APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-275/94-18
50-323/94-18

Licenses: DPR-80
DPR-82

Licensee: Pacific Gas and Electric Company
77 Beale Street, Room 1451
P.O. Box 770000
San Francisco, California

Facility Name: Diablo Canyon Nuclear Power Plant, Units 1 and 2

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

Inspection Conducted: June 5 through July 23, 1994

Inspectors: M. Miller, Senior Resident Inspector
M. Tschiltz, Resident Inspector
D. Acker, Senior Project Inspector
G. Johnston, Senior Project Inspector
D. Corporandy, Project Inspector

Approved: Palmer for 8/15/94
D. F. Kirsch, Chief, Project Branch E Date

Inspection Summary

Areas Inspected (Units 1 and 2): Routine, announced resident inspection of onsite response to events, operational safety verification, plant maintenance, surveillance observations, plant support activities, onsite engineering, followup engineering, followup corrective actions for violations, and in-office review of licensee event reports (LERs).

Results (Units 1 and 2):

Operations:

Strength:

- An alert control room operator noted a changing steam generator level during the performance of an instrumentation and control (I&C) surveillance where all affected instruments were not taken out of service prior to commencing testing. Operator attentiveness identified

the improper performance of a procedural step during the calibration of a steam flow channel and initiated action to secure from testing (Section 5.1).

Maintenance:

Weakness:

- The failure to perform a step in a procedure, and failure to properly resolve the resulting procedural ambiguity during the calibration of a steam flow channel, resulted in a test signal being improperly inserted into control circuitry, resulting in an undesired change in steam generator level (Section 5.1). This was a noncited violation.

- Inadequate consideration was given to the potential impact of the change in charcoal sample analysis on charcoal adsorber bed operability and a ventilation damper design change on charcoal adsorber bed in-service hours. The design change involved the replacement and testing of auxiliary building ventilation damper position switches. During the installation of the design change, the auxiliary building ventilation charcoal adsorber bed remained in service. As a result of delays during the modification and associated testing, the Technical Specifications (TS) required number of hours of service between samples was exceeded (Section 5.2). This was a noncited violation.

Engineering:

Strength:

- During two separate activities, the Quality Assurance organization identified deficiencies in a procedure which prevented four residual heat removal (RHR) check valves from being full-stroke exercised during the last Unit 2 cold shutdown (Section 3.2) and identified nonconservative throttling of component cooling water (CCW) flow to centrifugal charging (CC) pump coolers (Section 6.1).

Weakness:

- ASME Section XI inservice testing requirements were not properly implemented in a surveillance test procedure, which resulted in failure to accomplish required full-stroke testing of four RHR system check valves during the last Unit 2 cold shutdown period (Section 3.2). This was a Level IV violation.

- Nonconservative errors in an engineering calculation resulted in unacceptably low CCW flow to the CC pump coolers. This low flow rate was caused by improper positioning of cooling water throttle valves. The reduced CCW flow would have resulted in exceeding the maximum allowable CC pump bearing oil temperatures during design basis accident

conditions. Concern over the throttling of CCW flow had been raised by the licensee Quality Assurance organization in 1990; however, the issue was inadequately resolved at that time (Section 6.1).

Plant Support:

Strength:

Overall, plant support performance was good during the inspection period and remained unchanged from the last period. Inspectors observed that housekeeping practices in contaminated areas were generally adequate, and could be improved.

Summary of Inspection Findings:

- Violation 323/94-18-01 was identified (Section 3.2).
- Noncited Violation 323/94-18-02 was identified (Section 5.1).
- Noncited Violation 323/94-18-03 was identified (Section 5.2).
- Inspection Followup Item 323/93-30-01 was closed (Section 8.1).
- Violation 323/94-11-01 was closed (Section 9.1).
- LERs 275/94-14, 275/94-10, and 275/93-12, Revisions 0, 1, and 2 were closed (Section 10).

Attachments:

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

DETAILS

1 PLANT STATUS (71707)

1.1 Unit 1

Unit 1 operated at 100 percent power during the entire report period.

