APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-275/94-11
50-323/94-11

Operating Licenses: DPR-80
DPR-82

Licensee: Pacific Gas and Electric Company
Nuclear Power Generation, B14A
77 Beale Street, Room 1451
San Francisco, California 94177

Facility Name: Diablo Canyon, Units 1 and 2

Inspection At: Diablo Canyon Site, San Luis Obispo County, California

Inspection Conducted: March 20 through April 23, 1994

Inspectors: M. Miller, Senior Resident Inspector
M. Tschiltz, Resident Inspector

Approved By: D. Kirsch, Chief
Reactor Projects Branch E

5/19/94
Date Signed

Inspection Summary

Areas Inspected (Units 1 and 2): Routine, announced, resident inspection of onsite followup of events, operational safety verification, plant maintenance, surveillance observations, refueling preparations and operations, quality oversite activities, safety system walkdown, followup on corrective actions for violations, followup, and in-office review of licensee event reports.

Results (Units 1 and 2):

Operations:

Operations personnel performed well during this inspection period.

Strengths:

- Prompt conservative actions were implemented by the operations department to shut down the Unit 2 reactor within 6 hours when a leak was identified in the reactor coolant system (RCS).

Weakness:

- Communications and control of fuel movement were inconsistent with management's expectations.

Maintenance:

Certain maintenance activities were not conducted in accordance with applicable procedures and management expectations during this inspection period.

Weaknesses

- Actions to repair the leak in the Unit 2 reactor were not conducted using the required clearance for control of the repair work boundary. As a result, argon gas was injected into the RCS, which could have resulted in elevated RCS activity and radiation levels.

- Maintenance management had not clearly communicated expectations for signing work orders to the construction crew installing an inverter, resulting in recorded dates for installation work steps being back-dated to the date of performance, rather than indicating the date of signature. Licensee management took prompt action to communicate expectations to construction personnel and, later, to maintenance personnel.

Engineering:

The licensee's engineering organization responded well to the emergency diesel generator air flow concern; however, weakness was observed in the conduct of local leak rate testing (LLRT) and the evaluation of reduced diesel generator loading capability.

Strengths

- The engineering organizations promptly responded to concerns regarding EDG radiator airflow, contacted industry experts, conducted intensive testing of diesel generator radiator air flow, took conservative action to recommend delay of Unit 2 restart, and maintained Unit 2 in cold shutdown while EDG operability testing was ongoing. This evidenced a conservative safety perspective.

Weaknesses

- The operations and maintenance organizations identified inadequate administrative controls of LLRT temporary modifications by engineering. Plant management took actions to clarify expectations and implement additional training for LLRT personnel.

- The initial operability evaluation associated with Diesel Generator 1-3 declared the diesel generator operable for Modes 5 and 6 at reduced electrical loading without consideration of Technical Specification (TS) requirements for diesel generator loading during these modes and without adequate instruction to operators regarding selection of electrical loads which would be appropriate to shed in the event of a derated generator. Licensee management agreed with inspectors that this evaluation was not appropriate, and the evaluation was revised.

Plant Support:

The licensee's quality oversight organization performed well during this inspection period.

Strengths

- The quality organization evidenced strong, intrusive involvement, problem identification, and corrective action concerning weaknesses associated with the maintenance area. Issues included operability evaluations of past improper re-installation of motor-operated valve pinion keys, inspections to identify cracking in 480 V transformer insulators, improper temporary attachments in the Unit 1 containment, and lack of visual inspection of snubbers.

Summary of Inspection Findings:

- Violation 323/94-11-01 was identified (Section 4).
- Violation 275/93-32-01 was closed (Section 10).
- Licensee Event Reports 275/94-04, Revision 0, and 275/94-05, Revision 0, were closed (Section 11).

Attachments:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Acronyms

DETAILS

1 PLANT STATUS (71707)

1.1 Unit 1

The unit was shut down throughout the entire inspection period for a refueling outage (1R6). Core offload and reload operations occurred during this period.

1.2 Unit 2

At the beginning of this inspection period, the unit was operating at 100 percent of rated thermal power. On March 26, 1994, power was reduced to 50 percent to facilitate scraping of marine growth from the circulating water Pump 2-1 tunnel. On March 27, 1994, the unit was shut down after identification of RCS leakage. Following repair of the leak, and EDG radiator air flow testing, the unit transitioned to Mode 1 and returned to 100 percent power on April 10, 1994.

2 OPERATIONAL EVENTS (93702)

2.1 Unit 2 Unusual Event Due to Reactor Coolant System (RCS) Pressure Boundary Leakage

During a containment entry for the purpose of troubleshooting Reactor Coolant Pump 2-3 seal leakoff indication, a leak was identified near Reactor Coolant Pump 2-3, in an area containing piping from several systems normally inaccessible during reactor operation. The location of the leak was initially thought to be valve packing leakage from the resistance temperature detector (RTD) manifold isolation valve. This was difficult to confirm due to the high radiation levels in the area. Operations management concluded that a plant shutdown was prudent and reached Mode 3 within 6 hours. Immediately after reactor shutdown, on a containment entry for further investigation, the leak location was determined to be not from the RTD manifold, but from a nonisolable cracked socket weld on a 3/4-inch vent line connected to Safety Injection Accumulator 2-3 injection line. Operations concluded that a plant shutdown would have been required by TS since the 0.3 gpm leak was nonisolable pressure boundary leakage. An unusual event was declared. Upon transition to Mode 5 on March 28 1994, the licensee terminated the unusual event and began establishing conditions for repair of the leak. Failure analysis of the cracked weld, performed by Westinghouse, revealed that the cause of the leak was a weld defect on the inner diameter of the weld, an area not required to be inspected during fabrication.

