UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

August 1, 2002

Gregory M. Rueger, Senior Vice
President, Generation and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Power Plant
P.O. Box 3
Avila Beach, California 93424

SUBJECT: DIABLO CANYON INSPECTION REPORT 50-275/02-03; 50-323/02-03

Dear Mr. Rueger:

On July 6, 2002, the NRC completed an inspection at your Diablo Canyon Nuclear Power Plant, Units 1 and 2, facility. The enclosed integrated report documents the inspection findings that were discussed on May 10, 17, and July 9, 2002, with Mr. David H. Oatley and members of your staff as discussed in Section 40A6.

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.

Based on the results of this inspection, violations of NRC requirements were identified. Because these violations were determined to be of very low safety significance and have been entered into your corrective action program, the NRC is treating these issues as noncited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny the noncited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region IV; 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.

The NRC has increased security requirements at Diablo Canyon in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

Pacific Gas and Electric Company operated under voluntary bankruptcy proceedings during this inspection period. The NRC has monitored plant operations, maintenance, and planning to better understand the impact of the financial situation and how it relates to your responsibility to safely operate the Diablo Canyon reactors. NRC inspections, to date, have confirmed that you are operating these reactors safely and that public health and safety is assured.

In accordance with 10 CFR 2.790 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/

William B. Jones, Chief
Project Branch E
Division of Reactor Projects

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

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NRC Inspection Report
50-275/02-03; 50-323/02-03

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Dockets:	50-275 50-323
Licenses:	DPR-80 DPR-82
Report:	50-275/02-03 50-323/02-03
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Nuclear Power Plant, Unit 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	April 7, 2002, through July 6, 2002
Inspectors:	D. L. Proulx, Senior Resident Inspector T. W. Jackson, Resident Inspector G. A. Pick, Senior Project Engineer M. P. Shannon, Senior Health Physics Specialist C. A. Clark, Reactor Inspector G. M. Vasquez, Reactor Inspector G. F. Suber, Engineering Intern
Approved By:	W. B. Jones, Chief, Project Branch E Division of Reactor Projects
ATTACHMENT:	Supplemental Information

SUMMARY OF FINDINGS

IR 05000-275-02-03, IR 05000-323-02-03, Pacific Gas and Electric. Co., Diablo Canyon Nuclear Power Plant, Units 1 and 2, 04/07/02 to 07/06/02. Postmaintenance Test, Access to Rad Sig Areas.

This report covers a 13-week routine resident, radiation protection, and inservice inspection. The inspection identified three Green noncited violations. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using IMC 0609, "Significance Determination Process." Findings for which the Significance Determination Process does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the failure to perform an adequate post-maintenance test on Auxiliary Saltwater Pump 1-2 prior to placing the pump in service. The licensee installed new packing on Auxiliary Saltwater Pump 1-2 as part of the pump replacement that occurred between May 9 -16, 2002. The licensee performed a post-maintenance test on Auxiliary Saltwater Pump 1-2 on May 17 and documented there was adequate packing leak-off flow. Then on May 30 operators started Auxiliary Saltwater Pump 1-2 but identified no leak-off flow. The post maintenance test was not adequate to identify that the packing had been improperly installed and that the packing had shifted and swelled following the May 17 pump run. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This item was placed in the corrective action system as Action Request A0560036.

  This violation was more than minor because if the same condition, under similar circumstances, were present for a longer period of time, the finding would be of greater safety significance. An NRC senior reactor analyst performed a significance determination process Phase 3 safety assessment. The senior reactor analyst reviewed the licensee's risk assessment, and the safety significance insights obtained from the NRC's Standardized Plant Analysis Risk (SPAR) model for Diablo Canyon Units 1 and 2 (Revision 3i) as well as NRC Manual Chapter 0609, Significance Determination Process, Appendices A and G, Significance Determination of Reactor Inspection Findings for At-Power Situations and Shutdown Safety SDP [significance determination process] for those plant conditions utilizing residual heat removal, respectively. The senior reactor analyst considered, in part, the plant conditions, availability of the steam generators as a heat sink and the low decay heat for each of the plant modes during which the condition existed, and the availability of the Auxiliary Saltwater Unit 2 crosstie in assessing the overall safety significance. It was also noted that the temperature at the packing gland was elevated following the pump run on May 30 but did not indicate early pump failure was likely. Based on the quantitative and qualitative assessment for this condition, the senior reactor analyst concluded the condition was of very low safety significance (Section 1R19).

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a violation of 10 CFR 20.1902, based on the area outside the drum compactor room on the 115-foot elevation of the auxiliary building was not posted as a radiation area. On May 6, 2002, the licensee performed a survey of the area which identified that general radiation levels were as high as 8 millirem per hour. However, on May 7, 2002, the inspectors found that the area was not posted as a radiation area. The failure to post a radiation area is a 10 CFR 20.1902 violation. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0554991.

  The issue was more than minor because the failure to post a radiation area has a credible impact on safety and the occurrence had the potential to involve a worker's unplanned dose if radiation levels had been significantly greater. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised (Section 2OS1).

- Green. The inspectors identified a violation of Technical Specification 5.7.1.a because the entrance to a high radiation area boundary surrounding the reactor vessel head on the 140-foot elevation of the containment building was not barricaded. General radiation levels in the area were as high as 120 millirem per hour. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0555046.

  The issue was more than minor because the failure to barricade a high radiation area has a credible impact on safety and the occurrence had the potential to involve a worker's unplanned dose if radiation levels had been significantly greater. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised (Section 2OS1).

B. <u>Licensee Identified Violations</u>

Violations of very low significance were identified by the licensee and have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. The violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status

Diablo Canyon Unit 1 began this inspection period at 100 percent power. On April 28, 2002, operators commenced a reactor shutdown for Refueling Outage 1R11 and entered Mode 3 (Hot Standby). Operators initiated a plant cooldown and entered Mode 5 (Cold Shutdown) on April 29, and then entered Mode 6 (Refueling) on May 2 when maintenance personnel de-tensioned the reactor vessel head. On May 4, operators commenced core offload, and the reactor was defueled as of May 6. Following outage work, Unit 1 re-entered Mode 6 when operators began reloading the core on May 11. Core reload was completed on May 13, and the reactor vessel head was tensioned on May 20, entering Mode 5. Operators commenced reactor coolant system heatup, and Unit 1 entered Mode 4 (Hot Shutdown) on May 24 and Mode 3 on May 25. On May 26, operators commenced a reactor startup, entering Mode 2 (Startup). Mode 1 (Power Operation) was achieved on May 28, and Refueling Outage 1R11 concluded on the same day when operators synchronized the main generator to the grid. Unit 1 power level was held at 70 percent to await repairs to Heater Drip Pump 2.

On June 1, 2002, operators began ramping Unit 1 offline from 70 percent power due to an electrohydraulic oil leak on a main turbine stop valve. The leak was repaired when reactor power reached 28 percent and the ramp was halted. Operators began to ramp reactor power back to 70 percent on the same day.

On June 3, 2002, the Unit 1 reactor automatically tripped as the result of a failed-closed feedwater regulating valve. Following repairs to the feedwater regulating valve, operators restarted Unit 1 and entered Mode 1 on June 4. Unit 1 reached 100 percent power on June 10 and remained at this power level for the rest of the inspection period.

Diablo Canyon Unit 2 began this inspection period at 100 percent power. On June 18, 2002, operators reduced Unit 2 power level to 55 percent to clean the circulating water system tunnels and repair a main condenser saltwater leak. After repairs were completed, operators began increasing power on June 21 and reached 100 percent power on the same day. Unit 2 remained at 100 percent power for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R04 Equipment Alignments (71111.04)

.1 Charging Pump 2-2 (Unit 2)

a. Inspection Scope

On June 12, 2002, the inspectors performed a partial system walkdown of the safety-related portion of Charging Pump 2-2 and its associated valves. The inspectors observed valve alignment, labeling, lubrication, ventilation, seismic supports, and absence of obstructions that may prevent the pump from performing its safety function. The inspectors also reviewed where the electrical power was available and the proper working condition of associated electrical equipment. The following documents were used during the inspection:

- Procedure OP B-1A:XI, “CVCS - Charging Pumps - Clearing for Maintenance and Returning to Service,” Revision 13A

- Drawing OVID 107709, “Refueling Water Storage Tank 2-1, Cold Leg Injection Lines,” Sheet 3, Revision 43

b. Findings

No findings of significance were identified.