1.2 Unit 2

Unit 2 operated at 100 percent power for the entire report period, except on July 9 when power was curtailed to 90 percent for the performance of turbine valve testing.

2 ONSITE RESPONSE TO EVENTS (92701 and 93702)

2.1 Brush Fire Outside of the Protected Area

On June 22, 1994, the licensee declared an Unusual Event (UE) at 2:40 a.m. (PDT) due to a grass fire approximately 100 yards outside of the protected area. The fire was located on the hillside east of the plant and came within approximately 100 yards of the 500 KV transmission lines. The fire burned several acres of grassland to the south of the 500 KV transmission lines and to the northeast of the Nuclear Power Generation warehouse. The fire was caused by an electrical arc at connections on a 12 KV power line, independent of plant related loads. The operations department isolated the power to the line in the process of fighting the fire. Permanent removal of power to the 12 KV line was planned as a corrective action for a previous grass fire but had not been completed. The licensee notified the California Department of Forestry who responded to fight the fire with the licensee's fire response team. The fire was reported to be out and the unusual event terminated at 5:15 a.m.

Conclusion The fire did not at any time pose a significant threat to the safe operation of the facility. Licensee response and declaration of a UE appeared appropriate and well coordinated. However, the licensee's corrective actions for a previous similar event had not been aggressive enough to preclude recurrence.

2.2 Potential to Overpressurize the RHR System

Background During the Unit 1 outage on May 1, 1994, while RHR flow was throttled at the pump discharge due to low core decay heat loads, the licensee inadvertently pressurized the RHR system to 605 psig, while the reactor coolant system (RCS) was solid. The licensee determined that no design margins had been exceeded by that event. However, since the potential to reach RHR system pressure over 600 psig had not been anticipated, the licensee initiated further analysis of the vulnerabilities of the RHR system during throttled flow operations.

As a result of further analysis, the licensee determined that, within the operational controls of the RHR and RCS, it would have been possible to have pressurized the RHR heat exchanger to 675 psig a pressure greater than the design pressure of 600 psig and greater than the ASME code allowable of 110 percent of design pressure (660 psig). The concern is isolated to the RHR heat exchanger, since the piping, instrument tubing, and components such as valves all have design pressures or ASME code allowables above the 675 psig limits.

The licensee determined that the RHR heat exchanger was never subjected to pressure above that allowed by ASME code. Also, the licensee determined that the vulnerability to overpressurize the RHR system was brought about by a failure to properly consider the combined effects of the pump discharge head and suction pressure, while discharge flow was throttled and while the RCS was above atmospheric pressure.

The licensee has initiated a nonconformance report and plans to correct the vulnerability by operationally restricting the use of throttled RHR flow to those cases in Mode 6 where the RCS is vented and, therefore, not able to transfer excessive pressure to the RHR system.

Conclusion The licensee had not properly understood the potential effects of RHR pump discharge pressure under throttled RHR flow conditions with the RCS not vented. There were no negative effects on plant equipment, and corrective action appeared appropriate. Because the worst-case effects of this event were within ASME code allowable limits, this issue was considered of minor importance.

3 OPERATIONAL SAFETY VERIFICATION (71707)

3.1 Failure to Remove Caution Tag

On June 27, 1994, during a walkdown of the Unit 1 pipe rack area, the inspector noted a caution tag which was hung for a surveillance in April 1994 during the Unit 1 refueling outage. The inspector questioned the purpose of the caution tag. The caution tag was attached to Valve Air-I-1-4351 for the performance of Surveillance Test Procedure (STP) I-4-PCV-20, "10% Steam Dump Valve PCV-20 Calibration," which was completed on April 22, 1994. STP I-4-PCV-20, Step 8.5.1.c contains, instructions for hanging the caution tag, and Step 8.5.3.c directs the technician to close vent Valve AIR-I-1-4351, restore the vent valve test cap, and remove the caution tag. The step had been initialed as being complete and verified and initialed by a separate individual. Following identification of the inspector's concern, the vent valve, AIR-I-1-4351, was verified to be open with the test cap restored. Investigation revealed that there was an additional clearance hung on valves within the boundaries of this procedure at the time the surveillance was performed, and that the additional tags were a potential source of confusion during the system restoration portion of the surveillance. The licensee has initiated an Action Request to document this problem, which appeared appropriate.