2.1.1 Conclusion

The licensee operations staff initiated a conservative reactor plant shutdown following discovery of the RCS leak. Immediately after reactor shutdown, a more precise leak location was identified. The licensee Operations staff

promptly declared an Unusual Event upon determination that the leak was unisolable RCS leakage. Evaluation of repair efforts are described in Paragraph 4.

2.2 Unanticipated High Radiation Area During Chemical and Volume Control System (CVCS) System Fill and Vent

At about 4 p.m. on April 11, 1994, the licensee identified that a high radiation area had occurred in a radiologically controlled area hallway, originating in reactor cleanup piping mounted along the hallway. The area was posted promptly upon discovery, a root cause evaluation initiated, and a visiting NRC health physics inspector and resident inspectors were informed. Based on the configuration of the radiation area localized to a hot spot, and the low traffic in the area, the licensee determined that no personnel had been over exposed.

Earlier that day, at 2:50 a.m. on April 11, before CVCS fill and vent operations had commenced, Operations alerted Health Physics to the potential for changing radiation areas caused by the fill and vent. Apparently no radiation protection surveys were accomplished in that area after that announcement until the routine survey around 4 p.m. the same day, which identified the hot spot.

Licensee root cause investigation determined that the area radiation levels had risen sharply, at approximately the time the CVCS fill and vent procedure had been underway, while testing the CVCS diversion valve function. The licensee determined the most likely cause of the hot spot, which is discussed in the following paragraph.

Forced oxygenation, which was performed before shutdown to decrease RCS piping radiations levels, probably caused highly activated iron oxides (magnetite) to deposit in the deborating demineralizer. A newly replaced 0.2 micron filter upstream of the hot spot and downstream of the deborating demineralizer was later found to have failed. It is likely that a water slug was generated by the fill and vent procedure, which could have caused the magnetite to break loose from the resin bed and travel to the filter. The filter may have ruptured at that time or an earlier time. This would have allowed the magnetite to travel past the filter to the area of piping which was the source of the hot spot. The filter vendor and the licensee concluded that the filter, bowed out as if overpressurized, had been exposed to reverse flow, since the normal flow path is from the outside of the cylindrical filter to the inside of the filter. A reverse flow path may have occurred during hydrostatic testing of the volume control tank. Check valve leakage was suspected during the hydrostatic test. Licensee investigation is continuing and appears to be conducted in a thorough manner.

At the time that the hot spot was identified, no formal notification was made to the NRC. The licensee's emergency plan was not consistent with the safety significance of the event, since the only emergency plan guidance available was ambiguous and would have resulted in a highly conservative recommendation

of an Alert level of emergency response. The licensee determined that declaration of an Alert was incorrect, since no chance of off site radiation release was possible for this hot spot. After inquiries from the NRC, the licensee later discussed the basis for not declaring an Alert with NRC management. The basis appeared appropriate. The licensee agreed to revise the emergency response plan to more clearly and appropriately address these types of situations. The licensee's corrective action in this area will be reviewed during the next scheduled NRC inspection of the emergency response area.

The health physics aspects of this issue are addressed by the NRC inspection report issued by the Region-based NRC health physics inspector. The licensee received a violation for the unposted radiation area, issued in the NRC health physics Inspection Report 50-275/94-12; 50-323/94-12.

2.2.1 Conclusion

The licensee radiation protection organization did not appropriately survey for changes in radiation levels when warned by the operations staff. The root cause of this deficiency will be followed by the NRC health physics and emergency preparedness inspection efforts. Licensee investigation into the cause of the hot spot was prompt and appears to be continuing in a responsible, appropriate fashion. Corrective action observed to date appeared appropriate.

3 OPERATIONAL SAFETY VERIFICATION (71707)

3.1 Control of Plant Configuration and Status of Equipment Implemented in the Control Room

Inspectors performed frequent control board walkdowns in the control room to observe Operations control of plant configuration, clarity of clearance tags, appropriate availability of plant equipment and instrumentation during shutdown and operating modes, and effective, conservative transitions through the various modes of operation. Both units transitioned through several operating modes during this inspection period. Inspectors found operators to be knowledgeable and control of plant equipment to be appropriate. Equipment tags clearly referenced applicable clearances and noted additional cautions and restrictions. The TS and the outage safety plan was followed for both units with respect to availability of plant equipment.

3.1.1 Conclusion

The control room boards were maintained in an appropriate fashion during both unit outages. Control board indication and control for plant equipment and instrumentation were appropriately tagged, and restrictions were clearly documented or referenced for cases where multiple safety controls were applicable.

3.2 Inappropriate Documentation of Containment LLRT Jumper (Jumper 94-019)

On March 16, during routine Operations department review of the jumper logs, operations identified that a jumper had been documented but had not received proper reviews by operations. Operations issued Action Request A0331842 to document the deficiency. Later review determined that the jumper supplied pressure for containment isolation valve local leak rate testing in containment, being performed by engineering and maintenance. The pressurizing gas was supplied through the spare Containment Penetration 80.

