result in a loss of operability of a single train, for greater than Technical Specification allowed outage time, did not result in the loss of safety function, and was not potentially risk significant from a seismic, flooding or severe weather perspective. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because plant operators did not maintain a low threshold for identifying issues P.1(a). This issue was entered into the licensee’s corrective action program as Action Request A0710082 (Section 4OA2.1).

- SL IV. The inspectors identified a noncited Severity Level IV violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate safety evaluation of Unit 1 containment sump modifications. As a result, the licensee failed to obtain prior NRC approval for a change to the technical specifications incorporated in the license. On March 6, 2007, the licensee identified that the current refueling water storage tank minimum technical specification level was not adequate to ensure that the new containment sump would perform the required safety function. On April 20, 2007, Pacific Gas and Electric completed a 10 CFR 50.59, “Licensing Basis Impact Evaluation Screen of the Containment Sump Modification.” The licensee concluded that the modification did not involve a change to the plant technical specifications and that the required refueling water storage tank level was unaffected by the modification. On May 25, 2007, Pacific Gas and Electric placed Unit 1 into Mode 4 without an approved technical specification change.

  The inspectors concluded that the finding was more than minor because the modification required prior NRC approval. Because the issue affected the NRC’s ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions evaluated as having very low safety significance by the Significance Determination Process. The finding was determined to be of very low safety significance because the safety function was maintained since Pacific Gas and Electric had administratively maintained the refueling water storage tank at an adequate level during plant operation. On this basis, the item impacts the mitigating systems cornerstone and screens to Green, using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 evaluation, Appendix A, because (a) the finding is not a design or qualification deficiency, (b) there is no loss of safety function for the mitigating system; and, (c) there are no seismic, fire, flooding or severe weather initiating implications associated with the finding. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not appropriately prioritize and evaluate the problem of an inadequate refueling water storage tank level after the problem was entered into the corrective action program, P.1(c). This issue was entered into the licensee’s corrective action program as Action Request A07145625 (Section 4OA2.2).

REPORT DETAILS

Summary of Plant Status

At the beginning of the inspection period, Pacific Gas and Electric Company (PG&E) was operating both units at Diablo Canyon at full power. On December 3, 2007, PG&E reduced both units to 24 percent power to mitigate high storm seas. The licensee returned Unit 1 to full power on December 5 and Unit 2 to full power on December 8, 2007. PG&E operated both units at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignments (71111.04)

Partial System Walkdowns

a. Inspection Scope

The inspectors: (1) walked down portions of the below listed risk important system and reviewed plant procedures and documents to verify that critical portions of the selected system were correctly aligned; and, (2) compared deficiencies identified during the walk down to the FSAR Update and corrective action program (CAP) to ensure problems were being identified and corrected.

- October 15, 2007, Unit 1, Emergency Diesel Generator 1-1

The inspectors reviewed Procedure STP M-9A, “Diesel Generator Routine Surveillance,” and Drawings 106721, “Diesel Engine Generator” for the inspection.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Inspection

a. Inspection Scope

The inspectors walked down the 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 that they remained functional; (3) observed fire suppression systems to verify that 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 a 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 FSAR Update to determine if PG&E identified and corrected fire protection problems.

- November 19, 2007: Unit 1, Fire Area 14-E, Component cooling water heat exchanger room

- November 19, 2007: Unit 2, Fire Area 19-E, Component cooling water heat exchanger room

- November 19, 2007: Unit 1, Fire Area 7-A, Cable spreading room

- November 19, 2007: Unit 2, Fire Area 7-B, Cable spreading room

- November 23, 2007: Unit 1, Fire Area 3-Q-2, Motor driven auxiliary feedwater pump

- November 23, 2007: Unit 2, Fire Area 3-T-2, Motor driven auxiliary feedwater pump

- December 14, 2007: Unit 2, Fire Area 1-A, Containment annular area

- December 14, 2007: Unit 2, Fire Area 1-C, Containment operating deck

Documents reviewed by the inspectors included:

- Diablo Canyon Power Plant Units 1 and 2 FSAR Update, Appendix 9.5A, “Fire Hazards Analysis,” Revision 17

- Diablo Canyon Power Plant Fire Protection Pre-Plan, May 14, 2003

The inspectors completed eight samples.

b. <u>Findings</u>

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed PG&E's programs, verified performance against industry standards, and reviewed critical operating parameters and maintenance records for Component Cooling Water Heat Exchangers 1-1 and 1-2. The inspectors verified that: (1) performance tests were satisfactorily conducted for heat exchangers/heat sinks and reviewed for problems or errors; (2) PG&E utilized the periodic maintenance method outlined in EPRI NP-7552, "Heat Exchanger Performance Monitoring Guidelines;" (3) PG&E properly utilized biofouling controls; (4) PG&E's heat exchanger inspections adequately assessed the state of cleanliness of their tubes, and, (5) the heat exchangers were correctly categorized under the Maintenance Rule.

The inspectors reviewed Diablo Canyon Power Plant Component Cooling Water 1-1 and 1-2 Heat Exchanger Test, Pre-1R14, May 2007, and Procedure PEP M-234, "Component Cooling Water Heat Exchanger Performance Test," Revision 9, for the inspection.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Quarterly Inspection

a. Inspection Scope

On October 30, 2007, the inspectors observed a seismic event with anticipated transient without scram evaluation on the plant simulator. The inspectors observed the evaluation to identify any deficiencies and discrepancies in the training to assess operator performance and the evaluator's critique.

Documents reviewed by the inspectors included Lesson FRS1-A, "Seismic Event with ATWS," Revision 15.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Biennial Inspection

a. Inspection Scope

The inspectors reviewed the licensee's simulator activities using Inspection Procedure 71111.11, "Licensed Operator Requalification Program," and 10 CFR 55.46, "Simulation Facilities," as acceptance criteria. The purpose of this review was to determine if the simulator was capable of supporting initial examinations, supporting requalification training required for all licensed operators on shift, and supporting reactivity and control manipulations for initial license applications.