.2 Auxiliary Feedwater Pump 1-1 (Unit 1)

a. Inspection Scope

On June 18, 2002, with Auxiliary Feedwater Pump 1-3 in a maintenance outage window, the inspectors reviewed proper system alignment of Auxiliary Feedwater Pump 1-1 and its associated valves. The inspectors observed valve alignment, labeling, lubrication, ventilation, seismic supports, and absence of obstructions that may prevent the pump from performing its safety function. The inspectors also considered the availability of electrical power and the proper working condition of associated electrical equipment. Cooling water for pump bearing cooling was also evaluated for proper working condition. The following documents were used during the inspection:

- Procedure OP D-1:II, “Auxiliary Feedwater System - Alignment Verification for Plant Startup,” Revision 27B

- Drawing OVID 106703, “Feedwater,” Sheet 3, Revision 61

b. Findings

No findings of significance were identified.

.3 Vital Batteries/Battery Chargers 1-1 and 1-3 (Unit 1)

a. Inspection Scope

On June 25, 2002, with Battery Charger 1-2 in a maintenance outage window, the inspectors reviewed the alignment and operational condition of Vital Batteries and Battery Chargers 1-1 and 1-3. The inspectors reviewed whether power was available, the condition of electrical equipment, labeling, ventilation, seismic supports, and electrolyte level in batteries. Procedure OP J-9:II, “Operating the Battery Chargers,” Revision 10, was used during the inspection effort.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Observations

a. Inspection Scope

The inspectors performed fire protection walkdowns to assess the material condition of plant fire detection and suppression, fire seal operability, and proper control of transient combustibles. The inspectors used Section 9.5 of the Final Safety Analysis Report Update as guidance. The inspectors considered whether the suppression equipment and fire doors complied with regulatory requirements and conditions specified in Procedures STP M-69A, “Monthly Fire Extinguisher Inspection,” Revision 31B, STP M-69B, “Monthly CO2 Hose Reel and Deluge Valve Inspection,” Revision 14, and STP M-70C, “Inspection/Maintenance of Doors,” Revision 6. Specific risk-significant areas inspected included:

- Units 1 and 2 vital 480 V bus rooms
- Units 1 and 2 vital battery rooms
- Units 1 and 2 cable spreading rooms
- Units 1 and 2 intake structure

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors reviewed the licensee’s flood protection measures for Units 1 and 2 to ensure that the licensee had taken adequate precautions to mitigate internal and external flood risks. The inspectors reviewed the licensee’s probabilistic risk assessment for external and internal flooding, Chapter 3 of the Final Safety Analysis Report - Update, and applicable controlled drawings in support of this inspection. The inspectors toured the intake to ensure that flood protection boundaries were functional.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Inspection Activities Other than Steam Generator Tube Inspections

Performance of Nondestructive Examination (NDE) Activities

The Diablo Canyon, Unit 1, inservice inspection (ISI) program is committed to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1989 Edition, without Addenda for the second 120-month inspection interval. The Refueling Outage 1R11 inservice inspections will complete the second period of the second 10-year interval of the program.

a. Inspection Scope

The inspectors observed portions of the specified Refueling Outage 1R11 ISI examinations listed below:

System	Component/Weld Identification	Examination Method
Reactor Coolant	Elbow / Pipe weld / WIB 69	Ultrasonic Examination
Reactor Coolant	1-2 Reactor Coolant Pump Main Flange Bolts 2-9	Ultrasonic Examination
Auxiliary Steam	Containment Penetration	VT-2 Visual Examination

During the performance of each examination, the inspectors reviewed whether the correct NDE procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation and equipment were properly calibrated and within the allowable calibration period. The inspectors reviewed the NDE certification packages of the contractor personnel to verify they had been properly certified in accordance with ASME Code requirements.

The inspectors reviewed the licensee's NDE records for work that was performed for the current outage. This review of NDE records was performed to verify NDE activities were performed in accordance with ASME Boiler and Pressure Vessel Code requirements and indications and defects, if present, were appropriately disposition. See the attachment for NDE records reviewed.

b. Findings

No findings of significance were identified.

.2 ASME Code Repair and Replacement Activities

a. Inspection Scope

The inspectors reviewed ASME Section XI Code repair and replacement packages for work performed, to verify repairs and replacements met ASME Code requirements. The ASME Section XI Code repair and replacement packages reviewed performed work to install new suction piping for centrifugal charging Pump 1-2 (WO C0165633 02), replace feedwater piping (WO C0173392 02), and replace reactor coolant Pump 1-2 seal injection chemical and volume control system Valves CVCS-1-292 & 580 (WO C0175198 01).

b. Findings

No findings of significance were identified.

.3 Refueling Outage 1R11 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors reviewed the licensee's in-situ screening criteria to verify that the criteria were in accordance with industry guidelines. The estimated size and number of tube wear flaws and other damage mechanisms identified up to the date of the inspection were compared to the operational assessment predictions from the previous outage. The inspectors also reviewed the eddy current examination scope and expansion criteria to determine if the Technical Specifications, industry guidelines, and commitments to the NRC were being met.

The inspectors reviewed the areas of potential degradation (based on site-specific and industry experience) to verify that such areas were being inspected. The inspectors also reviewed the Cycle 11 leakage history for the Unit 1 primary-to-secondary leak rate and noted that the operational leakage rate had been identified as less than 3 gallons per day. The eddy current probes and equipment were reviewed to ascertain if they were properly qualified for the expected types of tube degradation.

The inspectors observed the collection and analysis of eddy current data by licensee personnel to evaluate a possible loose part. The inspectors also reviewed action requests identified in the attachment.

b. Findings

No findings of significance were identified.

1R11 Operator Requalification (71111.11)

a. Inspection Scope

The inspectors witnessed operator performance in the simulator during routine training and requalification examinations. The inspectors also attended the crew and individual debriefs to determine if the evaluators critically assessed operator performance. On June 25, 2002, the inspectors observed a simulator scenario associated with a main steam line break and subsequent steam generator tube rupture.

The inspectors used Procedures EOP E-0, “Reactor Trip or Safety Injection,” Revision 27, E-2 “Faulted Steam Generator Isolation,” Revision 12A, and E-3 “Steam Generator Tube Rupture,” Revision 22, to support the inspection activities.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

.1 Routine Reviews

a. Inspection Scope

The inspectors reviewed the licensee’s Maintenance Rule implementation for equipment performance problems. The inspectors assessed whether the equipment was properly placed into the scope of the rule, whether the failures were properly characterized, and whether goal setting was recommended, if required. Procedure MA1.ID17, “Maintenance Rule Monitoring Program,” Revision 8, was used as guidance. The inspectors reviewed the following Action Requests (AR):

- AR A0534742, Reagent gas bottles for CEL-82 and -83 found isolated (Unit 2)
- AR A0549032, Manual reactor trip due to failed solenoid valve (Unit 2)
- AR A0551919, Goal setting review for overpressure protection functions (Unit 1)
- AR A0551921, Goal setting review for containment fan cooler Unit 2-4 (Unit 2)

b. Findings

No findings of significance were identified.

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

.1 Risk Assessments

a. Inspection Scope

The inspectors reviewed daily work schedules and compensatory measures to confirm that the licensee had performed proper risk management for routine and emergent work. The inspectors considered whether risk assessments were performed according to their procedures and the licensee had properly used their risk assessment tools. The inspectors reviewed the licensee's entry into appropriate risk categories, preservation of key safety functions, and implementation of work controls. The inspectors used Procedure AD7.DC6, "On-line Maintenance Risk Management," Revision 6, as guidance. The inspectors specifically observed the following work activities on Unit 1 during the inspection period:

- Vital 4160 Volt Bus F maintenance outage window on May 1, 2002
- Auxiliary Feedwater Pump 1-3 maintenance outage window on June 18, 2002

b. Findings

No findings of significance were identified.