Further investigation revealed that poor communications had occurred between Engineering and Maintenance concerning jumper documentation and existing procedural controls for the LLRT, resulting in submission of incomplete jumper documentation. Quality Evaluation 11270 was issued to determine the root cause of the inadequate communications.

Engineering and maintenance management discussed appropriate expectations and controls for planning and documentation of LLRT jumpers, as well as conduct of LLRT evolutions, with their staffs.

To determine the safety significance, the inspector evaluated existing controls of the LLRT process regarding the jumper. Further review determined that the entire process of connection, control, disconnection, and use of the jumper was controlled by several plant procedures, including:
Procedure STP G-12, "Operation of the Portable Leak Test Monitor," Revision 3; several procedures for individual penetration testing, such as Procedure STP V-651A, Revision 3, "Penetration 51A Containment Isolation Valve Leak Testing," which, for example, included Step 11.3.15 "replace seal on SI-1-153," and Step 11.3.18 "verify removal of all test instrumentation," with sign offs on each step.

The inspector questioned whether the controls for ensuring containment integrity were established for the spare penetration, and noted that Procedure OP K-10B, "Sealed Valve Checklist for Manual Containment Isolation Valves," Revision 3, Step 2.1, stated "This sealed valve checklist verifies inside containment manual isolation valves are in the correct positions and sealed ... for Modes 1,2,3 and 4." Step 6.1 stated "Visually verify position of containment isolation valves in Appendix 9.1," which included the Penetration 80 spare instrument lines. These controls appeared comprehensive and appropriate.

The safety significance of the inadequate jumper documentation is very low in that multiple levels of control of the jumper exist in plant procedures.

3.2.1 Conclusion

Operations promptly identified incorrect documentation of a plant jumper. Engineering and maintenance management took steps to correct inadequate communications between the two groups coordinating LLRT work. The safety

significance of this issue is low, but provides an example of a communication and work control problem identification and corrective action.

3.3 EDG Radiator Air Flow

Licensee testing of EDG radiator air flow identified lower than expected flow values. The licensee initiated several tests of EDG air flow, including testing of Unit 2 EDGs, during the unscheduled Unit 2 shutdown. The engineering staff conservatively required Unit 2 to remain shut down until EDG operability was thoroughly addressed. An operability evaluation was issued identifying that, above the 78°F ambient design temperature, air flow may not be adequate to remove heat from the radiator. The Final Safety Analysis Report design basis concluded that temperatures above 78°F would occur only for 9 hours per year, with a maximum temperature of 91°F. Temperatures at the site had not exceeded 78°F for an extended period.

On April 1, 1994, during a review of EDG 1-3 operability evaluation, documented on Action Request A0333816, the inspector noted that the licensee concluded that the EDG was operable, with reduced radiator air flow during Modes 5 and 6, provided generator loading was maintained less than 1400 kw. The operability evaluation further stated that generator loading could be increased to above 1400 kw provided operations monitored jacket water temperature every 30 minutes and loads were reduced as required to maintain diesel engine jacket water temperature less than 177°F. This operability evaluation had been reviewed and concurred in by the Manager of Nuclear Engineering Services. The inspector identified that TS 3.8.1.2, Surveillance 4.8.1.2, requires that, during Modes 5 and 6, a diesel generator be capable of being loaded to greater than or equal to 2484 kw. The inspector questioned the validity of the operability assessment based on the TS electrical loading requirements.

The inspector also questioned whether any instructions had been provided to operations department personnel expected to initiate compensatory actions for generator load conditions of greater than 1400 kw. Apparently, no guidance had been provided to operators even though vital component loads on the associated bus had been estimated to be close to 1800 kw during Modes 5 and 6. Guidance for load shedding was provided to operators within 3 hours of the inspector raising the issue.

The operability evaluation was issued and questioned by inspectors when the Unit 1 core was off-loaded (No Mode). Although the operability evaluation documented it's applicability for Modes 5 and 6, the TS requirements were not applicable at that time.

As a result of the concerns raised by the inspector, the licensee revised the operability evaluation.

Licensee calculations completed during the first week of April 1994, concluded that, with roll up doors opened to provide air flow, diesel performance would

meet design basis loading requirements up to the maximum temperature of 91°F identified in the Final Safety Analysis Report.

During a meeting on April 11, 1994, to discuss more detailed aspects of the EDG air flow test results with the licensee, NRC inspectors raised the following issues:

- The need for issuing formal guidance in procedures to the operators to determine which loads should be shed.

- Apparently, the EDG cooling air supply had not been tested or calculated for conditions when more than one diesel is running. This may not be conservative, since three diesels may start and all use the same cooling air source when operators open doors for extra cooling.

- The calculated basis for EDG operability during periods when outside ambient temperature was greater than 85°F relied on EDG operation with engine jacket water temperatures at 205°F, which is greater than that recommended by the EDG vendor (190°F). The inspector questioned whether this condition was considered acceptable by the vendor.