The inspectors reviewed the simulator annual performance test book for 2007, in which most of the annual tests were conducted between September and November 2007, using ANS/ANSI 3.5-998, "Nuclear Power Plant Simulators for Use in Operator Training and Examination," as committed to by PG&E in the Simulator Testing Procedure "Configuration Management Plan for the Operator Training Simulator," CF2.DC1, Revision 4. Because the licensee informed the inspectors that the simulator would be used for reactivity manipulation credits on the next initial examination scheduled for June 2008, several core performance test documents were reviewed in order to assess the adequacy of the simulator in supporting reactivity and control manipulations as documented on NRC Form 398, "Personal Qualification Statement." While the simulator use for reactivity and control manipulation is permitted by 10 CFR 55.46, the simulator must meet the appropriate standards of fidelity, as required by 10 CFR 55.46(c)(2). The inspectors reviewed the criteria in 10 CFR 55.46(c)(2) against the core performance test document samples and the Cycle 15 test data from the plant. The simulator was using the Cycle 15 core load for the current training cycle and no issues were found.

Three transient tests, one scenario-based test package, and a work package closeout test were run on the simulator in order to verify that the data collected from the previous tests was an accurate representation of the test data run during the testing in November 2007, and also a verification of reasonable model performance based on the current design of the plant. These tests were: (1) design basis loss of coolant accident with subsequent loss of off-site power transient test eight; (2) maximum size unisolable main steam line rupture transient test nine; (3) slow primary system depressurization to saturation condition with pressurizer relief or safety valve stuck open (inhibit activation of high pressure emergency core cooling system) transient test ten; (4) scenario-based test package for mid-loop operations; and, (5) discrepancy work closeout package for radiation monitor failures discovered during a loss of coolant accident scenario test.

As part of this review, the inspectors interviewed one instructor, one evaluator, two reactor operators, two senior reactor operators, one simulator engineer, and the simulator support supervisor. The interviews were performed to collect feedback regarding the fidelity of the simulator, the simulator discrepancy reporting system effectiveness, and training on differences between the simulator and the plant. The inspectors reviewed several program documents that describe the overall simulator program. One item specifically related to this review was how management groups, such as the simulator review board, coordinate discrepancy priorities and subsequent repair decisions. These items were reviewed in order to satisfy the requirements of 10 CFR 55.46(d) for continued assurance of simulator fidelity through problem

identification and resolution, proper reporting, root cause evaluations, and a planned schedule for implementing timely corrective actions with proper content.

Documents reviewed by the inspectors are listed in the attachment.

b. Findings

The inspectors confirmed that the licensee's simulator was adequate for reactivity manipulation credits on the next initial licensing examination provided that they continue to maintain the simulator core model on the most recent core load as the plant for which licenses are being sought and the core testing program and results are maintained for the examiners to review on the respective examination validation week in accordance with NUREG-1021, Revision 9, Supplement 1, "Operator Licensing Examination Standards for Power Reactors."

1R12 Maintenance Effectiveness (71111.12)

Routine Maintenance Effectiveness Inspection

The inspectors reviewed the listed maintenance activities 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:

- October 26, 2007: Unit 1 and Unit 2, Plant process computer failures
- December 5, 2007: Unit 2, Containment Air Particulate Monitor RM-11 failures

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two samples.

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 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 9, 2007: Unit 1, Auxiliary Saltwater Pump 1-1 maintenance
- October 17, 2007: Unit 1, Reactor coolant pump under voltage and under frequency relay calibration
- November 6, 2007: Unit 2, Emergency Diesel Generator 2-1 maintenance outage
- November 20, 2007: Unit 1, Component Cooling Water Heat Exchanger 1-2 and Auxiliary Saltwater Pump 1-2
- December 14, 2007: Unit 2, Steam generator replacement project controls and plans to minimize adverse impact on the Unit 1 and common systems

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed five 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.

- October 17, 2007; Unit 1 Emergency Diesel Generator 1-1, unplanned outage to repair exhaust manifold

Documents reviewed by the inspectors included Procedure AD7.DC6, “On-line Maintenance Risk Management,” Revision 9.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

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; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and, (6) verified that PG&E has identified and implemented appropriate corrective actions associated with degraded components.

- October 9, 2007: Unit 1, Emergency Diesel Generator 1-3 oil leak
- October 31, 2007: Unit 2, Battery charger soldering deficiencies
- November 19, 2007: Unit 2, Emergency Diesel Generator 2-1 lube oil filter leak
- December 3, 2007: Unit 2, Auxiliary building ventilation system damper maintenance
- December 4, 2007: Unit 1, Turbine Driven AFW Pump Steam Supply Valve FCV-95 incorrect valve stem lubrication
- December 31, 2007: Unit 2, Seismic degradation of control room panels

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the 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 the 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:

- October 10, 2007: Unit 1 Auxiliary Saltwater Pump 1-1 preventive maintenance
- October 15 and 16, 2007: Unit 1 Diesel Generator 1-1 preventive maintenance
- November 1, 2007: Unit 2, Safety Injection Pump Motor 2-1 preventive maintenance
- November 2, 2007: Unit 2 Safety Injection Pump 2-1 preventive maintenance
- November 11, 2007: Unit 2 Emergency Diesel Generator 2-1 preventive maintenance
- December 4, 2007: Residual Heat Removal Pump 1-1 preventive maintenance
- December 9, 2007: Unit 2, Turbine Driven Auxiliary Feedwater Pump 2-1 preventive maintenance
- December 10, 2007: Unit 1 Fuel Handling Building Ventilation System Exhaust Fan E-6 preventive maintenance

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed eight 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 below listed surveillance activities demonstrated that the SSCs tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed the 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) engineering evaluations, root causes, and bases for returning tested SSCs not meeting the test acceptance criteria were correct; (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.

- October 9, 2007: Unit 1, Check valve inspections, inservice test

- October 23, 2007: Unit 1, Containment Air Particulate Radiation Detector RM-11

- November 1, 2007: Unit 2, Emergency Diesel Generator 2-3 room carbon dioxide fire system test

- November 21, 2007: Unit 2, Emergency core cooling venting

- December 9, 2007: Unit 2, Inservice test of turbine driven Auxiliary Feedwater Steam Stop Valve FCV-95

- December 9, 2007: Unit 2, Inservice test of steam supply to turbine driven Auxiliary Feedwater Turbine FCV-37 and FCV-38

- December 9, 2007: Unit 2, Inservice test of Auxiliary FeedwaterPump Discharge Valves LCV-106, 107, 108, and 109

- December 23, 2007: Unit 1, Reactor coolant system water inventory balance

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one RCS leak detection, three routine, and four inservice test 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 TSs to ensure that listed temporary modifications were properly implemented. The inspectors: (1) verified that the modifications did not have an effect on system operability/availability; (2) verified that the installation was consistent with modification documents; (3) ensured that the 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.