.2 Emergent Work

a. Inspection Scope

The inspectors observed emergent work activities to verify that actions were taken to minimize the probability of initiating events, maintain the functional capability of mitigating systems, and maintain barrier integrity. The scope of work activities reviewed includes troubleshooting, work planning, plant conditions and equipment alignment, tagging and clearances, and temporary modifications. The following activities were observed on Unit 1 during this inspection period:

- Loss of charging to Vital Battery 1-2 on May 7, 2002

- Movement of Low Pressure Turbine C Casing from a restricted area on May 10, 2002

- Auxiliary Saltwater (ASW) Pump 1-2 emergent work on May 31, 2002

- Recovery from the Unit 1 reactor trip on June 3, 2002

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Nonroutine Plant Evolutions and Events

.1 Inadvertent Increase in Unit 2 Reactor Power

a. Inspection Scope

The inspectors evaluated operator response to an inadvertent increase in Unit 2 reactor power that occurred on April 14, 2002. The inspectors reviewed the licensee evaluation of AR A0552992 and used Procedure OP L-4, "Normal Operation at Power," Revision 39.

b. Findings

No findings of significance were identified.

.2 Unit 1 Reactor Trip

a. Inspection Scope

The inspectors evaluated operator response to a Unit 1 reactor trip that occurred on June 3, 2002, as a result of Feedwater Regulating Valve FW-1-FCV-510 failing closed. A fitting in the air supply system broke off and bled off the air, closing the valve on loss of air pressure. The inspectors responded to the control room and observed operator performance following the reactor trip. In addition, the inspectors evaluated the licensee posttrip review to ensure that the licensee sufficiently determined the apparent cause of the trip, that all required safety system actuations occurred, and that the plant responded as expected. The inspectors used Nonconformance Report N0002147 to support this inspection.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations and supporting documents to determine if the associated systems could meet their intended safety functions despite the degraded status. The inspectors reviewed the applicable Technical Specifications, Codes/Standards, and Final Safety Analysis Report Update sections in support of this inspection. The inspectors reviewed the following ARs and Operability Evaluations (OE):

- A0553215, Oil container parts in centrifugal charging Pump 2-2 (Unit 2)

- A0558328, Auxiliary feedwater system leakage high temperature alarms (Unit 1)

- A0560008, Stroke time for Valve CVCS-1-8149A below action low value (Unit 1)

- OE 2002-03, Operability of Units 1 and 2 Nuclear Instrumentation System Positive Flux Rate Trip

- OE 2002-04, Revision 1, Operability of Units 1 and 2 with normally energized solenoid-operated valves that were in service beyond their qualified life

- OE 2002-05, Operability of Units 1 and 2 with solenoid-operated valves having diodes and splices that lack environmental qualification

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

Operator workarounds may impact the functionality of mitigating systems or the operator's ability to carry out the abnormal/emergency procedures. The inspectors reviewed the operator workaround log and examined the following two systems that were affected by operator workarounds:

- Units 1 and 2, Pressurizer Safety Valve Loop Seal Temperatures
- Units 1 and 2, Operator Field Communication Systems

In both workarounds, the inspectors analyzed the magnitude of impact on mitigating systems and operator response. The inspectors also reviewed procedure changes, training, and corrective action plans related to the operator workarounds. The following ARs were used during the inspection effort:

A0473008	A0507157	A0509165
A0524359	A0525302	A0526416
A0548936		

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed postmaintenance tests for selected risk-significant systems to verify their operability and functional capability. As part of the inspection process, the inspectors witnessed and/or reviewed the postmaintenance test acceptance criteria and results. The test acceptance criteria was compared to the Technical Specifications and the Final Safety Analysis Report Update for the Diablo Canyon Power Plant. Additionally, the inspectors concluded that the test was adequate for the scope of work, the test was performed as prescribed, jumpers and test equipment were properly removed after the test, and test equipment range, accuracy, and calibration were consistent for the application. The following are selected corrective maintenance activities reviewed by the inspectors:

- Unit 1 Check Valve CVCS-1-8479B, Replace valve on May 7-8, 2002

- Unit 1 Auxiliary Feedwater Pump 1-3, Coupling removal, inspection, and installation and marking the magnetic center on May 9-25, 2002

- Unit 1 Motor-Operated Valve CS-1-9001B, Adjusted torque limit switch and red light indicator on May 10, 2002

- Unit 1 ASW Pump 1-2, Replaced pump on May 10-15, 2002

- Unit 1 Diesel Generator 2-2, Adjust governor frequency on June 6-7, 2002

- Unit 1 Boric Acid Transfer Pump 1-1, Change oil and clean and inspect bearings, seals, and coupling on June 21, 2002

b. Findings

A violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” was identified for the failure to establish an adequate postmaintenance test to verify ASW Pump 1-2 operability. This finding was of very low safety significance (Green).

During Refueling Outage 1R11, the licensee installed new packing on ASW Pump 1-2 as part of a pump replacement that occurred between May 9-16, 2002. The licensee performed a postmaintenance test of the pump on May 17, 2002, and documented that there was adequate packing leak-off flow. The pump was declared operable at this time. An additional postmaintenance test was performed on May 21 to obtain sufficient data to re-baseline the pump for in-service testing purposes. During the second test, verification of packing leak-off was not required by the test procedure and no data was taken regarding the leak-off flow rate.

On May 30 operators started ASW Pump 1-2 and identified there was no packing leak-off flow. After maintenance personnel loosened the packing to its maximum limit, the packing leak-off flow was still below 60 drops per minute, which is the low action level listed in Procedure STP P-ASW-A, “Performance Test of Auxiliary Saltwater Pumps,” Revision 15. Operators secured ASW Pump 1-2 and declared it inoperable. Licensee analysis of the pump packing indicates that the packing rings were too long and the shape of the top packing ring was deformed. The licensee re-packed ASW Pump 1-2 with packing rings that were verified to be of the appropriate size before installation, and they removed one packing ring to prevent the top ring from becoming deformed.

The inspectors evaluated the as-found condition of ASW Pump 1-2 and determined the failure to establish an adequate postmaintenance test, to ensure ASW pump operability, to have a credible impact on safety. The inspectors review concluded that the failure to perform an adequate postmaintenance test resulted in a condition where the ASW pump would start but could subsequently fail in a manner that would result in the pump not meeting its safety function. The required time for ASW Pump operation is governed by the systems it supports. The ASW system provides heat removal from the component cooling water (CCW) system. In turn, the CCW system provides heat removal for emergency core cooling system pump bearings and seals, reactor coolant pump thermal barrier heat exchangers, and residual heat removal system heat exchangers. Loss of the ASW system would impact the plant’s ability to perform initial and long-term decay heat removal in the event of an accident. For redundancy, there are two normally crosstied ASW pumps for each unit and there is the capability to crosstie ASW systems between Units 1 and 2. In addition, there are diverse methods to provide core cooling in the event the ASW system is not available, using a combination of the turbine-driven auxiliary feedwater pump and steam generator power-operated relief valves, or the centrifugal charging pump, fire water system, and pressurizer power-operated relief valves.

The issue is more than minor since the same as-found condition, under the same circumstances for a longer period of time, would have a greater safety significance. An NRC senior reactor analyst performed a Significance Determination Process Phase 3 evaluation. The senior reactor analyst reviewed the licensee’s risk assessment, the insights obtained from the NRC’s Standardized Plant Analysis Risk (SPAR) model for Diablo Canyon Units 1 and 2 (Revision 3i) and NRC Manual Chapter 0609, Significance Determination Process, Appendix A, Significance Determination of Reactor Inspection Findings for At-Power Situations and Appendix G, Shutdown Safety SDP for those conditions utilizing residual heat removal.