- Operability of EDGs depends on operator actions to open roll-up doors upon alarm of the diesel jacket water high temperature. However, the alarm circuit was not safety related and does not undergo routine surveillance to ensure it will perform it's required function. Similarly, the roll-up doors were not qualified for design basis functions. Various design basis scenarios for the doors and alarm circuit had not been evaluated regarding seismic, fire protection, equipment qualification, and control room evacuation requirements.

- The capability for alarm circuit reflash was not known, i.e., whether or not a separate alarm on that circuit would preclude annunciation of a jacket water temperature (or other temperature) alarm.

- No acceptance criteria appeared to be stated for the cleanliness of the air pathways in the EDG radiators, although debris in the radiators may ffect air flow through the radiator.

- Several roll up doors were involved in supplying air to radiators; however, the doors required to be opened were not identified clearly and specifically.

- It was not clear that appropriate conservatism was used in the calculations for the radiator performance, such as dry air, wind effects, or other heat transfer conservatism.

- It was not clear that the vendor had concurred in the air flow and diesel performance calculations.

- The licensee had not shared air flow measurement and radiator performance information with other licensees with similar diesel radiator designs.

- Past surveillance test performance of diesel generators while air temperatures were above 78°F had not been reviewed to validate operability calculations.

The licensee provided satisfactory resolution of each of the above issues, obtained vendor concurrence with performance conditions and expectations, or is performing detailed calculations and evaluations to address the concern. Testing was later performed with all three Unit 1 EDGs running, with satisfactory air flow results. Other licensee identified technical issues are under review or undergoing calculations. The information has been shared with the industry. The licensee has initiated a hot weather plan, to be effective upon alarm indication in the control room for ambient temperature greater than 78°F.

A search of past operations during high ambient temperatures indicated that, during a 7-hour run at 85°F ambient temperatures, jacket water temperature reached and stabilized at 200°F, which the vendor concurred was acceptable for sustained operation. This satisfactory EDG performance occurred without opening roll up doors.

3.3.1 Conclusion

The licensee appeared to have promptly addressed the operability issues. However, inspectors were able to identify several potential issues, which indicated some weaknesses in the completeness of engineering and technical work.

Overall, licensee engineering involvement appeared to be acceptable.

3.4 Flow Oscillations in EDG Radiators

The licensee identified that air flow through the EDG radiators evidenced pressure oscillations. The fans were designed for a pressure drop of 1.8 inches of water and were installed in a configuration which resulted in approximately 2.4 inches of water. The licensee tested fan blades at lower angles to determine if oscillations could be reduced, but minimal benefit was gained, and air flow was reduced. The licensee returned the blades to the original configuration.

Detailed inspection of the blade hubs was performed, and no indications of fatigue or growth of existing surface discontinuities was identified. Despite the lack of fatigue indications, the licensee implemented a previously approved design change during the unscheduled Unit 2 outage, replacing Unit 2 EDGs fan hubs with more robust design hubs. This design change had already been performed on Unit 1 during the scheduled outage. The licensee concluded,

and obtained vendor concurrence, that continued operation of the EDGs with flow oscillations would not be detrimental to long term EDG operation.

3.4.1 Conclusion

The licensee appeared to have appropriately addressed these issues with timely, intrusive engineering involvement.

3.5 Reactor Vessel Level Indication System

On April 20, 1994, during preparations to perform RCS maintenance at reduced inventory while at an RCS level of 109 feet (RCS loops full, and 2 feet above the midloop level of 107 feet), the licensee found that the narrow range level indicator, LT 400, did not properly agree with the other narrow range transmitter or the wide range transmitter. Operations staff halted the procedure for continuing to midloop operations while the RCS was at 109 feet, until the level indication was repaired. Licensee management of maintenance, engineering, and operations became actively involved, along with plant staff, in agreeing that operations would remain halted until the problem was resolved.

Troubleshooting revealed that the indicator consistently indicated about a 1.5-inch lag in actual level, resulting in the indicated level being approximately 1.5 inches high under conditions of decreasing level, a nonconservative indication. The level indicator consisted of a reference leg in which a float containing a magnet rose and fell with reference leg level. The magnet actuated magnetic sensors outside the reference leg, allowing visual indication of the level. The licensee duplicated the observed offset in a bench test by changing the axial orientation of the float, resulting in the magnet not facing the sensors and changing the sensed magnetic field. Efforts to realign the float in the plant were not successful. The lag offset was consistently repeated during troubleshooting in the plant, and, after contact with the vendor, plant Engineering, Operations, and Instrument and Control staff concluded that the indicator's calculated instrument error should be increased to include the observed error and procedures revised accordingly before entering midloop operations. After revision of the procedures to raise the minimum level of the RCS level midloop operating band, which resulted in narrowing the operating band, operations at reduced inventory were initiated. No further problems were identified during reduced inventory operations.

3.5.1 Conclusion

Major work activities were halted when plant management concluded that further investigation and testing of the level indicated was required. This was a conservative and responsible approach to plant safety during reduced inventory.

4 PLANT MAINTENANCE (62703)

During the inspection period, the inspector observed and reviewed selected documentation associated with maintenance and problem investigation activities listed below to verify compliance with regulatory requirements, compliance with administrative and maintenance procedures, required quality assurance/quality control department involvement, proper use of safety tags, proper equipment alignment and use of jumpers, personnel qualifications, and proper retesting.