- October 29, 2007: Unauthorized temporary modification installing 480 volt power in the Unit 1 component cooling water heat exchanger room

- November 15, 2007: Unit 1, 12kV Bus E potential transformer temporary connection

- December 13, 2007: Unit 2, Steam generator replacement project engineering design, modification, and analysis associated with steam generator lifting and rigging

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Emergency Preparedness Evaluation (71114.06)

a. Inspection Scope

For the listed drill contributing to Drill/Exercise Performance and Emergency Response Organization Performance Indicators, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in the emergency response organization; (2) compared the identified weaknesses and deficiencies against PG&E identified findings to determine whether PG&E is properly identifying failures; and (3) determined whether PG&E performance is in accordance with the guidance of the NEI 99-02, "Voluntary Submission of Performance Indicator Data," acceptance criteria.

- July 25, 2007, Units 1 and 2, full emergency drill

Documents reviewed by the inspectors included the Diablo Canyon Power Plant Emergency Plan, Revision 4.

The inspectors completed one sample.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors assessed the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspectors used the requirements in 10 CFR Part 20, the TSs, and the licensee's procedures required by the TSs as criteria for determining compliance. During the inspection, the inspectors interviewed the radiation protection manager, radiation protection supervisors, and radiation workers.

The inspectors performed independent radiation dose rate measurements and reviewed the following:

- Performance indicator events and associated documentation packages reported by PG&E in the occupational radiation safety cornerstone
- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Barrier integrity and performance of engineering controls in airborne radioactivity areas
- Adequacy of the licensee's internal dose assessment for any actual internal exposure greater than 50 millirem committed effective dose equivalent
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Radiation work permit briefings and worker instructions
- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas
- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed 13 samples.

b. Findings

No findings of significance were identified.

2OS2 As Low As is Reasonably Achievable (ALARA) Planning and Controls (71121.02)

a. Inspection Scope

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

- Current 3-year rolling average collective exposure
- Site-specific trends in collective exposures, plant historical data, and source-term measurements
- Site-specific ALARA procedures

- Five work activities of highest exposure significance completed during the last outage

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

- Intended versus actual work activity doses and the reasons for any inconsistencies

- Person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements

- Post-job (work activity) reviews

- Method for adjusting exposure estimates, or replanning work, when unexpected changes in scope or emergent work were encountered
- Exposures of individuals from selected work groups

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

- Self-assessments, audits, and special reports related to the ALARA program since the last inspection

- Resolution through the corrective action process of problems identified through post-job reviews and post-outage ALARA report critiques

- Corrective action documents related to the ALARA program and followup activities, such as initial problem identification, characterization, and tracking

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed 14 samples.

b. <u>Findings</u>

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 <u>Performance Indicator Verification (71151)</u>

.1 <u>Cornerstone: Barrier Integrity</u>

a. <u>Inspection Scope</u>

The inspectors sampled PG&E submittals for the two performance indicators listed below for the period from September 2006 to September 2007, for Units 1 and 2. The definitions and guidance of NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 4, were used to verify PG&E's basis for reporting each data element in order to verify the accuracy of PI data reported during the assessment period.

- RCS Specific Activity
- RCS Leakage

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

.2 Cornerstone: Occupational Radiation Safety

a. Inspection Scope

The inspectors reviewed licensee documents from April 1 through September 30, 2007, in regards to occupational exposure control effectiveness. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 5). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspectors interviewed PG&E personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspectors toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

Document reviewed by the inspectors included the Action Request A0696907.

The inspectors completed one inspection sample.

b. Findings

No findings of significance were identified.

.3 Cornerstone: Public Radiation Safety

The inspectors reviewed licensee documents from April 1 through September 30, 2007, in regards to radiological effluent technical specification/offsite dose calculation manual radiological effluent occurrences. The review included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspectors interviewed PG&E personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in NEI 99-02, Revision 5, were used to verify the basis in reporting for each data element.

The inspectors completed one sample in this cornerstone.

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 the corrective action program. This assessment was accomplished by reviewing action requests and event trend reports, and attending daily operational meetings. On November 5, 2007, the inspectors attended the condition adverse to quality screening meeting. 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 corrective action program; (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

Introduction. The inspectors identified a Green noncited violation of 10 CFR 50, Appendix B, "Corrective Action," after PG&E failed to identify a degraded emergency diesel generator.

Description. On October 15, 2007, the inspectors identified black soot on the Emergency Diesel Generator 1-1 exhaust manifold. Licensee personnel subsequently identified that one of four fasteners connecting the exhaust manifold to the turbo charger was missing. The licensee declared the diesel generator inoperable based on the potential reduction of electrical power output due to exhaust gas bypassing the turbo charger and the adverse affect of the missing fastener on seismic qualification. Plant operators determined that overall plant risk elevated to significantly degraded (Orange) due the combination of the unavailable diesel generator and other plant equipment removed from service at the time. The licensee repaired and returned the diesel generator to service on October 19, 2007. Plant engineering personnel subsequently concluded that the loss of the fastener would not have prevented the diesel generator from performing the required safety function. The soot buildup was present since the last previous operation of the diesel generator on September 23, 2007. The licensee had prior opportunity to identify the degraded diesel generator. Plant operators performed at least one inspection of the diesel generator each shift in accordance with Procedure OP1.DC3, "Operator Routine Plant Equipment Inspections," Revision 8. Operations Policy A-22, "Expectations for Nuclear Operator Watchstanders," November 16, 2004, required operators to maintain an awareness of equipment condition and to report problems in a timely manner.