The licensee’s risk evaluation established an allowed outage time of 286 hours for the plant in Mode 3 with the turbine driven auxiliary feedwater pump unavailable. An allowed outage time of 370 hours was established for Mode 1 operation with the turbine driven auxiliary feedwater pump operable. These allowed outage times included external events. The period the ASW Pump 1-2 was assessed as being unavailable was approximately 144 hours. This analysis specified that the issue was of very low safety significance. The SPAR model identified that the small break loss of coolant accident with loss of component cooling water was the most safety significant sequence for the loss of all ASW. The senior reactor analyst reviewed the most risk significance sequences for a loss ASW and considered the specific plant conditions that existed during the 6 day period. These conditions included the availability of the steam generators as a heat sink with relatively low decay heat for each of the plant modes during which the condition existed and the availability of fire water and the ASW Unit 2 crosstie. The systems needed to mitigate a small break loss of coolant accident would be available during the 144 hour period the ASW pump was assessed as being unavailable. Based on the quantitative and qualitative assessment for this condition, the senior reactor analyst concluded the condition was of very low safety significance (Green).

The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for the failure to perform an adequate postmaintenance test on safety-related equipment prior to placing it in service. 10 CFR Part 50, Appendix B, Criterion XI, requires that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Contrary to the above, the licensee installed new packing on ASW Pump 1-2 as part of a pump replacement that occurred between May 9-16, 2002. The postmaintenance test performed on May 17, 2002, was not adequate to ensure operability of the pump. Specifically, the postmaintenance test was not adequate to identify that the packing had been improperly installed and that the packing had shifted and swelled following the May 17 pump run. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This item was placed in the corrective action system as AR A0560036 (Noncited violation (NCV) 275/2002003-01).

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors witnessed and evaluated licensee performance during the 11th refueling outage for Unit 1. The outage lasted from April 28 to May 28, 2002. Before and during the outage, the inspectors evaluated the licensee’s consideration of risk in developing outage schedules, use of risk reduction methodologies in control of plant configurations, development of mitigation strategies for losses of key safety functions, and adherence to the operating license and Technical Specification requirements. Specifically, the inspectors observed the licensee’s actions in the following areas:

- Outage risk control plan, prior to and during implementation
- Mode transitions from power operation (Mode 1) to defueled reactor vessel and then the return to power operation
- Defense-in-depth and handling of unexpected conditions
- Plant configuration control, particularly clearance of equipment
- Supply and control of electrical power in regard to Technical Specification requirements and outage risk plans
- Adequacy of decay heat removal for the reactor vessel, refueling cavity, and spent fuel pool
- Fuel assembly movement, tracking, and inspections
- Containment closure and containment closure capability with respect to the Technical Specifications and outage risk plans

- Adequate control of reduced inventory and midloop conditions

- Movement of heavy loads inside containment and the turbine building

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

Routine Observations

a. Inspection Scope

The inspectors evaluated several routine surveillance tests to determine if the licensee complied with the applicable Technical Specification requirements to demonstrate that equipment was capable of performing its intended safety functions and operational readiness. The inspectors performed a technical review of the procedure, witnessed portions of the surveillance test, and reviewed the completed test data. The inspectors also considered whether proper test equipment was utilized, there was no preconditioning, test acceptance criteria agreed with the equipment design basis, and equipment was returned to normal alignment following the test. The following tests were evaluated during the inspection period:

- Procedure STP M-77, “Safety and Relief Valve Testing,” Revision 25, on April 4, 2002, for Unit 1

- Procedure STP M-8C3, “Leak Rate Testing Penetration 58 Mini Equipment Hatch Seal,” Revision 3, on April 30, 2002, for Unit 1

- Procedure STP V-620, “Penetration 20 Containment Isolation Valve Leak Testing,” Revision 6, on May 1, 2002, for Unit 1 (CCW return lines from Reactor Coolant Pump Bearing Lube Oil Coolers and Reactor Vessel Supports Coolers)

- Procedure STP V-630, “Penetration 30 Containment Isolation Valve Leak Testing,” Revision 21, on May 1, 2002, for Unit 1 (Containment Spray Isolation Valves)

- Procedure TP TB-9501, “MOV Flow Test – AFW Pump 1 Turbine Steam Supply Valve FCV-95,” Revision 2, on May 29, 2002, for Unit 1

- Procedure STP M-9D1, “Diesel Generator Full Load Rejection Test,” Revision 10, on May 10, 2002, for Unit 1

- Procedure STP M-15, “Integrated Test of Engineered Safeguards and Diesel Generators,” Revision 37, on May 19, 2002, for Unit 1

- Procedure STP X-1, “Visual Examination of the Reactor Vessel Interior,” Revision 2, on May 16, 2002, for Unit 1

- Procedure STP V-15, “ECCS Flow Balance Test,” Revision 23, on May 12, 2002, for Unit 1

- Procedure STP M-9G, “Diesel Generator 24-Hour Load Test and Hot Restart Test,” Revision 34, on May 10, 2002, for Unit 1

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following temporary modifications/plant jumpers. The inspectors reviewed the 10 CFR 50.59 screenings to verify that the applicable drawings were annotated. The inspectors also observed the required tag information and placement and, if required, that transient combustible administrative controls were properly implemented. The temporary alterations were performed in accordance with Procedure CF4.ID7, “Temporary Modifications,” Revision 9A.

- Units 1 and 2, Installation of a temporary seismic recorder, AR A0559764

- Unit 1, Installation of a fiber-optic cable tube for Reactor Coolant Pump 1-3, AR A0555999

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during Refueling Outage 1R11 operations to assess the licensee’s exposure control programs. The inspectors also conducted plant tours within the radiologically controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Nuclear Quality Services Audits: EDMS-011770001, “2001 Radiation Protection Program,” EDMS-013130017, “2002 Radioactive Effluent Controls Program,” and Radiation Protection Assessment Report 013410056, “Performance of Containment Entries at Power”

- Area postings and other controls for airborne radioactivity areas, radiation areas, high radiation areas, locked high radiation areas, and very high radiation areas

- Radiological surveys involving airborne radioactivity areas and high radiation areas

- Locked high radiation area key control program

- Access controls, surveys, and radiation work permits for the following four significant high dose work areas during Refueling Outage 1R11: RWP 02-1050-0, “RCP Maintenance,” RWP 02-1070-0, “Rx Head Penetration Inspections,” RWP 02-1042-0, “S/G Nozzle Dam Installation and Removal,” and RWP 02-1020-0, “Reactor Disassembly”

- Dosimetry placement when work involved a significant dose gradient

- Controls involved with the storage of highly radioactive items in the spent fuel pool

- A summary of operational radiation protection corrective action documents written since May 1, 2001 (19 of these documents were reviewed in detail: A0530581, A0530808, A0532219, A0534120, A0536016, A0536999, A0537015, A0537485, A0539934, A0539937, A0539939, A0541207, A0543771, A0543828, A0547721, A0547888, A0549159, A0553220, and A0553835)

b. Findings

.1 Failure to post a radiation area

The inspectors identified a violation with very low safety significance (Green) for the licensee’s failure to post a radiation area. On May 6, 2002, the licensee performed a survey of the area outside the drum compactor room on 115-foot elevation of the auxiliary building and documented that general radiation levels were as high as 8 millirem per hour. However, on May 7, 2002, inspectors found that the area was not posted as a radiation area.

The issue was more than minor because the failure to post a radiation area has a credible impact on safety and the occurrence had the potential to involve a worker’s unplanned dose if radiation levels had been significantly greater. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA (as-low-as-is-reasonably-achievable) finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

10 CFR 20.1003 defines a radiation area as an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 millirem in an hour at 30 centimeters from the radiation source. 10 CFR 20.1902 requires each radiation area be posted with a conspicuous sign or signs. The failure to post the above area as a radiation area is a violation of 10 CFR 20.1902. This violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as AR A0554991 (NCV 275/2002-03-02).

.2 Failure to barricade a high radiation area

A violation with very low safety significance (Green) was identified for a failure to barricade a high radiation area. On May 8, 2002, the inspectors identified that a high radiation area boundary surrounding the reactor vessel head on the 140-foot elevation of the containment building was down and did not barricade the entrance. General radiation levels in the area were as high as 120 millirem per hour.