Conclusion The failure to remove the tag appeared to have been an administrative error with no safety significance in this instance, since plant configuration control appears to have been maintained. However, this does not lessen the significance of two people improperly initialing the completion and verification of the procedural step without completing all of the required actions. The NRC will review the licensee's response to the Action Request and the resolution of this problem.

3.2 RHR Check Valve Inservice Testing

Background During a licensee quality organization review of STP V-4B, "Functional Test of the ECCS Check Valves at Cold Shutdown," it was noted that the recorded data did not verify the required 2200 gpm flow through RHR heat exchanger discharge check Valves 2-8742A and 2-8742B. During the surveillance, a bypass flow path allowed an unmeasured amount of the flow to be diverted around the Valves 2-8742A and 2-8742B. Further investigation of the testing requirements revealed the specified flow rate of 2200 gpm was less than the system design flow rate and, therefore, would not accomplish the required full-stroke testing of the valves. This procedural deficiency affected both RHR heat exchanger discharge check Valves 2-8742A and 2-8742B and the RHR pump discharge check Valves 2-8730A and 2-8730B. Following the discovery of the inadequate testing on June 23, 1994, the licensee entered into TS 4.0.3 for Unit 2. In this situation, T.S. 4.0.3 allowed the action requirements of TS 3.0.3 for both RHR trains inoperable to be extended for up to 24 hours.

Full-stroke testing through the four aforementioned RHR check valves is required by TS 4.0.5.a. TS 4.0.5.a. specifies the inservice surveillance requirements of ASME Code Class 1, 2, and 3 components shall be in accordance with Section XI of the ASME Boiler and Pressure Vessel Code requirements.

Unit 1 valves were not affected since all cold shutdown periods were within 3 months of refueling outage full-stroke exercising. Section XI of the ASME Code Subsection IWV-3522 requires full-stroke check valve exercising where the interval since previous shutdown testing has been 3 months or greater. During refueling outages, the full-stroke testing of the check valves is accomplished by STP V-4A, "Functional Test of RHR Check Valves." STP V-4A accomplishes the full-stroke tests by passing full RHR design flow through RHR heat exchanger discharge check Valves 8742A and 8742B and RHR pump discharge check Valves 8730A and 8730B.

Effect of Condition on Safety Function The licensee requested NRC enforcement discretion be exercised for the performance of cold shutdown full-stoke tests of Unit 2 RHR pump discharge check valves and Unit 2 RHR heat exchanger check valves until the next Unit 2 cold shutdown, and no later than the next refueling outage (2R6). The licensee evaluated RHR system response during the cold shutdown period and previous test results to provide the NRC with justification for continued operation. The NRC exercised enforcement discretion, to not enforce compliance with TS 4.0.3 until a temporary relief request was processed by the NRC, verbally at 6:35 p.m. EDT on June 24, 1994,

followed by letter on June 28, 1994 (Notice of Enforcement Discretion 94-6-011). The temporary relief from ASME Section XI cold shutdown full-stroke requirements for the four Unit 2 RHR check valves was approved by the NRC letter dated July 11, 1994.

Conclusion Failure to properly implement inservice testing requirements in STP V-4B resulted in failure to full-stroke test Unit 2 RHR heat exchanger discharge check Valves 2-8742A and 2-8742B and RHR pump discharge check Valves 2-8730A and 2-8730B during the most recent cold shutdown period. Failure to full-stroke these RHR check valves during cold shutdown periods was a violation of TS 4.0.5.a, which requires inservice testing to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code for ASME Code Class 1, 2 and 3 pumps and valves (323/94-18-01). This is a Severity Level IV violation.