Analysis. Failure of PG&E operations personnel to identify the degraded diesel generator during operator rounds was a performance deficiency. This finding is greater than minor because, if left uncorrected, continued failure to perform adequate operator rounds would become a more significant safety concern. This finding involved an emergency diesel generator and affected the mitigating systems cornerstone. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, this finding was determined to have very low safety significance because it did not result in a loss of operability of a single train, for greater than Technical Specification allowed outage time, did not result in the loss of safety function, and was not potentially risk significant from a seismic, flooding or severe weather perspective. This finding has a

crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because plant operators did not maintain a low threshold for identifying issues, P.1(a).

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI, "Corrective Action," requires that measures be taken to assure that conditions adverse to quality are promptly identified and corrected. Contrary to the above, the licensee failed to identify a condition adverse to quality. Specifically, between September 23 and October 15, 2007, plant operators did not identify the degraded Emergency Diesel Generator 1-1 exhaust. Because this finding is of very low safety significance and was entered into the corrective action program as Action Request A0710082, this violation is being treated as a noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000275/2007005-01, "Plant Operators Failed to Identify a Degraded Emergency Diesel Generator."

.2 Selected Issue Follow-Up Inspection

a. Inspection Scope

In addition to the daily screening, the inspectors conducted an in-depth review of the listed issues. 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.

- A0710652, November 21, 2007, Discrepancies associated with ultrasonic flow monitor nozzle fouling factor input into secondary system heat balance calculations

- A0712404, November 20, 2007, Evaluate if licensee bases impact evaluation screen for less than adequate reactor water storage tank level required a license amendment

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed two in-depth review samples.

b. Findings

Introduction. The inspectors identified a noncited Severity Level IV violation of 10 CFR 50.59 after PG&E failed to perform an adequate safety evaluation of Unit 1 containment sump modifications. As a result, the licensee failed to obtained prior NRC approval for a change to the technical specifications incorporated in the license.

Description. PG&E modified the Unit 1 containment sump during the Spring 2007 refueling outage. Technical Specification 3.5.4, "Refueling Water Storage Tank," required 81.5 percent indicated level (400,000 gallons). This technical specification ensured enough water was provided during accident mitigation to ensure the residual heat removal pump net positive suction head during the transition from cold leg injection

mode to cold leg recirculation mode emergency core cooling. After the containment sump modification, 93.6 percent level in the refueling water storage tank (RWST) was required to ensure residual heat removal pump net positive suction head. On March 6, 2007, the licensee identified that the current RWST minimum technical specification level was not adequate to ensure the new containment sump would perform the required safety function. This was entered into the corrective action program as Action Request A0690337. On April 20, 2007, PG&E completed a 10 CFR 50.59 Licensing Basis Impact Evaluation Screen of the containment sump modification. The licensee concluded that the modification did not involve a change to the technical specifications and that the RWST Technical Specification was unaffected by the modification. On May 25, 2007, PG&E placed Unit 1 into Mode 4 without an approved technical specification change. On October 2, 2007, the licensee submitted a License Amendment Request to raise the RWST technical specification minimum level to meet the new sump design requirements. Plant operators had administratively maintained the Unit 1 RWST at the higher level since entry into Mode 4.

Analysis. The failure of PG&E to perform an adequate safety evaluation of the containment sump modification, to identify that prior NRC approval was required, is a performance deficiency. The inspectors concluded that the finding was more than minor because the modification required NRC prior review and approval. Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions evaluated as having very low safety significance by the SDP. The finding was determined to be of very low safety significance because the safety function was maintained since PG&E had administratively maintained the RWST at least 93.6 percent indicated level during plant operation. On this basis, the item impacts the Mitigating System Cornerstone and screens to GREEN using IMC 0609, Phase 1 SDP Evaluation, Appendix A, because (a) the finding is not a design or qualification deficiency, (b) there is no loss of safety function for a mitigating system and, (c) there are no seismic, fire, flooding or severe weather initiating implications associated with the finding. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not appropriately prioritize and evaluate the problem of an inadequate refueling water storage tank level after the problem was entered into the corrective action program, P.1(c).

Enforcement. Title 10 of the Code of Federal Regulations, Part 50.59(c)(1) requires, in part, that a licensee may make changes in the facility as described in the final safety analysis report without obtaining a license amendment only if a change to the technical specifications incorporated in the license is not required. Title 10 CFR 50.36 requires, in part, that a technical specification limiting condition for operation of a nuclear reactor must be established for a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Contrary to the above, on May 25, 2007, PG&E made changes to the facility as described in the final safety analysis report without obtaining a license amendment when a change to the technical specifications incorporated in the license was required. Specifically, PG&E changed the containment sump design such that additional minimum Technical Specification level was required to ensure the design basis accident primary success path. Because this finding is of very low safety significance and was entered into the corrective action program as Action Request A0715625, this violation is being treated as a noncited violation in accordance

with Section VI.A.1 of the Enforcement Policy: NCV 05000275/2007005-02, “Inadequate 50.59 Evaluation for Unit 1 Containment Sump Modification.”

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors completed a semiannual trend review of repetitive or closely related issues that were documented in action requests, maintenance rule reports, system health reports, problem lists, and performance indicators to identify trends that might indicate the existence of more safety significant issues. The inspectors' review consisted of the six-month period from July to December 2007. When warranted, some of the samples expanded beyond those dates to fully assess the issue. Corrective actions associated with a sample of the issues identified in PG&E's trend report were reviewed for adequacy.

b. Findings

Continued Adverse Trend in Plant Equipment Material Condition

The inspectors concluded that the adverse trend in plant material condition, discussed in Diablo Canyon Power Plant Integrated Inspection Report 05000275/2007003 and 05000323/2007003, continued through the inspection period. Current examples of poor material condition identified by the inspectors included:

- A0710082, Diesel Generator 1-1 Exhaust Leak
- A0710921, Control room air conditioner condenser missing nine fasteners
- A0706792, Control Room Air Conditioner CR-38 corrosion
- A0710807, Component Cooling Water Heat Exchanger Saltwater Inlet Valve FCV-603 corrosion
- A0710817, Oil seeping from Component Cooling Water Valve 1-21
- A0719815 and A0710809, Component cooling water corroded conduits
- A0710801, Corrosion on Component Cooling Water Valve FCV-602
- A0710987, Component Cooling Water gasket for ASW to S/Gs

PG&E capture this adverse trend in the corrective action program as Action Request A0711113.