Specifically, the inspector witnessed portions of the following maintenance activities:

Unit 1

- Diesel Engine Generator Inspection (18-Month interval)
- Inspection/Replacement of 480 Volt Bus G Insulators
- IY-13 Cable Installation and Termination

Unit 2

- Replacement of Safety Injection Vent Line
- FCV-439 Motor Operator Inspection
- EDG 2-1 Postmaintenance Test (PMT 27.21)

4.1 Inverter IY-13 Installation

On April 4, 1994, during review of a work order which included connection of wiring associated with Inverter IY-13, the inspector noted that the foreman verification of the clearance had not been signed for more than 1 week after the start of the work, and subsequent work order steps had been performed. The inspector questioned the foreman in charge of the work, who indicated the signature should have been made prior to starting the work. Subsequent review of the work package revealed that the foreman back-dated his signature for having verified the clearance to March 26, 1994, although the work order step was signed on April 4, 1994. Discussions with licensee management revealed that this situation did not meet their expectations. The management, however, indicated that the foreman stated that he had verified the clearance prior to starting the work and had not signed the work package at that time.

The inspector also inquired of the foreman when the work was scheduled to be completed. The foreman indicated that the work should be completed within the next several hours. While reviewing the work package, the inspector noted that a significant number of steps had not been signed as complete. The inspector questioned the foreman concerning the lack of signatures. The

foreman indicated that most of the work had been completed but not signed for and that he was in the process of verifying the work which had been performed by the other shift. The inspector raised the concern to licensee management regarding the lack of discipline of the involved workers in that they were not completing sign-offs for their work as work was completed. Additionally, later review by the inspector found that the dates for some of the recently signed steps had been back-dated to an earlier time.

The inspectors were concerned that construction workers had not understood management expectations that steps would be signed off as soon as they were completed, and dated on the date they were signed, rather than allowing steps to be signed days later and dated back to the date the step was completed.

Licensee management expressed concern that workers had not understood these expectations, particularly in light of past procedural compliance issues. On April 7, 1994, a bulletin was issued to all nuclear construction services personnel outlining the expectations that steps would be signed and dated as soon as practicable upon completion of the step, as well as the potential safety and work control consequences associated with improper completion of sign-offs. The inspectors questioned whether the plant maintenance personnel were also fully aware of these expectations. The licensee stated that the contents of the bulletin would be discussed with plant maintenance personnel as well as construction personnel. These actions adequately addressed the inspector's concerns.

4.1.1 Safety Significance

There is no safety significance specifically associated with the lack of a signature for the clearance verification step, since the same clearance number and clearance points had been used to remove the old inverter, IY-13, and the step in that work order which required the foreman to walk down the clearance had been performed and signed as complete at the start of the work. The construction crew was aware of that the same clearance was being used in the installation of the new inverter.

4.1.2 Conclusion

The licensee had not clearly conveyed expectations associated with timely sign off and correct dating of work order steps. Although the construction work in the field appeared to have been properly completed, the inspector identified a case of lack of sign off of a step and late sign off and back-dating of some steps of a work order which replaced a safety-related inverter. The licensee actions in response to the concern appeared appropriate. The safety significance of these specific findings was negligible, since work appeared to have been completed appropriately and the identical sign off in a separate work order referring to the same clearance had been signed by the foreman.

4.2 Replacement of Safety Injection (SI) Accumulator 2-3 Injection Vent Line

On March 31, 1994, the inspector reviewed operator logs and interviewed the shift supervisor following an unexpected delay in the work schedule for the replacement of SI Accumulator 2-3 vent line. The Shift Supervisor indicated that the delay was caused in part by the concern for hydrogen off-gassing in the RCS and, later, by presence of a vacuum in the vent line which was observed after removal of the vent line pipe cap. The delay was due to concerns regarding the potential for introduction of material into the RCS while the pipe was being cut with the presence of a vacuum in the pipe.

Subsequent discussion with the Directors of the Operations and Mechanical Maintenance Departments revealed that no clearance had been used to perform the leak repair. A clearance had been written by the work planning group for the performance of the work but was not used. The decision to not implement a clearance for performance of the work was made at the prejob briefing by Operations Department personnel. This is a violation of the work procedure. The Operations and Mechanical Maintenance managers said that the decision to not set the clearance was based on concern over disturbing the failed weld for analysis. As a result, residual heat removal (RHR) cold leg injection to Loops 3 and 4 was not isolated from the maintenance area and flow past the safety injection pipe penetration created a vacuum in the vent line.

An additional concern over the need to purge hydrogen from the vent line had been raised at the prejob briefing. Individuals in the prejob briefing decided to remove the hydrogen from the line by establishing an inert gas purge prior to cutting the pipe. The use of a nitrogen purge was then discussed with the Mechanical Maintenance manager and agreed upon. Subsequent to the discussion, personnel involved with the work decided to use argon as the purge gas. Argon was readily available, since it was planned to be used as a cover gas during the welding, and necessary hoses and regulators were already staged at the work site. Management was not consulted following the decision to use argon vice nitrogen as the purge gas. The argon purge equipment was connected and purging was commenced at a rate of 50 cubic feet per hour. No instructions specifying the use, connection, or removal of argon purge equipment were issued. After purging argon for approximately 40 minutes, the welding personnel questioned the continued existence of a negative pressure in the vent line and contacted operations personnel to investigate. Operations personnel then told the welders to secure the purge and investigated the cause of the vacuum in the line. It was determined that Loop 3 RHR flow past the SI pipe connection was creating the negative pressure in the vent line. Valve RHR-2-8809B, which was initially planned to be shut in accordance with the unimplemented clearance, was then shut. This secured RHR flow to Loops 2 and 3 and eliminated the vacuum in the vent line. Work to replace the vent line was then resumed.