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 and 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 Generator 1-1 Fuel Oil Level Control Valve (LCV-88) Maintenance
- Eagle 21 Loop Processor Board Replacement
- Diesel Generator 1-1 Overcrank Alarm Troubleshooting

Unit 2

- Spent Fuel Pool Swing Gate Seal Replacement
- Spent Fuel Pool Pump 2-2 Maintenance

Conclusion The inspected maintenance activities appeared to have been performed properly. Administrative and maintenance procedures appeared adequate and were followed. There was appropriate quality assurance/quality control department involvement. Technician knowledge and understanding of the activities appeared appropriate during discussions involving the various activities. Radiation protection practices appeared appropriate.

5 SURVEILLANCE OBSERVATIONS (61726)

Selected surveillance tests required to be performed by the Technical Specifications 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 TS; 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

- TP TB-9423; Centrifugal Charging Pump 1-2 CCW Flow Measurements

Unit 2

- STP M-4; Routine Surveillance Test of the Auxiliary Building Safeguards Air Filtration System
- STP I-12B; Channel Calibration Steam Generator Feed Flow, Steam Flow and Steam Pressure Channels

5.1 STP I-12B; Channel Calibration Steam Generator Feed Flow, Steam Flow and Steam Pressure Channels

On July 13, 1994, during surveillance testing which calibrates steam generator pressure and steam flow analog channels, and associated circuitry, two steam flow channels were not removed from service prior to inserting simulated inputs to obtain "as-found" readings. As a result, Steam Generator 2-2 experienced changes in level and feed flow. A control room operator, aware of the ongoing testing, noted the changes in parameters associated with Steam Generator 2-2 testing and initiated action to secure from the testing and restore system parameters.

To accomplish the surveillance testing, several procedures were utilized including:

- STP I-12B1; Removal From Service Steam Generator Feedflow, Steamflow and Pressure Channels
- STP I-12B3; Calibration Analog Electronics Steam Generator Pressure (Flow Compensating)
- STP I-12B4; Calibration Analog Electronics Steam Generator Feedflow

- STP I-12B6; Calibration - Comparators Steam Generator Feedflow, Steamflow and Pressure Channels

STP I-12B1 provides procedural guidance for removal of a steamflow or feedflow channel from service. The technician misunderstood the applicability of the step concerning the steamflow instrument, since the instruments which were being calibrated were steam generator pressure and feedflow. Therefore, the technician omitted removing the steamflow channels from service. When preparing to take the "as-found" readings, the technician questioned the need to take data for the steamflow instrument since this instrument was not being calibrated. The technician stopped the procedure and discussed this with his foreman. The foreman did not see a problem with obtaining the data. It was not communicated that the steamflow instrument had not previously been removed from service. During the performance of the "as found" trip and reset values for a high steam flow comparator, simulated steam flow inputs were inserted. The response of the Digital Feedwater Control System and the resultant change in steam generator parameters was noted by the control room operator, who stopped the surveillance.

Conclusion The procedural step which removes the steam flow instrument from service was read and reviewed, but the incorrect decision was made. The licensee is revising the surveillance procedure requirements for removal of these instruments from service for maintenance and testing. Prompt operator response to changing steam generator parameters prevented an improperly performed surveillance from impacting plant operation. Plant equipment was not negatively affected. The failure to adequately plan and perform the surveillance is a violation of TS 6.8.1, which states, in part, that written procedures shall be established, implemented, and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, dated February 1978. Appendix A of Regulatory Guide 1.33, Revision 2 recommends procedures covering surveillance testing; preventative maintenance; and startup, operation, and shutdown of safety-related systems. Contrary to this requirement, on July 12, 1994, steam flow instruments were not removed form service prior to testing as required by Step 6.2 of STP I-12B1. Since this violation was identified by the licensee, and other criteria of Section VII.B(2) of the Enforcement Policy were satisfied, this violation was not cited (323/94-18-02).