NRC Identified Adverse Trend in Managing Maintenance Risk

The inspectors identified that the maintenance risk management thresholds established by the licensee were considerably below thresholds established in the industry guidance provided in NUMARC 93-01, “Nuclear Energy Institute, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Revision 3. Because of low threshold, PG&E typically entered elevated “Yellow Risk” (Integral Core Damage

Probability $1x10^{-6}$ to $1x10^{-5}$) an average of 12 times per week. The licensee also entered elevated “Orange Risk” (Integral Core Damage Probability greater than $1x10^{-5}$) several times during the inspection. The inspectors concluded that routine and repetitive declaration of elevated plant maintenance risk resulted in plant of personal desensitized to the subsequent risk management actions. Examples included:

- A0711074, Failure to control access to redundant equipment following removal of a Unit 2 component cooling water pump during declared “Yellow Risk.”
- A0711107, October 31, 2007, Removal of protective area around Unit 2 main feed water pumps while Diesel Generator 2-3 was planned to be removed from service for fire protection testing, during declared “Yellow Risk.”
- A07164454, November 6, 2007, Unit 1 Shift Foreman was unaware of risk management actions following removal of a control room ventilation system during declared “Yellow Risk.”
- A0711876, Unit 2 did not document entry into “Orange risk.”

The inspectors concluded that each example was minor because actual industry risk threshold values for the required risk management actions were not exceeded. The licensee entered this adverse trend into the corrective action program as Action Request A0711061.

Documents reviewed by the inspectors are listed in the attachment.

The inspectors completed one sample in this inspection.

.4 Operator Workaround Review

The inspectors conducted a review to verify that the licensee is identifying operator workaround problems at an appropriate threshold, entered these issues into the corrective action program and that the licensee has proposed or implemented appropriate corrective actions. The inspectors reviewed the November 9, 2007, Operator Workaround and Control Room Deficiency Report.

The inspectors completed one workaround sample.

.5 Occupational Radiation Safety

a. Inspection Scope

The inspectors evaluated the effectiveness of PG&E’s problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report 05000275/2007001-00, Emergency Diesel Generator Auto-start on Loss of Offsite 230kV Startup Power

On May 12, 2007, offsite startup power was lost to both units at Diablo Canyon after a Morro Bay-Diablo Canyon 230 kV line transmission failed. The Unit 1 reactor was defueled at the time and powered by the startup bus. Diesel Generators 1-1 and 1-2 automatically started and powered vital buses per plant design. This event was previously discussed in Diablo Canyon Power Plant Integrated Inspection Report 323/2007003. No violation of NRC requirements was identified in this LER. This LER is closed.

.2 (Closed) Licensee Event Report 05000275/2007002-00, Manual Reactor Trips During Mode 3 Rod Testing Due to Crud Related Rod

On May 27, 2007, Control Rod N-13 slipped from 42 steps to 24 steps while operators were performing Surveillance Test Procedure STP R-1C, "Digital Rod Position Indicator Functional Test," Revision 16, while Unit 1 was in Mode 3. Plant operators manually tripped the reactor. The licensee concluded that the rod slippage was due to the crud buildup on the control rod drive shaft. The vendor recommended that operators exercise Control Bank 'C' out and back in five times in order to remove the crud from the drive shaft. During the sequence of five rod exercises, Control Rod N-13 slipped three more times, with each sequential slip occurring at higher steps out of the core. Operators exercised Control Bank 'C' five more times without any additional control rod slippage. PG&E staff subsequently concluded that the crud on the Control Rod N-13 drive shaft had been removed to the reactor coolant system. This event was previously discussed in Diablo Canyon Power Plant Integrated Inspection Report 05000323/2007003. No violation of NRC requirements was identified in this LER. This LER is closed.

.3 (Closed) Licensee Event Report 05000275/2007003-00, Emergency Diesel Generator Actuation Due to a Transient Undervoltage Condition

On May 28, 2007, an unplanned automatic start of Diesel Generator 1-2 occurred due to degraded bus voltage after Circulating Water Pump 1-2 was started. Plant operators had transferred all plant electrical loads from the auxiliary power supply to startup power in preparation of paralleling the main generator to the electric grid. Once all plant electrical loads were connected to the startup power, operators started Circulating Water Pump 1-2. Due to the large in-rush current required to start the pump, voltage degraded on all plant electrical buses, including the vital 4 kV buses. The voltage on the vital 4 kV buses degraded to the point that the second-level undervoltage relays actuated and began the time delay sequence. The voltage on Vital 4 kV Buses F and H recovered prior to the end of the time sequence. However, the voltage on Vital Bus G did not sufficiently recover resulting in a trip of the startup power supply breaker and an automatic start of the diesel generator. Plant engineering personnel determined that the large circulating water pump motor in-rush current along with low startup power supply margin resulted in the degraded voltage on the vital buses. The inspectors reviewed the event sequence, equipment performance, operator actions, and plant electrical design as they relate to this event. No violation of NRC requirements was identified in the LER. This LER is closed.

4OA5 Other

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

a. Inspection Scope

The inspectors witnessed the final portion of the vertical cask transporter (VCT) functional testing performed at Diablo Canyon and completed reviews of documentation from tests that had been conducted at the fabricator facility. The VCT is classified as Important to Safety (ITS) and used to transport the loaded multi-purpose canister (MPC) inside the HI-TRAC transfer cask from the fuel building to the ISFSI. The VCT is seismically qualified and has redundant drop protection features in accordance with the applicable guidelines of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.” The VCT will also serve as the cask transfer facility (CTF) to lower the MPC into the HI-STORM concrete overpack.

The NRC performed an inspection of the VCT fabrication at Lift Systems, Inc., located at Moline, Illinois, in January 22-26, 2007 (ML070400122). The factory acceptance testing called for a 100 percent functional test, 125 percent static load test, 150 percent MPC downloader test, 125 percent test of the seismic restraint lugs and an inclined functional test. Overall, the fabrication activities were found to be in compliance with 10 CFR Part 72 regulations and the NRC approved Holtec QA program. The MPC downloader test and nondestructive examinations of the critical welds were performed after the NRC inspection. The inclined functional test was scheduled to be performed after the VCT was delivered to Diablo Canyon.