The issue was more than minor because the failure to barricade a high radiation area has a credible impact on safety, and the occurrence had the potential to involve a worker's unplanned dose if radiation levels had been significantly greater. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

10 CFR 20.1003 defines a high radiation area as an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 100 millirem in an hour. Technical Specification 5.7.1.a states, in part, that each entrance to a high radiation area shall be barricaded. The failure to barricade the above area is a violation of Technical Specification 5.7.1.a. This violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as AR A0555046 (NCV 275/2002-03-03).

4. OTHER ACTIVITIES

40A1 Performance Indicator Verification (71151)

.1 Reactor Safety Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed the following performance indicators for each quarter of 2001 to assess the accuracy and completeness of the indicator. The inspectors reviewed plant operating logs and licensee monthly operating reports to support this inspection. The inspectors used NEI 99-02, "Regulatory Assessment Performance Indicator Verification," Revision 2, as guidance for this inspection.

- Units 1 and 2 Auxiliary Feedwater Availability
- Units 1 and 2 High Pressure Safety Injection Availability
- Units 1 and 2 Residual Heat Removal Availability

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors reviewed corrective action program records for high radiation areas, locked high radiation areas, and unplanned exposure occurrences for the past 12 months to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area exit transactions with exposures greater than 100 millirem for the past four quarters were also reviewed. Selected examples were investigated to determine whether they were within the dose projections of the governing radiation work permits.

b. Findings

No findings of significance were identified.

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors reviewed radiological effluent release program corrective action records, licensee event reports (LERs), and annual effluent release reports documented during the past four quarters to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) LER 275; 323/2001-001-00: Automatic diesel generator start upon loss of startup power due to 230 kV line arcing in heavy smoke from escaped fire caused by inadequate administrative controls.

This LER discussed an event in which both units lost one source of offsite power (startup transformers) because of heavy smoke near a transmission tower during a controlled burn. All six diesel engine generators (both units) started as required. This event was discussed in detail in NRC Special Inspection Report 50-275; 323/2001-10.

The LER provided additional detailed information with respect to long-term corrective actions not discussed in the inspection report, but the inspectors determined that no additional NRC action is necessary. This item is closed.

.2 (Closed) LER 323/2001-003-00: Technical Specification 3.3.3 not met due to inadequate procedure.

This LER discussed an event in which both Unit 2 containment hydrogen analyzers (CEL-82 and -83) were inoperable for a period in excess of the limiting conditions for operation action statement. Maintenance technicians found Valve VAC-2-672, the isolation valve for the reagent gas supply for the hydrogen analyzers, closed when performing a surveillance on May 30, 2001. Licensee investigation revealed that Valve VAC-2-672 was likely closed on May 23, 2001, during a previous surveillance test. Technical Specification 3.3.3 requires that, if both hydrogen analyzers are inoperable for 72 hours, the licensee place the plant in Mode 3 (hot shutdown) within the next 6 hours. Because the Unit 2 hydrogen analyzers were inoperable for 7 days, a violation of Technical Specification 3.3.3 occurred, which is further discussed in Section 4OA7 of this inspection report. This item is closed.

.3 (Closed) LER 50-275/2001-003-00: Technical Specification 3.7.6 not met when the fire water storage tank was isolated from the auxiliary feedwater pump suction because of personnel error.

On September 13, 2001, the licensee discovered Valve MU-1-297, manual isolation from fire water storage tank to condensate storage tank, closed. The licensee determined that personnel had closed the isolation valve on June 22, 2001, to stop a raw water leak into the condensate system. Since the allowed outage time expired on June 29, 2001, the licensee had been in violation of Technical Specification 3.7.6 for 82 days. The closure of Valve MU-1-297 violated the Technical Specification because it eliminated the redundancy to prevent a single failure as specified in Supplemental Safety Evaluation Report 8.

The inspectors concluded that closure of the valve had more than minor significance, since it could have had a credible impact on safety (Inspection Manual Chapter *0610, Group 1, Question 1) in that it eliminated a redundant water source in the event of a seismic event. Similarly, the closed isolation valve affected operability of the suction water source to the auxiliary feedwater pumps (Inspection Manual Chapter *0610, Group 2, Question 2). Using the Significance Determination Process Phase 1 screening worksheet related to mitigating systems, the inspectors determined closure of Valve MU-1-297 affected the secondary heat removal capability of the auxiliary feedwater system. Updated Final Safety Analysis Report, Section 6.5.2.1.1 identifies several water sources for auxiliary feedwater in addition to the fire water storage tank. Because of these additional water sources combined with the time available to refill the condensate storage tank, the closure of Valve MU-1-297 did not result in loss of a safety function. The inspectors concluded that this was not potentially risk significant because of fire, seismic, flood, or severe weather event. Consequently, the inspectors concluded that this licensee-identified violation had very low safety significance (Green). This finding is summarized in Section 4OA7.

40A5 Other

.1 Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles (NRC Bulletin 2001-01)

Temporary Instruction 2515/145 provided guidelines to verify compliance with licensee commitments to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." Further, this evaluation confirmed licensee compliance with applicable regulatory requirements (i.e., 10 CFR Part 50, Appendix A; 10 CFR Part 50, Appendix B; and 10 CFR 50.55a). As identified in the Temporary Instruction, Diablo Canyon Unit 1 fell within the category of moderate-susceptible plants. Consequently, the inspectors used the criteria for evaluating moderate-susceptible plants to conduct this inspection.

a. Inspection Scope

The inspectors conducted this performance-based evaluation and assessment to ensure that the NRC had an independent review of the condition of the reactor vessel head and vessel head penetrations. The inspectors assessed the effectiveness of the licensee examinations of the vessel head penetrations. Specifically, the inspectors: (1) reviewed the examination criteria used by the examiners, (2) interviewed the examiners, (3) evaluated the training conducted to support the examinations, (4) assessed adequacy of the examination plan and procedures, (5) observed in-process examinations, (6) evaluated the quality and resolution of the examination equipment, (7) reviewed completed records, (8) considered whether the licensee documented deficiencies in their corrective action process, and (9) assessed the overall effectiveness of the process used to perform the bare metal visual examination.

The inspectors reviewed the following documents during this inspection:

- Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants,” dated March 17, 1988

- Information Notice 90-10, “Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600,” dated February 23, 1990

- Information Notice 2001-05, “Through-wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3,” dated April 30, 2001 [ML011160588]

- MRP-44, Part 2, EPRI TR-1001491, “PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants, Part 2: Reactor Vessel Top Head Penetrations,” dated May 2001

- NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” dated August 3, 2001 [ML012080284]

- Response to NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” dated August 30, 2001 [ML012570306]

- MRP-48, EPRI TR-1006284, “PWR Materials Reliability Program Response to NRC Bulletin 2001-01,” dated August 2001

- Supplement to Response to NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” dated November 26, 2001

- Diablo Canyon Nuclear Power Plant, Units 1 and 2 Response to Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” (TAC Nos MB2627 and MB2628), dated February 26, 2002

- Visual Examination for Leakage of Reactor Head Penetrations on Top of Head, dated August 10, 2001

- Information Notice 2002-11, “Recent Experience With Degradation of Reactor Pressure Vessel Head,” dated March 12, 2002 [ML020700556]

- NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” dated March 18, 2002 [ML020770497]

- Information Notice 2002-13, “Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradations,” dated April 4, 2002 [ML020930617]

- 15-Day Response to NRC Bulletin 2002-01, “Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity,” dated April 1, 2002

- Procedure ISI VT 2-1, “Visual Examination During Section XI System Pressure Test,” Revision 5, dated March 15, 2001

- Procedure ISI X-CRDM, “Reactor Vessel CRDM Inspection,” Revision 2, dated April 26, 2002

- Procedure TQ1.DC84, “Qualification and Certification of Plant Visual Examiners,” Revision 3, dated February 18, 1999

- Procedure TQ1.DC12, “Qualification and Certification of NDE Personnel,” Revision 1, May 29, 2001

- ALARA Prejob Planning Package

- Diablo Canyon Power Plant, Unit 1 Head Penetration Exam Scan Plan

- VT-2 Qualification Records for the Vessel Head Examination personnel

- Excerpts from Dominion Engineering, Inc. Report R-3818-00-1, “Reactor Vessel Top Head Nozzle Operating Fit Analysis, Diablo Canyon Power Plant,” Revision 0, dated April 2002

- Drawing 232-448, “Closure Head Assembly for Westinghouse Electric Corporation 173" I.D. Reactor Vessel.”