5.2 STP M-4; Routine Surveillance Test of the Auxiliary Building Safeguards Air Filtration System

Background Ventilation system charcoal adsorber beds are required to be periodically sampled to verify that carbon allows only a small percentage of methyl iodide penetration. Previous samples had been analyzed in accordance with ASTM D-3803-79, "Standard Test Method for Nuclear Grade Activated Carbon." NRC Information Notice (IN) 87-32 identified problems with the test methodology in ASTM D-3803-79. ASTM D-3803-89 is recognized by industry as being a more accurate analysis of methyl iodide penetration. The licensee, after performing a review of ASTM D-3803-89, adopted the revised standard for

the analysis of charcoal bed samples. The revised version was considered acceptable since it provided more conservative results than existing ASTM D-3803-79 requirements. The new analysis, performed at a lower temperature, reduces the reaction rate between the charcoal and the iodine and, as a result, increases the organic iodine penetration and provides a more representative indication of charcoal performance during an accident. The licensee adopted this updated method of analysis for the first time for the Unit 1 charcoal samples taken during Refueling Outage 1R6. Results of Unit 1 charcoal analysis indicated a much higher percentage of methyl iodide penetration than the previous samples. The licensee's decision to implement the revised standard test method for nuclear grade activated carbon is viewed by the NRC as proactive and positive.

Charcoal Bed Service TS 4.7.6.1.c requires the licensee to obtain a charcoal sample after each 720 hours of ventilation flow through the auxiliary building ventilation system (ABVS) charcoal bed. In the past, the licensee has normally performed these samples during outages. During the current Unit 2 operating cycle, an unexpectedly high number of in-service hours accumulated on the charcoal adsorber bed during the installation of a damper position switch design change. To support the design change installation, it was necessary to operate the Unit 2 auxiliary building ventilation in safeguards mode, exhausting through the charcoal adsorber bed. Accumulation of in-service hours during the previous refueling outage, combined with delays in the accomplishment of the design change, resulted in the charcoal adsorber bank remaining in service for longer than anticipated.

Charcoal Bed Surveillance During the routine surveillance test of the auxiliary building safeguards air filtration system, performed on June 11, 1994, the licensee identified that the charcoal adsorber bed had been in service for over 780 hours. If the ASTM D-3803-89 sample results indicated unacceptably high methyl iodide penetration, such results would require declaring the charcoal adsorber bank inoperable and replacing the charcoal within a 24-hour period.

Part of the scope of STP M-4, "Routine Surveillance Test of the Auxiliary Building Safeguards Air Filtration System," is to verify that operation of the charcoal adsorber bank has not exceeded 720 hours since the last laboratory analysis of a representative carbon sample. The NRC identified that the procedure is deficient in that it does not preclude exceeding the 720 service hours between samples.

TS 4.7.6.1.c requires that the laboratory sample results be verified within 31 days after sampling. The licensee determined that it would be possible to comply with the sample results time requirements, in this particular situation, by calculating the date at which the charcoal adsorber bed exceeded 720 hours of service and assuming that the 31-day period for obtaining the results started on that date.

Evaluation of Charcoal Sample Results After obtaining the initial sample results for the Unit 2 auxiliary building charcoal adsorber bed, which

revealed an increase in the methyl iodide penetration, evaluations of the operability of other charcoal adsorber beds were not immediately performed. Following receipt of confirmatory sample results, the licensee concluded that it was likely that the Unit 2 Fuel Handling Building (FHB) E-5 charcoal bed was inoperable and declared it as such. In the interim period, between obtaining the first and second sample results, spent fuel was moved in the Unit 2 spent fuel pool with the FHB E-5 charcoal bed in service. The licensee has scheduled replacement of the FHB E-5 charcoal in the near future, although analysis of the charcoal sample has not been completed.

Licensee Management Involvement After the discovery that the Unit 2 auxiliary building charcoal adsorber bed exceeded its sample periodicity, thus requiring a charcoal sample and analysis, it was evident that licensee management suspected that sample results would be above the allowed limit for methyl iodide penetration. More timely evaluation of the initial Unit 1 charcoal sample results could have alerted the licensee of the significance of those results regarding the performance of other charcoal beds. The licensee has completed a review of the operational status of all remaining safety related ventilation charcoal beds. The FHB E-5 charcoal bed has been declared inoperable and the licensee has preliminarily evaluated all others as operable. This evaluation was based either upon the time the bed has been in service since charcoal replacement and previous ASTM D-3803-79 sample analyses or sample analyses using the ASTM D-3803-89 method. In addition, the licensee has initiated procedures to closely track charcoal bed inservice hours to prevent exceeding required sampling intervals in the future.