Holtec provided a copy of the completed factory acceptance testing Holtec Project Procedure (HPP) 1073-6, Revision 2, which documented the successful 150 percent load test of the MPC downloader. The MPC downloader is used to transfer the loaded MPC from the HI-TRAC transfer cask into the HI-STORM overpack. The MPC downloader test documentation stated that a test load of 67.5 tons (+10 tons/-0 tons) had been used to perform the load test and the test load had been held for at least 10 minutes. A list of the VCT critical welds which were examined by nondestructive examination following the load tests was provided by Holtec. The welds were classified as critical per Section 3.4 of Holtec Standard Procedure HSP 187, “Interface Procedure for Manufacturing of ITS B Transporters at Lift Systems,” Revision 3, as a weld on an essential component whose failure would directly lead to an uncontrolled lowering of the lifted load or failure of other critical design function. The classification was stated to conform to the evaluation criteria contained in NUREG-0612, Section 5.1, and the definition of a critical load per ANSI N14.6, Section 3.4. The inspectors reviewed the visual and magnetic particle test reports for selected welds that were classified as critical. No discrepancies were found during the review. The VCT cask restraint system design and fabrication were documented to meet the requirements of the ASME NF Code. NUREG-0612 does not address horizontal loads such as those associated with the cask restraint system. The spent fuel storage and transportation staff determined that meeting the design and fabrication requirements of the ASME NF Code would be sufficient to ensure that the cask restraint system operated safely.

The VCT was shipped to Diablo Canyon to perform the inclined functional test. The inclined functional test utilized the HI-TRAC transfer cask with an MPC that included sufficient weight (82,500 pounds) to simulate a fully loaded MPC. The inclined functional

test consisted of transporting the HI-TRAC and MPC from the entrance of the radiological controlled area (RCA) down the 8.5 percent grade and up the nominal 6 percent grade to the CTF. The distance of the transport path was approximately 1.2 miles in length.

The transport path or roadway from the fuel building to the ISFSI was designed and constructed to meet AASHTO H-20 loadings. The licensee had recently discovered that the loadings imposed by the VCT tracks exceeded the H-20 design loadings. Minor modification package AT-MM A0710693 was originated to evaluate the impact upon embedded structures located beneath the roadway from the actual loads that were exerted by the VCT along the route affected by the inclined function test. The minor modification package verified that there were no safety related components located underneath the portion of the roadway used for the inclined functional test. However, several portions of the roadway were required to be reinforced and minimum stand-off distances were specified from the VCT to several of the boxes/vaults located along the transport route. The licensee noted that the use of the roadway for movement of spent fuel (including the area inside the RCA) will be addressed by a separate design change package.

The inclined functional load test was conducted after 1800 on Monday, December 10, 2007. The HI-TRAC with the simulated weight MPC was lifted by the VCT and secured by the cask restraint system. The VCT was taken inside the protected area of the plant up to the RCA fence. The VCT then traversed down the 8.5 percent grade and across the relatively flat area of the plant site. Prior to beginning the trip up the nominal 6 percent grade, the 50 gallon fuel tank was refilled and the VCT then negotiated the nominal 6 percent incline to the CTF. A minor delay was experienced when hydraulic fluid was discovered leaking from an “O” ring fitting. The VCT inclined functional load test was completed satisfactorily without any safety significant findings.

Diablo Canyon Technical Specification 4.3.1.c required that the cask transporter be designed, fabricated, inspected, maintained, operated, and tested in accordance with the applicable guidelines of NUREG 0612. The staff reviewed the cask transporter documentation to determine if the cask transporter met the applicable guidelines of NUREG 0612. During this review, two questions were raised by the staff that required additional clarification from the licensee.

The first question was associated with how the licensee had determined the testing requirements for the transporter load/lift links, which were part of the VCT lifting mechanism. The load/lift links had been designed to meet the increased design stress factors of 6 for yield and 10 for ultimate strength as required by ANSI N14.6. The load testing of the load/lift links had been performed at 125 percent of the VCT rated load as required by ASME B30.2, “Overhead and Gantry Cranes.” The licensee stated that this methodology was based on the previously approved use of similar lifting devices that were required to meet multiple criteria of NUREG 0612. The staff determined that the licensee testing of the VCT load/lift links met the applicable portions of NUREG 0612.

The second question raised by the staff was how the requirements of NUREG 0612, Section 5.1.1(6) which specified that the crane (transporter) be inspected, tested, and maintained in accordance with Chapter 2-2 of ASME B30.2 was to be achieved. The vendor had provided a maintenance manual for the transporter. However, the maintenance manual did not specify any frequent or periodic inspections for the VCT which would parallel the inspection requirements that were contained in Chapter 2-2 of

ASME B30.2. The licensee committed to providing additional instructions in the maintenance manual that would meet the inspection and maintenance recommendations contained in NUREG 0612, Section 5.1.1(6) and instructions for meeting the specifications contained in ANSI N14.6 for the MPC downloader. The revised VCT maintenance manual will be reviewed during a future inspection.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On November 8, 2007, the inspectors presented the occupational radiation safety inspection results to Mr. John Conway, Site Vice President, and other members of his staff who acknowledged the findings.

On December 11, 2007, the inspectors presented the independent spent fuel storage installation inspection results to Mr. John Conway, Site Vice President, and other members of his staff.

On December 12, 2007, the inspectors discussed the inspection results of the simulator fidelity portion of the licensed operator biennial requalification inspection with Mr. Jim Welsch, Operations Manager, and other members of his staff.

On January 9, 2008, the resident inspection results were presented to Mr. John Conway, Site Vice President and other members of PG&E management. PG&E acknowledged the findings presented.