- Rejection Notice 3658, “Penetration 56 has a gouge at O.D. of closure head dome,” dated February 19, 1970

- AR A0530166, Address Oconee Head Leaks Due to Head Pen Weld Cracks

- Event Investigation Report 88-01, “Control Rod Drive Mechanism Leak Event,” dated December 20, 1988

The inspectors observed two peripheral in-process vessel head penetration examinations performed using an articulated video probe. The inspectors evaluated the videotape for each of the vessel head penetration examinations. The examiners used the articulated video probe to examine eight center vessel head penetrations because the low clearance of the insulation rendered them inaccessible to the robot. Similarly, the licensee examined 10 of the peripheral vessel head penetrations using the articulated video probe because they did not desire to remove the insulation to keep exposures ALARA. In addition, the inspectors visually evaluated the condition of the reactor vessel head from the two control rod drive mechanism cooling duct holes in the shroud.

b. Findings

No findings of significance were identified.

The inspectors concluded that the licensee performed a good, qualified, bare metal examination of the vessel head penetrations. The clarity and resolution of the examination equipment combined with the training, qualification, and procedures ensured that the examiners could detect small boron deposits. The inspectors have provided the following details of the inspection as required by Temporary Instruction 2515/145.

Examination

The licensee established two teams with three ISI examiners on each team in order to perform the examinations in an expedited manner. One individual on each team had practiced driving the robot on a full scale mockup; also, these same individuals had obtained experience performing similar examinations at another facility using their robot. The three examiners coordinated the examination of the reactor vessel head penetration nozzles. One individual drove the robot or moved the articulated video probe according to the scan plan. A second individual verified the location on the scan plan and voice overlaid the video tape with tape counter index value for the nozzle quadrant(s) being reviewed. The third individual independently verified the location on a second scan plan and documented the digital tape counts. The examiners established an indexing routine that evaluated the vessel head penetration nozzles in quadrants. The inspectors concluded that the scan plan implemented during the examinations ensured that the licensee had inspected all nozzles 360° around the nozzle circumference.

The inspectors interviewed the personnel who performed the VT-2 examinations of the reactor vessel head. The examiners used a articulated video probe in difficult to reach locations and a robot for the remainder of the vessel head penetrations. The inspectors verified that the examiners had current VT-2 qualification records and noted that three of the five examiners had Level III qualifications. During interviews, the inspectors found that all the examiners had a minimum of 20 years experience and knew how to identify indications of boric acid leakage. The examiners had seen numerous photographs detailing leakage from vessel head penetrations. The licensee provided a training session that included: (1) the examination criterion and (2) photographs of vessel head penetrations with leakage and previously existing leakage stains.

The inspectors verified that Procedure ISI X-CRDM provided: (1) explicit descriptions of the types of boric acid indications that might be identified, (2) appropriate descriptions of the conduct of the examination (i.e., use of the scan plan), and (3) sufficient guidance to satisfy licensee commitments for inspection of the vessel head penetration nozzles and the general surface of the reactor vessel head. The inspectors concluded that the procedure combined with the training had provided adequate guidance for the examiners to identify, disposition, and resolve deficiencies. The procedure stated that any deposits of boric acid shall be immediately investigated and emphasized that boric acid in the juncture of a vessel head penetration and the vessel head were of prime concern.

The inspectors reviewed the in-process sheet used to document the inspections and reviewed the completed surveillance documented in Procedure ISI X-CRDM. The work package accurately documented the condition of the reactor vessel head, documented the examination of each vessel head penetration, identified the qualification of the test equipment used, and identified personnel who performed the work. In addition to this quality record, the licensee had videotaped the examination process and had indexed the penetrations. The inspectors verified that the licensee had performed the required qualification examinations for the robot and the articulated video probe as required by the ASME code.

The licensee had designed and fabricated a robot especially for performing this bare metal examination of the vessel head and its penetrations. The robot measured approximately 7 inches long by 7 inches wide by 2¼ inches high. In anticipation of potential problems, the licensee developed: (1) a scraping tool, (2) an excavating tool, (3) a guide tube for the articulated video probe, and (4) an attachment that held a nitrogen supply tube for blowing debris. These modifications to the robot would have allowed the examiners to move debris and evaluate any questionable debris, deposits, or indications. The examiners used the nitrogen supply tube to remove the debris from the vessel head penetration area, which allowed for a clear unobstructed view of the vessel head penetration to vessel head joint.

The inspectors noted that the high resolution video equipment enabled the examiners to easily discern the type of debris (e.g., metal shavings or paint chips) located at the vessel head penetration area. However, the examiners did not always remove the debris prior to using the rear camera on the robot. The inspectors determined that the camera on the robot provided up to 400 lines of resolution and concluded that the articulated video probe multiplied the images by a factor of 2 or 3, provided excellent resolution, and allowed Jaeger J-1 images to be easily discerned.

Condition of the reactor vessel head

The inspectors noted, during the direct visual evaluation through the control rod drive mechanism cooling duct openings, that the reactor vessel head had no indications of boric acid leakage nor any boric acid stains. The inspectors estimate that this review allowed them to assess directly the condition of approximately 40 percent of the reactor vessel head.

The inspectors noted that this vessel head had a significant amount of metal shavings scattered on the vessel head in general and collected on the uphill side of the vessel head penetrations. The licensee had cut and capped Vessel Head Penetrations 17, 19, 27, and 28 during Refueling Outage 1R2 after identifying leaking canopy seal welds on these spare vessel head penetrations. The licensee had also placed mechanical clamps on an additional eight spare vessel head penetrations as a preemptive measure since spare vessel head penetrations tend to develop leaks after the stagnant primary water concentrates chlorides in crevices formed by lack of weld penetration. The licensee indicated that the shavings and debris most likely resulted from the cut and cap activities during Refueling Outage 1R2. The inspectors noted that the color of the shavings matched the caps and nozzle assemblies located on the vessel head penetrations.

The inspectors noted that the vessel head had been painted with two coats of heat resisting aluminum paint. The inspectors noted that: (1) all of the vessel head penetration assemblies had a ring of paint on the nozzle assembly at the base where it met the vessel head and (2) much of the vessel head had paint adhered in spots and peeling. The inspectors noted that this was easily discernable during the inspection process and did not interfere with the conduct of the examination.

The inspectors noted that Vessel Head Penetration 56 had an indentation during review of the video tape and no presence of boric acid although paint was present. The examiners concluded that the indentation in the area of the vessel head penetration was a manufacturing defect. Upon questioning, the licensee indicated that they recalled that the vessel head had been supplied with this defect. The inspectors verified that: (1) Rejection Notice 3658 documented acceptability of a gouge near the outside diameter of Vessel Head Penetration 56, (2) the rejection notice referenced the fabrication drawing supplied with the Diablo Canyon Unit 1 vessel head, and (3) the rejection notice indicated the gauge was ½ x 1¾ and acceptable if the rough edges were ground.

Capability to identify and characterize small boron deposits

The inspectors concluded that the examiners and equipment used during the examinations could reliably detect and accurately characterize any identified leakage. The inspectors verified that the examiners: (1) consisted of the same group of individuals, (2) received the training on different types of boric acid indications, particularly what to expect if a leak occurred from a cracked vessel head penetration weld, (3) had a process for evaluation of deposits, and (4) used equipment with appropriate resolution. Since the work went around the clock, they used two separate teams of examiners.

During evaluation of the videotapes of the vessel head penetrations, the inspectors noted that the equipment provided excellent clarity that allowed for a complete evaluation. The inspectors found it easy to distinguish the size, type, consistency and configuration of any identified debris.

Material deficiencies identified that required repair

None.