Safety Significance The licensee's analysis of the as-found condition concluded that the Unit 2 auxiliary building charcoal bed iodine penetration was outside of the design basis. This was reported to the NRC in a 4-hour nonemergency report made on July 7, 1994. The licensee has since corrected the condition by replacement of the charcoal bed.

Conclusion The licensee's procedures for verification of charcoal adsorber bed operability did not ensure that charcoal samples were obtained at the specified periodicity. The failure to implement a procedure which ensures that the charcoal adsorber bed is sampled at the required periodicity is a violation of 10 CFR Part 50, Appendix B, Criterion V, which requires that activities affecting quality be prescribed by procedures appropriate to the circumstances and shall be accomplished in accordance with established procedures. Since this violation was identified by the licensee, and other criteria of Section VII.B(2) of the Enforcement Policy were satisfied, this violation was not cited (323/94-18-02).

6 ONSITE ENGINEERING (37551)

6.1 Insufficient CCW Flow to CC Pump Coolers

Background The CC pump includes five component coolers which are cooled by CCW: a gear oil cooler, lube oil cooler, two seal coolers, and a seal plate cooler. Each cooler has an upstream CCW isolation valve. The lube oil

cooler, gear oil cooler, and seal plate cooler have a common discharge header throttle valve which, until recently, was throttled to control the flow to the coolers. In 1990, a licensee Safety System Functional Audit and Review (SSFAR) questioned throttled CCW flow to the CC pumps. The audit raised the concern that throttled CCW flow may not provide design basis flow rates to the individual components during accident conditions and heat loads. The resolution of this audit finding accepted the throttled CCW condition and was based, in part, on an engineering calculation which contained two nonconservative assumptions. The licensee now considers the resolution of the audit finding, and acceptance of throttled CCW flow, to be in error.

On June 29, 1994, a 1-hour nonemergency report was made to the NRC regarding past operability of the CC pumps. The report stated that, previously, CCW flow to the CC pump coolers had been throttled during normal operation. This would not have provided adequate cooling to the CCW gear oil cooler during a design basis accident. CCW was throttled to maintain lube oil temperatures within the vendor recommended temperature band and, therefore, maintain proper viscosity during normal CC pump operation. Recent licensee investigation of CC pump lube oil characteristics revealed that there is very little change in viscosity, for the type of oil used, over the entire range of possible CCW temperatures. Therefore, throttling CCW flow to maintain lube oil temperatures was no longer a concern.

<u>CC Pump CCW Flow Testing</u> Prior to the 1-hour nonemergency report, preliminary CCW system flow rate testing had been completed. The tests indicated potentially insufficient CCW cooling to the CC pumps under design basis conditions. After the preliminary testing, the CCW throttle valve on the common discharge header of the CC pump gear oil, bearing oil, and seal plate coolers was fully opened. A prompt operability assessment was initiated to document the adequacy of the existing cooling flow with the common outlet throttle valve fully open.

The licensee is continuing testing and calculations to support proper adjustment of flow to each of the coolers. Flow is being measured by an acoustic sensor instrument, since the system does not have installed equipment which measures the flow rates. The throttling of CCW flow during testing is accomplished using the throttle valves to individual coolers. The licensee plans to use these test results to determine final valve throttle positions for establishing adequate flow rates to the CC pump components cooled by CCW.

<u>Conclusion</u> The licensee resolution of issues raised in the 1990 safety system functional audit and review regarding the CCW system incorrectly accepted the adequacy of the existing throttled cooling flow. The initial decision to throttle the CCW supply to the CC pumps does not appear to have been properly reviewed prior to establishing the throttled configuration procedure. The licensee's interim actions, which have been communicated to the CC pump vendor, appear to provide adequate cooling to the CC pumps. Further NRC review of this issue will be concluded during the review of the Licensee Event Report (LER).

7 PLANT SUPPORT ACTIVITIES (71750)

The inspectors evaluated plant support activities based on observation of work activities, review of records, and facility tours. The inspectors noted the following during this evaluation.