In each case, 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
S. Hamilton, Supervisor, Regulatory Services
R. Hite, Manager, Radiation Protection
D. Jacobs, Vice President - Nuclear Services
S. Ketelsen, Manager, Regulatory Services
K. Langdon, Director, Operations Services
R. Lovell, Senior Nuclear Engineer
M. Meko, Director, Site Services
K. Peters, Director, Engineering Services
J. Purkis, Director, Maintenance Services
P. Roller, Director, Performance Improvement
M. Somerville, Manager, Radiation Protection
D. Taggart, Manager, Quality Verification
R. Waltos, Manager, Emergency Preparedness

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000275/2007005-01	NCV	Plant Operators Failed to Identify a Degraded Emergency Diesel Generator (Section 4OA2.1)
05000275/2007005-02	NCV	Inadequate 50.59 Evaluation for Unit 1 Containment Sump Modification (Section 4OA2.2)

Closed

05000275/2007001-00	LER	Emergency Diesel Generator Auto-start on Loss of Offsite 230kV Startup Power (Section 4OA3.1)
05000275/2007002-00	LER	Manual Reactor Trips During Mode 3 Rod Testing Due to Crud Related Rod Slippage (Section 4OA3.2)
05000275/2007003-00	LER	Emergency Diesel Generator Actuation Due To A Transient Undervoltage Condition (Section 4OA3.3)

LIST OF DOCUMENTS REVIEWED

Section 1R11: Licensed Operator Requalification (71111.11)

Biennial Inspection

Procedures

Number	Title	Revision
CF2.DC1	Configuration Management Plan for the Operator Training Simulator	4
SQA 99-2	Operator Training Simulator Software Quality Assurance Plan	0
STP R7-A	Determination of Moderator Temperature Coefficient at HZP, BOL	15
STP R31	Rod Worth Measurements Using Rod Swap Method	12

Other Documents

Simulator Scenario for Mid-Loop Operations, R075S1, Rev. 0

Open Simulator discrepancy report (all)

Closed Simulator discrepancy report from November 2005 thru November, 2007

Annual Operability Test packages

- Steady state power test (83% and 25%)
- Transients Reviewed (All 11)
- Core test package for cycle 15

Simulator versus Plant differences list

Work package closeout and post-test for simulator SCR # 04-062, where high radiation alarms during a Loss of Coolant Accident scenario failed to work

Section 1R12: Maintenance Effectiveness (71111.12)

Action Requests

A0595654	A0684348	A0692558	A0702988	A0671226	A0697363
A0709074	A0709405	A0712454	A0712518		

Procedures

Number	Title	Revision
STP I-100A	Containment Air Particulate/Gas Radiation Monitor RM-11/RM-12 Functional Test	14

Other Documents

Title	Date/Revision
System Report Details, System 43A “Plant Process Computer”	7/1 - 9/30/2007
System Report Details, System 39A/B “Rad. Monitoring System”	10/1 - 12/31/2007

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

Action Requests

A0711265 A0711289 A0711270

Procedures

Number	Title	Revision
AD7.DC6	On-Line Maintenance Risk Management	10
OP J-6B:IX	Diesel Generator Extended On-Line Maintenance	1
AD8.ID1	Outage Planning and Management	9
TP 22.0	Risk Management Task Plan	0
OP AP-9	Loss of Instrument Air	23
OP AP SD-4	Loss of Component Cooling Water	16

Section 1R15: Operability Evaluations (71111.15)

Action Requests

A0691325	A0697545	A0709245	A0710347	A0710345	A0710346
A0690663	A0710353	A0712157	A0454787	A0711781	A0664398
A0712272	A0712386	A0715324			

Procedures

Number	Title	Revision
MP I-2.17-1	Guidelines for Printed Circuit Board Repair and Inspection	3
OM4.ID3	Assessment of Industry Operating Experience	12
STP M-21-ENG.1	Diesel Engine Generator Procedure (Every Refueling Outage)	8

Other Documents

Title	Date/Revision
Work Order, C0216111 "DEG 2-1 Repair (Lube Oil Filter Housing Leak)"	11/17/2007

Section 1R19: Postmaintenance Testing (71111.19)

Action Requests

A0712336	A0712337	A0701654	A0711712

Procedures

Number	Title	Revision
STP M-9A	Diesel Engine Generator Routine Surveillance Test	74
STP M-9X	Diesel generator Operability Verification	20
STP M-9D1	Diesel Generator Full Load Rejection Test	12
MP E-57.10C	Generic 4kV Motor Preventive Maintenance	0
STP P-RHR-11	Routine Surveillance Test of RHR Pump 1-1	20
STP P-SIP-21	Routine Surveillance test of Safety Injection Pump 2-1	20
STP P-AFW-21	Routine Surveillance Test of Turbine Driven AFW Pump	20
MP M-17.9	Auxiliary Salt Water Pump Maintenance	20
STP M-5	Surveillance Test of the Fuel Handling Building Ventilation System	26

Other Documents

Title	Date/Revision
Work Order, R0285110, “RHR PP 1-1 Discharge Miniflow Control”	11/30/2007
Work Order, R0285591, “4KV RHR PP 1-1"	11/29/2007
Work Order, R0299925, “RHR Pump 1"	11/21/2007
Work Order, R0306079, “Residual Heat Removal Pump 1-1"	11/21/2007
Design Criteria Memorandum S-10, “Residual Heat Removal System”	15
Work Order, R0295991, “E-6 Motor: Clean, Inspect, and Test”	12/6/2007

Section 1R22: Surveillance Testing (71111.22)

Action Requests

A0710155 A0711107 A0691531

Procedures

Number	Title	Revision
STP I-100A	Containment Air Particulate/Gas Radiation Monitor RM-11/RM-12 Functional Test	14
STP M-39A3	Routine Surveillance Test of Diesel Generator 1-3 (2-3) Room Carbon Dioxide Fire System Operation	12
STP M-89	Unit 2, ECCS System Venting	37
STP R-10C	Reactor Coolant System Water Inventory Balance	34
STP V-3P5	Exercising Valves LCV-106, 107, 108, and 109 Auxiliary Feedwater pump Discharge	20
STP V-18M	Check Valve Inspections - High Maintenance Valves	10
STP-V-3R5	Exercising Steam Supply to AFW Pump Turbine Stop Valve	19
STP-V-3R6	Exercising Steam Supply to AFW Pump Turbine Isolation Valves	10

Drawings

Number	Title	Revision
663227-55	Air Flow Diagram - RE 11/RE 12	4
102931-11	Instrument Schematic - Radiation Instruments	57

Section 1R23: Temporary Plant Modifications (71111.23)

Action Requests

AR0711967

Procedures

Number	Title	Revision
CF4.ID7	Temporalty Modifications	19
TP TD-0703	12kV Bus "E" Potential Transformer Repair at Power	0
TS3.ID2	Licensing Basis Impact Evaluation	22