During restart operations, removal of argon from the RCS was performed by filling and venting the volume control tank.

4.2.1 Conclusion

Investigation of the work activities revealed that there was confusion over the maintenance activity requirements in several areas which contributed to failure to set the proper clearance for the work and the improper use of argon as a purge gas. In cases where a purge is required, the work order normally specifies the installation and removal of purge equipment. The inspector questioned the lack of a clearance and the adequacy of both the planning and the procedures for performing the work. Personnel involved in the decision to use argon as a purge gas were not aware of the potential radiological consequences due to the high energy activation of argon after it is exposed to a neutron flux.

The failure of the licensee to implement the required clearance, thus allowing argon gas into the RCS, was a violation of TS 6.8.1, which requires that procedures be implemented for the recommended processes of Regulatory Guide 1.33. Regulatory Guide 1.33, paragraph 9.a., recommends that procedures associated with maintenance of equipment important to safety be properly preplanned and controlled (50-323/94-11-01).

5 SURVEILLANCE OBSERVATIONS (61726)

Selected surveillance tests required to be performed by the TS were reviewed on a sampling basis to verify that: (1) the surveillance tests were correctly included on the facility schedule; (2) a technically adequate procedure existed for performance of the surveillance tests; (3) the surveillance tests had been performed at a frequency specified in the Technical Specifications; and (4) test results satisfied acceptance criteria or were properly dispositioned.

Specifically, portions of the following surveillances were observed by the inspector during this inspection period:

Unit 1

- Reactor Cavity Manipulator Crane
- Diesel Generator Radiator Air Flow

Unit 2

- Control Room Ventilation Functional Test

5.1 Surveillance of Reactor Cavity Manipulator Crane

During routine inspection activities, the inspector reviewed documentation of replacement of several relays in the motor circuitry of the refueling manipulator crane (WO C0121933). Although the crane is not safety related, it is controlled by TS, and it's function of lifting and moving reactor fuel is important to safety. The inspector questioned whether the crane had been

returned to service without performance of a formal surveillance test. After completion of relay replacement, the crane vendor and reactor engineers had informally performed parts of an integrated crane performance test, Procedure STP M-27, "Fuel Handling System Interlock Verification and Functional Test," and maintenance personnel had performed required postmaintenance testing of the relay circuity and motor circuits. Reactor engineers and engineering management explained that, since none of the relays associated with the TS required overload and underload circuitry (associated with fuel lift) had been replaced, the operability of the crane did not require formal surveillance testing. Surveillance Test M-27 was performed in a formal manner later, as a conservative measure.

The inspector identified that the formality of the control of crane operability was less than desirable, since the clearance to remove the crane from service for relay replacement had referenced the TS surveillance test, performed several days earlier on the TS underload and overload interlock circuits, as the basis for returning the crane to operable status after the relay replacement. This was not appropriate documentation of equipment operability, and engineering management agreed that proper formal documentation of operability would be discussed with engineering staff.

An additional concern was identified by the inspector. During preparation of some work orders, identification of applicable postmaintenance surveillance test requirements may be inadequate. Planners determine the need for a surveillance test by determining if the equipment is safety related or listed in Equipment Control Guidelines, developed for administrative control of important equipment which is not listed in TS. The inspector pointed out that surveillance requirements for equipment which is not safety related but is included in TS would not be readily identified. The licensee acknowledged this vulnerability, and documented this in an action request, for correction.

5.1.1 Conclusion

The Reactor Engineering evaluation of the TS requirements and the effect of the maintenance on the reactor cavity manipulator crane operability appeared to be technically adequate, although the formality of documentation was not in accordance with management expectations.

6 SAFETY SYSTEM WALKDOWN, 125 V DC SYSTEM (71710)

The inspector reviewed the Final Safety Analysis Report, TS, and design control memorandum (DCM S-67) ensuring there was consistency between the different documents. TS surveillance requirements were reviewed along with the surveillance procedures which accomplished them. The modifications to the 125 V DC system being performed during the Unit 1 outage were discussed with the system engineer concerning the scope of the modifications, alternate power supplies, and the changes to emergency procedures associated with the modifications. One administrative discrepancy was found where the design control memorandum referenced a TS table which did not exist. This discrepancy was pointed out to the licensee. The licensee initiated an action

request to track correction of the deficiency. Inspectors walked down safety-related 125 V DC circuitry in Unit 1 containment and in the auxiliary building to evaluate the condition and locations of selected 125 V DC circuitry. The inspectors performed a visual inspection of the internals of two battery chargers and a 125 V DC distribution panel. No deficiencies were noted.

6.1 Conclusion

The 125 V DC system appeared to be in good order and the system engineer appeared knowledgeable.