7.1 Fire Protection

During inspection of fire barrier penetration seals, the inspectors observed a break in the sealed barrier in the floor of the Unit 2 cable spreading room. The inspectors learned that the breach was a result of work ongoing in preparation of installing conduit for part of the new Eagle 21 reactor protection system, scheduled for completion during the upcoming Unit 2 refueling outage. The licensee was implementing the appropriate hourly fire watch compensatory measures for the breach.

7.2 Radiation Protection Controls

The inspectors periodically observed radiological protection practices to determine whether the licensee's program was being implemented in conformance with facility policies and procedures and in compliance with regulatory requirements. The inspectors also observed compliance with radiation work permits, proper wearing of protective equipment and personnel monitoring devices, and personnel frisking practices. Radiation monitoring equipment was frequently monitored to verify operability and adherence to calibration frequency. Several resident inspector tours in the radiological control area revealed minor cases of deficient contamination control practices in the 140-foot level of the FHB surface contaminated areas (SCAs). On several occasions, an area where work was being performed on ventilation components and ducting was poorly controlled. At several different locations, material from inside the SCA crossed over the SCA boundary. Similar deficiencies had been previously noted in the same area on several prior occasions. These deficiencies and the past poor performance was pointed out to plant management, and action was initiated to correct the deficiencies.

Conclusion It was apparent in each instance that action had been initiated to correct the deficiency, but the corrective action was not always adequate to prevent recurrence. Because prior corrective action did not result in effective long-term resolution, additional management involvement is needed.

7.3 Plant Housekeeping

The inspectors observed plant conditions and material/equipment storage to determine the general state of cleanliness and housekeeping. Housekeeping in the radiologically controlled area was evaluated with respect to controlling the spread of surface and airborne contamination.

On one plant tour the inspectors noted a small puddle of liquid on the floor near the Unit 1 Reciprocating Charging Pump 1-3. The inspectors informed Health Physics and noted on their next tour that the puddle was gone.

The inspectors observed a weakness in general plant cleanliness in some areas. This was of particular concern in the radiologically controlled areas of the plant. Two examples were as follows:

- Rubber gloves and glove liners were left within the SCA on the skid of CC Pump 2-1.
- A significant portion of the floor in the Unit 2 turbine-driven auxiliary feed pump room was stained due to a previously leaking component.

7.4 Security

The inspectors periodically observed security practices to ascertain that the licensee's implementation of the security plan was in accordance with site procedures, that the search equipment at the access control points was operational, that the vital area portals were kept locked and alarmed, and that personnel allowed access to the protected area were badged and monitored and that monitoring equipment was functional. The inspectors noted no problems in this area during this inspection period.

7.5 Conclusion

Overall, plant support performance was good during the inspection period and remained unchanged from the last period. Inspectors observed that housekeeping practices in contaminated areas were generally adequate, and could be improved.

8 FOLLOWUP - ENGINEERING (92903)

8.1 (Closed) Followup Item 323/93-30-01: Definition of Safe Shutdown Earthquake (SSE)

During review of SSE requirements for Diablo Canyon, an inspector noted that SSER 7, dated May 1978, and SSER 31, dated June 1991, implied that the NRC considered that the SSE for Diablo Canyon was the maximum credible force of an earthquake generated from the Hosgri fault, with a peak ground acceleration (PGA) of 0.75g. However, the licensee's Updated Final Safety Analysis Report (UFSAR), Section 3, indicated that the SSE for Diablo Canyon was the DDE, with a PGA of 0.4g. The inspector did not identify any failure of the licensee to comply with any NRC seismic requirements. The inspector initiated a followup item for further NRC review of the definition of SSE at Diablo Canyon.

Subsequent to the initial inspection, the NRC staff reviewed the seismic licensing basis for Diablo Canyon and clarified that the SSE for Diablo Canyon was the double design earthquake (DDE) as described in the UFSAR, Section 3. The NRC staff noted that all new modifications required analysis for both the DDE and Hosgri earthquakes. The NRC staff also noted that the NRC had

required the licensee to ensure they could safely shutdown both units following either a Hosgri or DDE earthquake. Based on NRC staff's agreement with the licensee's definitions for the SSE at Diablo Canyon, the inspector concluded that the followup item was resolved.