Impediments to effective examinations and/or ALARA issues

The inspectors concluded that, in general, the licensee encountered no impediments to performing a qualified bare metal examination of the vessel head penetration nozzles. The inspectors noted that the licensee had to use an articulated video probe in the areas near the top of the vessel head because the clearance was too low. The licensee had purchased an articulated video probe with a 30-foot cable to ensure easy access to the center vessel head penetrations. The licensee installed a guide tube on their robot to create a platform for feeding the articulated video probe cable. Also, the licensee had to use an articulated video probe on the peripheral vessel head penetrations because of inadequate clearance for the robot. The licensee used wedges to raise the insulation to allow access by the articulated video probe.

The inspectors noted that, on the periphery where the insulation corners met, the examiners had to be careful to ensure that all quadrants were examined because of the limited space. Upon questioning whether Vessel Head Penetration 67 had been evaluated 360° around the circumference, the examiners demonstrated that they had performed a complete examination.

The Digital Rod Position Indication stacks are a stagnant area, consequently, during operating Cycle 9 contaminated Cobalt-60 particles migrated into the digital rod position indication stacks and increased the source term. The licensee determined that the dose rate at the vessel head was approximately 1 R. Because of this dose rate, the licensee held numerous meetings planning for this task in order keep exposure ALARA. Some of the things implemented to reduce the dose received by all personnel included: (1) developing a mockup of the vessel head to allow the examiners to practice under actual conditions (dressed out) using the articulated video probe and the robot, (2) using the articulated video probe for the inner most penetrations, (3) leaving the insulation in place and raising it with wedges on the peripheral examinations to limit the dose, (4) adding handles to the insulation panels so that future examinations, if required, will result in less exposure to the insulators, (5) performing the examinations from the same shielded bunker used for many other ISI activities, and (6) photographing the condition of the insulation on the vessel head prior to the insulators beginning their removal to reduce exposure time. Coincidently, two of the examiners had performed a vessel head examination at another facility that provided another opportunity to practice manipulating the robot and the articulated video probe.

.2 NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

a. Inspection Scope

As discussed in Section I, the inspectors evaluated the general condition of the reactor vessel head to determine whether the licensee had identified any evidence of wastage similar to that described in NRC Bulletin 2002-01. Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants," required that the licensee establish a program to evaluate corrosion of ASME Class 1 carbon steel components. Consequently, the inspectors evaluated the licensee commitments to Generic Letter 88-05 and reviewed the results of their inspections of ASME Class 1 components.

The inspectors reviewed the program and procedures that implemented the inspections committed to in the licensee response to Generic Letter 88-05. The inspectors reviewed completed tests performed for the last two outages on each unit and interviewed ISI personnel who participated in the inspections.

The inspectors reviewed the following documents during this inspection:

- Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants," dated March 17, 1988

- Response to Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants," dated June 2, 1988

- Closeout of Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants," dated February 24, 1989

- Procedure AD4.ID2, "Plant Leakage Evaluation," Revision 4A, dated March 26, 2002

- Procedure STP R-8A, "Reactor Coolant System Leakage Test," Revision 11, dated November 30, 2001

- Procedure STP R-8C, "Containment Walk-on for Evidence of Boric Acid Leakage," Revision 7, dated March 28, 2002

- Procedure ISI VT 2-1, "Visual Examination During Section XI System Pressure Test," Revision 5, dated March 15, 2001

- AR listings related to walkdowns completed in accordance with Procedures STP R-8A and STP R-8C initiated since 1998

- Completed tests for the past two outages on each unit for both Procedures STP R-8C and STP R-8A

- Drawing 102028, "ASME Code Boundaries for Inservice Inspection and Testing Program-ISITP

- Plant Manual Volume 9B, Section VII, "NRC Generic Letter 88-05 Component List - Carbon Steel Reactor Coolant Pressure Boundary Components (Including Supports); and Potential Sources of Boric Acid Leakage," Revision 2, dated May 18, 1990

- AR A0516915, Industry Event - V.C. Summer Reactor Vessel to Hot Leg Pipe Crack

The inspectors confirmed that Procedures AD4.ID2, STP R-8A, STP R-8C, and ISI VT 2-1 implemented their commitments to Generic Letter 88-05. As specified in the response to NRC Bulletin 2002-01, procedures required review for any leakage from the reactor vessel head with the insulation installed. The inspectors verified that licensee procedures met the requirements in the ASME code for inspecting insulated components.

From review of completed Procedures STP R-8A and STP R-8C and corrective maintenance ARs, the inspectors determined that the licensee performed detailed walkdowns that identified numerous leaking components that were properly dispositioned. From interviews the inspectors found that the licensee conducts a 100 percent walkdown of all components inside containment to evaluate for leakage. This was done to ensure that deficient components will be identified so that they may be maintained in good working order. As an aide to help engineers disposition components identified as having leaks, personnel performing the walkdowns obtained digital photographs of the as-found condition.

b. Findings

No findings of significance were identified.

.3 Evaluation of Diablo Canyon Safety Condition in Light of Financial Conditions

a. Inspection Scope

Because of the licensee's financial condition, Region IV initiated special review processes for Diablo Canyon. The resident inspectors continued to evaluate the following factors to determine whether the financial condition and power needs of the station impacted plant safety. The factors reviewed included: (1) impact on staffing, (2) corrective maintenance backlog, (3) corrective action system backlogs, (4) changes to the planned maintenance schedule, (5) reduction in outage scope, (6) availability of emergency facilities and operability of emergency sirens, and (7) grid stability (i.e., availability of offsite power to the switchyard, status of the operating reserves especially at the onset of rolling blackouts, and main generator volt-ampere reactive loading).

Additionally, the resident inspectors observed the energy supply and operating reserves available in the California market. Inspectors have also increased attention to areas such as employee morale, licensee activities, and specific technical issues.

b. Findings

No findings of significance were identified.

40A6 Management Meetings

Exit Meeting Summary

The resident inspection results were presented on July 9, 2002, to Mr. Jeffrey A. Hays, Director - Maintenance Services, and other members of licensee management. The licensee acknowledged the finding presented. Discussion of region-based inspection results are described in the following paragraphs.

The radiation protection inspection results were presented to Mr. James R. Becker, Station Director, and other members of licensee management at the conclusion of the inspection on May 10, 2002. The licensee acknowledged the findings presented.

The ISI activities inspection results were presented to Mr. David H. Oatley on May 17, 2002. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information is contained in the inspection report.

40A7 Licensee Identified Violations

The following findings of very low significance were identified by the licensee and were violations of NRC requirements which meet the criteria of Section VI.A.1 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

NCV Tracking Number	Requirement Licensee Failed to Meet
275/2002003-04	10 CFR Part 50, Appendix B, Criterion V, states in part, that activities affecting quality shall be accomplished in accordance with documented procedures. Procedure MA1.ID14, “Plant Crane Operating Restrictions,” Revision 8, partially implemented this

requirement and stated in Section 4.2.2.d that "no part of any load weighing more than 20,000 lbs may be moved over a restricted area." Contrary to this requirement, a procedure affecting quality was not performed in accordance with a documented procedure. Specifically, on April 30, 2002, the 70 ton (140,000 lbs) Low Pressure Turbine Hood C was lifted over the restricted area, above the three Unit 1 diesel engine generator rooms and a cable run for ASW Pump 1-2. This event is described in the licensee's corrective action program, reference AR A0554228. This is being treated as an NCV.

This event is more than minor because it had a credible impact on safety. If the heavy load was dropped, it could have rendered all three diesel generators and an ASW pump unavailable. Using the Significance Determination Process, the inspectors determined that this event had very low safety significance (Green). Unit 1 was in Mode 5 (Cold Shutdown) during this event. The heavy load passed over the restricted area for a total of 10 minutes, during which the combined probability of a loss of offsite power and heavy drop load was very low.