Drawings

Number	Title	Revision
437612-1	Unit 1 Electrical Schematic Diagram Bus Potential & Synchronizing 12kV System	23

Other Documents

Title	Date/Revision
Design Change Package C-050743, "SGRP - Rigging and Handling Inside Containment"	0
Design Change Package C-050745, "SGRP - Rigging and Handling Outside Containment"	0
Calculation # 52.15.141, "SGRP Load Path Evaluation - Impact on Elev. 140' Auxiliary Building Area GE and K Response Spectra due to Movement of New/Old Steam Generators (Enova calc. 0104-031-C01)	2A
Calculation # 52.15.142, "SGRP Load Path Evaluation - Seismic Stability Analysis of the Steam Generator/Transporter at Elev. 140' Auxiliary Building Area GE and K (Enova calc. 0104-032-C01)	1A

Calculation # 52.15.143, “SGRP Load Path Evaluation - Evaluation of Floor at Elev. 140' Auxiliary Building Area GE and K due to Movement of New/Old Steam Generators (Enova calc. 0104-032-C03)	2
R0309046 02 Diesel generator 2-3 installation of recorder to monitor DEG start, December 21, 2007	

Section 2OS1: Access Controls to Radiologically Significant Areas (71121.01)

Action Requests

A0695011	A0695199	A0695353	A0695683	A0695740	A0695961
A0696887	A0696907	A0697100	A0697727	A0698424	A0700022
A0700672	A0703336	A0703338	A0703339		

Audits and Self-Assessments

Diablo Canyon Power Plant Quality Performance Assessment Report, 1st Period 2007
Diablo Canyon Power Plant Quality Performance Assessment Report, 2nd Period 2007
2007 Radiation Protection Program Audit dated August 9, 2007

Procedures

Number	Title	Revision
RP1	Radiation Protection	4A
RCP D-240	Radiological Posting	17
RCP D-370	Evaluation of Internal Deposition of Radioactive Material	7
RCP D-500	Routine and Job Coverage Surveys	23

Section 2OS2: ALARA Planning and Controls (71121.02)

Action Requests

A0695351	A0695591	A0695595	A0695730	A0695936	A0695974
A0696277	A0696286	A0696352	A0696741	A0697472	A0697722
A0698373	A0698378	A0698387	A0698396	A0698548	A0698582
A0699099	A0701401	A0703330	A0703352		

Radiation Work Permits

SWP 1002	1R14 Scaffolding in Containment
SWP 1033	1R14 RHR Recirc & Containment Sump
SWP 1044	1R14 Eddy Current Inspection & Tube Work

SWP 1074 1R14 Fire Stops and Insulation Replacement
SWP 1077 1R14 Insulation Double Banding

Procedures

Number	Title	Revision
RP1 D-200	Writing Radiation Work Permits	37
RCP D-205	Performing ALARA Reviews	15
RCP D-211	Control of Work in Radiologically Significant Areas	3
RCP D-220	Control of Access to High, Locked High, and Very High Radiation Areas	32
RCP D-222	Radiation Protection Lock and Key Control	5

Section 4OA2: Identification and Resolution of Problems (71152)

Action Requests

A0695011	A0695199	A0695353	A0695683	A0695740	A0695961
A0487341	A0487674	A0616447	A0619113	A0638538	A0650060
A0711967	A0777733	A0710082	A0677367	A0675465	A0710868
A0710817	A0715625	A0657428	A0686900	A0706229	A0708847
A0710652					

Audits and Self-Assessments

Diablo Canyon Power Plant Quality Performance Assessment Report, 1st Period 2007
Diablo Canyon Power Plant Quality Performance Assessment Report, 2nd Period 2007
2007 Radiation Protection Program Audit dated August 9, 2007

Procedures

Number	Title	Revision
TS3.ID2	Licensing Basis Impact Evaluations	8
PEP M-98A	Comparison of Final Feedwater Flow Nozzles to “AMAG” Ultrasonic Flow Instrumentation	17

Miscellaneous

Title	Date/Revision
CENPD-397-P, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology"	01-P
A-DP1-PS-0001, "Feedwater Flow Measurement Using the Crossflow Ultrasonic Flowmeter at PG&E Diablo Canyon Unit 1"	000

Semiannual Trend Review

Assessment Reports

Diablo Canyon Power Plant, Quality Performance Assessment Report, 4th Period 2006
Diablo Canyon Power Plant, Quality Performance Assessment Report, 1st Period 2007
Diablo Canyon Power Plant, Quality Performance Assessment Report, 2nd Period 2007
Diablo Canyon Power Plant, Plant Performance Improvement Report, September 2007
Diablo Canyon Power Plant, Plant Performance Improvement Report, August 2007
Quality Verification Department Bi-Weekly Observation Report, December 27, 2007

Miscellaneous

PCPP Observation Card (ID# 21788) Operation Decision Meeting following failed reactor coolant pump breaker undervoltage relay, November 2, 2007

Section 4OA5: Other

Miscellaneous

Holtec Project Procedure HPP-1073-6, "VCT Factory Test Procedure," Revision 2
Holtec Project Procedure HPP-1073-6, "VCT Factory Test Procedure," Revision 3
Everett Shipyard SS MPC Dummy Training Weight Equipment No. 505 Certified Weight
Holtec Standard Procedure HSP-187, "Interface Procedure For Manufacturing of ITS B Transporters at Lift Systems," Revision 3
Diablo Canyon Transporter Manual, Revision 0

LIST OF ACRONYMS

ADAMS	agency document and management system
AFW	auxiliary feedwater
ALARA	as low as is reasonably achievable
AR	action request
AASHTO	American Association of State Highway and Transportation Officials
CFR	Code of Federal Regulations
DEG	diesel engine generator
FSAR	Final Safety Analysis Report
HPP	Holtec project procedure
HSP	Holtec standard procedure
ISFSI	independent spent fuel storage installation
ITS	Important to Safety
LER	licensee event report
MPC	multi-purpose canister
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records System
PDT	Pacific daylight time
PG&E	Pacific Gas and Electric Company
RWST	refueling water storage tank
SSC	structure, system and component
SLURS	second-level undervoltage relays
TS	Technical Specifications
VCT	vertical cask transporter