7 PREPARATIONS FOR REFUELING (60705)

The inspectors reviewed the licensee's preparations for refueling to determine if adequate safety and procedural control was properly implemented in preparation for refueling. The inspectors examined surveillance test completion, equipment checkouts, fuel handling and inspection operations, shift manning requirements, crew briefings, prerequisite lists, outage safety planning, control of reactivity, quality oversite involvement, and other areas. The inspectors also examined preparations for fuel movement in the containment refueling areas, the fuel handling building, and the control room. The licensee procedures, outage planning, and training addressed control of reactivity, fuel inventory, movement of fuel modules, communications between refueling crew, control room crew, fuel handling building crew and reactor engineering monitors. The engineering and control room staff had been trained and appeared to have the tools to maintain rigorous control of the observed evolutions.

7.1 Conclusion

The preparations for refueling appeared well planned and appropriate.

8 REFUELING OPERATIONS (60710)

8.1 Overview

The inspectors observed refueling operations to evaluate the control of fuel modules and reactivity, refueling equipment operability, effectiveness of procedures and communications, and overall safety of fuel manipulations and reactivity control. All of these areas appeared to be effectively implemented.

8.2 Nonconservative Fuel Handling Practices

In one instance, when a fuel module was being loaded into Core Location F-8 with difficulty, a reactor engineer observed the refueling crew place the difficult module to the side and place a different fuel module in Core Location F-8, to determine if the original module was bowed. The reactor engineer immediately identified that, by procedure, no module may be loaded into a location next to other modules unless it is in that module's final core

location. Additionally, the reactor engineer identified that the control room had not been informed that the new module was to be tried in Core Location F-8, which was a violation of the intent of the procedure to have the control room staff aware at all times of what fuel module moves were to occur. These concerns were promptly documented in Action Request 336402.

It was later discovered that the refueling crew in containment had discussed the use of the next fuel module as a test for a bowed module and determined that placement of the fuel in Core Location F-8 would not violate intent of the refueling procedure, since the module would not be loaded, since it would not be unlatched. The refueling crew was aware that the reactivity control of this temporary module configuration was within TS and bounded by core analysis, assuming the existing boron concentration of the refueling cavity.

The inspector discussed the fuel movement issue with reactor engineering management, who considered that the refueling crew should have informed the control room of the intended movements of fuel, regardless of the intent to not unlatch the module in Core Location F-8. Also, licensee management considered that a dummy module should have been used rather than a new fuel module to test for bowed fuel to minimize the chance of damage to fuel modules.

The movement of the new fuel to Control Location F-8 determined that the module was not bowed. Core alterations were halted, and closer inspection found a 1/4-inch fastener on the lower core plate which had interfered with fuel alignment. The fastener was retrieved, and inspection for additional foreign material was conducted. No additional material was identified. The fastener was activated, indicating that it had been irradiated for at least one cycle. The licensee concluded that no fastener of this design was used in any safety-related application, and it had not detached from reactor coolant system internals. The identification of the source of the fastener is continuing.

8.2.1 *Conclusion*

The safety significance of the placement of fuel in Core Location F-8 was minimal, since the reactivity analysis was bounded, and the refueling crew was fully aware of the negligible reactivity consequences of this fuel movement. Though licensee management expectations for communications and control of fuel movement and use of dummy modules to minimize the chance of fuel damage were appropriate and were communicated to refueling crews, the crew did not meet these expectations. The resolution of the fastener on the core support plate was being resolved appropriately.

8.3 <u>Conservative Response to Elevated Counts on a Source Range Channel</u>

On two occasions the inspector observed licensee operations personnel suspend refueling operations due to elevated counts on one of the source range nuclear instrumentation channels. Refueling operations in both instances remained secured until the source range channel level returned to its reference level,

in agreement with the other channel. The licensee initiated actions to determine and resolve the cause of the elevated counts in both instances. The signal on the source range circuit appeared to have been low level electrical interference. However, core alterations were not resumed until the circuit was clear.

8.3.1 Conclusion

The response to elevated counts on a source range channel during refueling operations was conducted in an appropriately conservative manner.

9 QUALITY OVERSITE (40500)

The inspector reviewed findings of quality oversite group audits and surveillances to determine if the audits were sufficiently probing and implemented conservative safety expectations. Separate issues are discussed below.

9.1 Snubber Inspections

During an audit of snubber inspections, the licensee quality organization identified that some snubbers had not been visually inspected as required and had been declared operable without conclusion of the visual portion of the inspection, without signing off the step requiring completion of the visual inspection. The corrective action involved completing visual inspection and counseling mechanical maintenance management and staff on proper documentation and completion of safety-related work. No significant safety issues were identified.

9.2 Refueling Crew Activities

A quality oversite surveillance identified that the refuelling crew involved in the improper location of the fuel module in the core location, discussed earlier, had also been involved in the lack of proper foreign material exclusion postings described in an earlier report and in a miscommunication regarding identification of specific fuel module designations during receipt inspection fuel handling. Further review of the root cause of the communication issues was initiated by the Independent Safety Engineering Group to determine if a common thread and corrective action could be identified for this refueling crew.

9.3 Motor-Operated Valve Motor Pinion Key Inspections

An operability assessment was issued to address a 10 CFR Part 21 identified vulnerability in motor-operated valves in which motor pinion keys may not have been properly installed. The Plant Staff Review Committee and the Independent Safety Engineering Group identified that the scope of inspection of valves and the identification of vulnerability of specific valves was not conservatively addressed in the evaluation. Subsequent inspections and discussions with licensee management resulted in additional valves being inspected, and one key

was found operable but improperly installed in a valve inspected as a result of the quality group's involvement.