323/2002003-05

Technical Specification 3.3.3 states, in part, that two hydrogen monitoring channels shall be operable. With two hydrogen monitoring channels inoperable, restore one channel to operable status in 72 hours or be in Mode 3 (hot shutdown) within the next 6 hours. Contrary to the above, Unit 2 Hydrogen Monitoring Channels CEL-82 and -83 were inoperable from May 23-30, 2001, but Unit 2 was not placed in Mode 3. A technician closed the reagent gas isolation valve (required for operation of the hydrogen monitors) on May 23, 2001, and they remained closed until May 30, 2001. This event is described in the licensee's corrective action program, reference AR A0534742. This is being treated as an NCV.

This item was more than minor because it had a credible impact on safety, in that a total loss of function existed for the ability of operators to monitor for the presence of hydrogen in the containment atmosphere. Using the Significance Determination Process, the inspectors determined that this issue was of very low safety significance (Green) because an actual breach of the containment barrier did not exist. In addition, operators could have had adequate opportunity for recovery of a failed channel in that the hydrogen monitor panel had a low reagent gas pressure alarm to alert the operator of a problem with the reagent gas. The Final Safety Analysis Report Update stated that the containment would reach the flammable limit of hydrogen concentration in 5 days, ample time for operators to determine that the reagent gas isolation valve was closed.

275/2002003-06	Technical Specification 3.7.6 requires that the fire water storage tank level shall be greater than or equal to 41.7 percent for two-unit operation. To satisfy seismic concerns, Supplemental Safety Evaluation Report 8 specifies that redundant flow paths be available to bypass any assumed single failure between the fire water storage tank and the auxiliary feedwater pump suction. Contrary to the above, personnel had closed Valve MU-1-297 for greater than the 7-day allowed outage time, which eliminated one of several redundant suction flow paths. Upon discovery, the licensee opened Valve MU-1-297, restoring operability. The licensee placed this deficiency in their corrective action system as AR A0540528. This is being treated as an NCV. The violation had more than minor significance because it had credible impact on safety in that it could have made a required, redundant auxiliary feedwater suction source unavailable during a seismic event. The inspectors concluded that this issue had very low safety significance (Green) since multiple other suction sources were available.
275/2002003-07	10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made, surveys that are reasonable under the circumstances to evaluate the extent of radiation levels. On January 18, 2002, the licensee identified that radiation levels as high as 9 millirem per hour were found outside a posted radiation area near the drum compactor room on the 115-foot elevation of the auxiliary building. This violation is being treated as an NCV and is in the licensee's corrective action program, reference AR A0547888. The safety significance of this violation was determined to be very low (Green) by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

275/2002003-08

Technical Specification 5.7.1.a requires, in part, that areas with radiation levels greater than 100 millirem be guarded or posted as a High Radiation Area. On May 4, 2002, the licensee identified that an area above the reactor coolant system letdown mixed bed demineralizer cubicle was not posted or guarded for about one hour after it was identified. General radiation levels in the area were as high as 250 millirem per hour. This violation is being treated as an NCV and is in the licensee's corrective action program, reference AR A0554586.

The safety significance of this violation was determined to be very low (Green) by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Becker, Station Director
C. Belmont, Manager, Nuclear Quality Services
D. Christensen, Engineer, Nuclear Quality Analysis and Licensing
C. Gillies, Director, Site Services
J. Hays, Director, Maintenance Services
D. Miklush, Director, Engineering Services
P. Nugent, Manager, Regulatory Services
D. Oatley, Vice President, Diablo Canyon Operations
P. Roller, Manager, Operations Services
R. Todaro, Manager, Security Services
J. Tompkins, Director, Nuclear Quality Analysis and Licensing
L. Womack, Vice President, Nuclear Services

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Opened and Closed During this Inspection

275/2002003-01	NCV	Failure to perform adequate postmaintenance test on auxiliary saltwater pump (Section 1R19)
275/2002003-02	NCV	Failure to post a radiation area (Section 2OS1)
275/2002003-03	NCV	Failure to barricade a high radiation area (Section 2OS1)
275/2002003-04	NCV	Heavy load lifted over restricted area above diesel generators (Section 4OA7)
323/2002003-05	NCV	Technical Specification 3.3.3 violation for two inoperable hydrogen monitors (Section 4OA7)
275/2002003-06	NCV	Technical Specification 3.7.6 Not Met When the Fire Water Storage Tank Was Isolated from the Auxiliary Feedwater Pump Suction Because of Personnel Error (Section 4OA7)

275/2002003-07	NCV	Failure to survey (Section 4OA7)
275/2002003-08	NCV	Failure to post or guard a high radiation area (Section 4OA7)

Previous Items Closed

275; 323/2001-001-00	LER	Automatic diesel generator start upon loss of startup power due to 230 kV line arcing in heavy smoke from escaped fire caused by inadequate administrative controls (Section 4OA3.1)
323/2001-003-00	LER	Technical Specification 3.3.3 not met due to inadequate procedure (Section 4OA3.2)
275/2001-003-00:	LER	Technical Specification 3.7.6 Not Met When the Fire Water Storage Tank Was Isolated from the Auxiliary Feedwater Pump Suction Because of Personnel Error (Section 4OA3.3)

LIST OF ACRONYMS USED

ALARA	as low as reasonably achievable
AR	action request
ASME	American Society of Mechanical Engineers
ASW	auxiliary saltwater
CCW	component cooling water
CFR	Code of Federal Regulations
ISI	inservice inspection
kV	kilovolt
LER	Licensee Event Report
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NCV	Noncited Violation
NRC	U.S. Nuclear Regulatory Commission
OE	operability evaluation
PARS	publicly available records system

DOCUMENTS REVIEWED

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
	Diablo Canyon Power Plant Units 1 and 2 Inservice Inspection Program Plan Second Ten-Year Interval	1
EDMS Item #010740012	NQS Engineering Assessment, 1R10 Steam Generator Eddy Current Inspection Program, 10CFR50 Appendix B	3/19/01
EDMS Item #012000017	NQS Engineering Assessment, 2R10 Steam Generator Eddy Current Inspection Program, 10CFR50 Appendix B	7/24/01
EDMS Item #020640001	NQS Assessment Framatome ANP and DCCP Problem Identification and Corrective Action Program Interface / INPO Steam Generator Review Report	5/01/02
	Diablo Canyon Unit 1 Refueling Outage 1R11 May 2002, Steam Generator Tubing DEGRADATION ASSESSMENT	0
	Diablo Canyon Unit 1 Refueling Outage 1R10 October 2000, 1R10 STEAM GENERATOR REPORT	0
PG&E Letter DCL-01-010	Special Report 00-05 - Results of Steam Generator Alternate Repair Criteria for Diablo Canyon Power Plant Unit 1 Tenth Refueling Outage	2/5/01
PG&E Letter DCL-01-023	Special Report 01-01 - Steam Generator Condition Monitoring Unit 1 Cycle 10	3/24/01
	Unit 1 Primary-to-Secondary Leak Rate	4/25/02
1169333A-36	Field Procedure for Remote Rolled Plug Removal by TIG Relaxation	36
1241428A-08	Field Procedure for Steam Generator Closeout	8

Miscellaneous

NUMBER	TITLE	REVISION/ DATE
1280230A-02	Field Procedure/Operating Instructions for Remote Ribbed Plug Removal by Mandel Disengagement, TIG and Pull	2
54-ISI-75-03	Administrative Procedure for the Design, Procurement, Fabrication, Documentation, and Certification of Calibration Standards for ASME Code Eddy Current Examinations	3
FRA-1275284A	Field Procedure for Remote Rolled Plugging Utilizing the LAN SAD Box (Delta Plugging System)	5
FRA-6002121	Operating Instructions for ROGER in Recirculation Steam Generator	1

Procedures

NUMBER	TITLE	REVISION
NDE ET-7	Eddy Current Examination of Steam Generator Tubing	1
STP M-SGT1	Steam Generator Tube Inspection	6
TS1.ID3	Steam Generator Management Program	4
TS1.NE3	Steam Generator Secondary Side Integrity Program	1

Examinations Records List by ISI 1R11 Exam Matrix Item

1, 2, 6, 8-10, 12-19, 21-24, 28, 30, 35-43, 43.5, 46-48, 184,197, 198, 202, 217, 218, 235

Action Requests

A0518589
A0555463
A0555465
A0555752
A0556015
A0556255