9 FOLLOWUP ON CORRECTIVE ACTIONS FOR VIOLATIONS (92702)

9.1 (Closed) Violation 50-323/94-11-01: Failure to Plan and Perform Maintenance in Accordance with Written Procedures

The NRC identified an instance where the licensee failed to establish a clearance during a repair of the Unit 2 reactor coolant system safety injection inlet to RCS Loop 2-3, which was the subject of a citation with NRC Inspection Report 50-323/94-11. In a letter dated June 20, 1994, the licensee acknowledged the violation and stated that corrective action had been completed for the specific instance cited and that further action to prevent recurrence had been initiated by revision of the licensee procedure AP C-4S1, "Temporary Modification Control - Plant Jumpers and M&TE." The inspector reviewed and verified these actions. The licensee's actions appeared to be appropriate and properly implemented.

10 IN-OFFICE REVIEW OF LERS (90712)

The following LERs were closed based on in-office review:

- 275/94-14, Revision 0 — Unplanned DG Start (ESF Actuation) Due to Shorting Indicating Lights
- 275/94-10, Revision 0 — Main Bank Phase "C" Transformer Degraded Condition
- 275/93-012, Revision 0 — ASW System Potentially Outside Design Basis
- 275/93-012, Revision 1 — ASW System Potentially Outside Design Basis
- 275/93-012, Revision 2 — ASW System Potentially Outside Design Basis

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

G. M. Rueger, Senior Vice President and General Manager, Nuclear Power Generation Business Unit
*W. H. Fujimoto, Vice President and Plant Manager, Diablo Canyon Operations
*R. P. Powers, Manager, Nuclear Quality Services
T. L. Grebel, Supervisor, Regulatory Compliance
J. S. Bard, Director, Mechanical Maintenance
*G. M. Burgess, Director, Systems Engineering
S. G. Chesnut, Reactor Engineer Supervisor
W. G. Crockett, Manager, Technical and Support Services
S. R. Fridley, Director, Operations
*B. W. Giffin, Manager, Maintenance Services
*J. D. Grammer, Engineer, Systems Engineering
*C. R. Groff, Director, Plant Engineering
*C. D. Harbor, Engineer, Systems 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
J. C. Kelly, Mechanical Group Leader, Nuclear Engineering Services
M. E. Leppke, Assistant Manager, Technical Services
J. J. McCann, General Foreman, Instrument Maintenance
*D. B. Miklush, Manager, Operations Services
M. D. Nowlen, Director, Instrumentation and Controls
P. T. Nugent, Engineer, Regulatory Compliance
S. R. Ortore, Director, Electrical Maintenance
B. H. Patton, Director, Technical and Support Services
*J. A. Shoulders, Director, Onsite Nuclear Engineering Services
D. P. Sisk, Senior Engineer, Regulatory Compliance
D. W. Spencer, Power Production Engineer, Plant Engineering
D. R. Stermer, Engineer, Systems Engineering
*D. A. Taggart, Director, Onsite Quality Assurance

1.2 NRC Personnel

*M. Miller, Senior Resident Inspector
*M. Tschiltz, Resident Inspector

*Denotes those attending the exit meeting July 27, 1994.

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

2 EXIT MEETING

An exit meeting was conducted on July 27, 1994. During this meeting, the inspectors reviewed the scope and findings of the report. 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

ABVS	auxiliary building ventilation system
ASME	American Society of Mechanical Engineers
CC	centrifugal charging (high head injection)
CCW	component cooling water
DDE	double design earthquake
FHB	fuel handling building
I&C	instrumentation and controls
IN	Information Notice
KV	kilovolts
LER	licensee event report
PGA	peak ground acceleration
RCS	reactor coolant system
RHR	residual heat removal
SCA	surface contamination area
SSE	safe shutdown earthquake
SSER	Supplemental Safety Evaluation Report
SSFAR	safety system functional audit and review
STP	surveillance test procedure
TS	Technical Specification
UE	unusual event
UFSAR	Updated Final Safety Analysis Report