9.4 Inspection for Cracking of 480 V Transformers

The potential for cracking of insulators in 480 V transformers was identified and inspections were conducted. The Quality Control staff identified that, because of mechanical interferences and minor accumulated dust, the inspections could be conducted without adequate visibility to identify cracking on the top and sides of the insulators. Intrusive involvement by QC resulted in removal of the layer of dust and improving the visibility of the insulators. Further involvement by the NRC regarding crack acceptance criteria, and lack of identification of potential root causes for the cracking, led to Engineering performing a calculation concluding that the insulators were sufficiently supported to perform their safety function under design basis conditions despite the cracks.

9.5 Temporary Attachments in Containment

During an audit of containment work practices, Quality Assurance staff noted that several temporary attachments, such as electrical extension cords, service air, and other temporary service connections not being properly attached to supports and attached to inappropriate supports, such as instrumentation lines and other plant equipment. The Quality Assurance staff took steps to identify improperly routed attachments and informed the work crews and applicable management of the improper practices. This resulted in correction of improper work practices and re-emphasizing management expectations regarding work in containment.

9.6 Conclusion

Each of these issues were of low safety significance. However, the involvement of quality oversite groups was intrusive, timely, and conservative. Corrective action appeared timely and commensurate with safety significance. As a result of involvement by the quality groups, the safety performance of the plant was improved.

10 FOLLOWUP ON CORRECTIVE ACTIONS FOR VIOLATIONS (92901)

10.1 (Closed) Violation 275/93-32-01, Failures to Follow Procedures

The NRC identified four instances of the licensee's failure to follow procedures, which was issued as a citation with NRC Inspection Report 50-275/93-32. In a letter dated February 4, 1994, the licensee acknowledged the violation and stated that corrective action had been completed for the specific violations cited and that corrective action had been initiated for the overall concern of failure to follow procedures. For each of the specific instances of failure to follow procedures, the licensee documented actions which addressed the concern. The licensee stated that these actions would correct the identified violation and help preclude future

violations. The inspector reviewed and verified these actions. The licensee's actions appeared to be appropriate and properly implemented.

11 IN-OFFICE REVIEW OF LICENSEE EVENT REPORTS (90712)

The following licensee event reports were closed based on in-office review:

- 275/94-04, Revision 0 — Main Steam Safety Valves Outside Design Bases Due to Vendor Identified Deficiency
- 275/94-05, Revision 0 — Failure to Control Reactor Vessel Inventory During Draindown Due to Personnel Error

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

G. M. Rueger, Senior Vice President and General Manager, Nuclear Power Generation Business Unit
J. D. Townsend, Vice President and Plant Manager, Diablo Canyon Operations
W. H. Fujimoto, Vice President, Nuclear Technical Services
*R. P. Powers, Manager, Nuclear Quality Services
*J. S. Bard, Director, Mechanical Maintenance
*G. M. Burgess, Director, Systems Engineering
R. N. Curb, Manager, Nuclear Technical Services
*W. G. Crockett, Manager, Technical and Support Services
S. R. Fridley, Director, Operations
*R. D. Glynn III, Supervisor, Quality Assurance
*T. L. Grebel, Supervisor, Regulatory Compliance
*B. W. Giffin, Manager, Maintenance Services
P. B. Grable, Engineer, Mechanical Maintenance
*C. R. Groff, Director, Plant Engineering
*J. A. Hays, Director, Onsite Quality Control
R. W. Hess, Assistant Director, Onsite Nuclear Engineering Services
*J. R. Hinds, Director, Nuclear Safety Engineering
*K. A. Hubbard, Engineer, Regulatory Compliance
M. E. Leppke, Assistant Manager, Technical Services
*D. B. Miklush, Operations Manager, Acting Plant Manager
*J. E. Molden, Director, Instrumentation and Controls
*T. A. Moulia, Assistant to Vice President, Plant Management
P. T. Nugent, Engineer, Regulatory Compliance
D. H. Oatley, Director, Materials Services
*S. R. Ortore, Director, Electrical Maintenance
P. G. Sarafian, Senior Engineer, Nuclear Quality Services
R. A. Savard, Director, Technical Services
*J. A. Shoulders, Director, Onsite Nuclear Engineering Services
D. P. Sisk, Senior Engineer, Regulatory Compliance
*D. A. Taggart, Director, Onsite Quality Assurance
E. R. Willis, Engineer, Mechanical Maintenance

1.2 NRC Personnel

*M. Miller, Senior Resident Inspector
M. Tschiltz, Resident Inspector
J. Winton, NRR Inspector Intern

In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

*Denotes personnel that attended the exit meeting.

2 EXIT MEETING

An exit meeting was conducted on April 22, 1994. During this meeting, the inspectors reviewed the scope and findings of the inspection. The licensee acknowledged the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 2

ACRONYMS

CVCS	Chemical Volume and Control System
EDG	emergency diesel generator
LLRT	Local Leak Rate Test
RCS	Reactor Coolant System
SI	safety injection
TS	Technical Specifications