APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-275/94-27
50-323/94-27

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: October 16 through November 26, 1994

Inspectors: M. Tschiltz, Resident Inspector
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Approved: D. F. Kirsch 12/21/94
D. F. Kirsch, Chief, Reactor Project Branch E Date

Inspection Summary

Areas Inspected (Units 1 and 2): Routine, announced inspection of operational safety verification, plant maintenance, surveillance observations, plant support activities, onsite engineering, foreign material exclusion control (TI 2515/125) and in-office review of licensee event reports (LERs).

Results (Units 1 and 2):

Operations

- Unit 2 initial criticality and physics testing was performed in a controlled and conservative manner. Personnel who performed the testing were well versed on the sequence of testing, procedural requirements and purpose of the testing.

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Engineering

- Surveillance test procedures for molded case circuit breakers were inappropriate to the circumstances in that the as-found condition of the circuit breakers was not verified prior to performing preventive maintenance and cycling of the breaker. This preconditioning situation is a violation.

- Vendor information indicated that an undersized spring pack had been used in the assembly of auxiliary feedwater (AFW) level control Valve LCV-107 during 1R6. Approximately six months after receiving information regarding the spring pack from the vendor, testing was performed which indicated that the incorrect spring pack had been installed in the LCV-107 motor operator. The timeliness of the investigation of this issue was inconsistent with the undersized spring pack's potential impact on system operability.

Maintenance

- Motor operated valve testing was performed in a manner which provided inaccurate test results for LCV-107 during 1R6. The licensee's initial investigation of the testing was inconclusive in determining the reason for the anomalous test results.

- During Power Range Nuclear Instrument N-43 incore/excore calibration, I&C technicians failed to place overtemperature delta temperature (OTDT) comparators in the tripped condition within the Technical Specification (TS) required time (3 minutes outside requirement). Inadequate communications during the calibration resulted in failure to place the inoperable OTDT channel in the tripped condition within the 6-hour limit.

Plant Support

- Radiation Protection efforts in response to the recently identified fuel defect were proactive in identifying areas where radiological conditions were affected. Efforts to understand the changes in radiological conditions, update postings and adjust work practices to minimize exposure appeared to be thorough and timely.

Summary of Inspection Findings:

- Inspection Followup Item 275/9427-01 was identified (Section 3.1).

- Noncited Violation 323/9427-02 was identified (Section 4.1).

- Violation 275/9427-03; 323/9427-03 was identified (Section 5.2.2).

- Noncited violation 275/9427-04 was identified (Section 5.4).

- LERs 275/93-001, Revision 2, and 323/94-011, Revision 0, were closed (Section 8).

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 was shutdown in Mode 6 for Refueling Outage 2R6 at the start of the report period. On October 28, Unit 2 entered Mode 1 (Power Operation) and was paralleled to the grid. On November 3, Unit 2 attained 100 percent power and operated at 100 percent power for the remainder of the report period.

2 OPERATIONAL SAFETY VERIFICATION (71707)

2.1 Diesel Generator 2-1 Walkdown

The inspectors performed a safety system verification inspection which involved a detailed inspection of a sample of Diesel Generator 2-1 system components. The inspectors used Operating Procedure J-6B:I, "Diesel Generator 2-1 Make Available," Revision 10, as guidance for verification of the system alignment. The inspectors found valve and circuit breaker positions to be aligned in accordance with the procedure. The inspectors concluded that for the sampled portions of the system, the Diesel Generator 2-1 system was in proper alignment.

2.2 AFW System Walkdown

The inspectors performed a safety system verification inspection which involved a detailed inspection of a sample of AFW pump 2-1 system components. The inspectors used Operating Procedure D-1:II, "AFW System - Alignment Verification for Plant Startup," Revision 17, as a reference for system alignment verification. The inspectors concluded that for the sampled portions of the system, the AFW pump 2-1 had been properly aligned.

2.3 Unit 2 Containment Walkdown Inspection

On October 23, 1994, while Unit 2 was in Mode 4, an NRC inspector conducted a walkdown of Unit 2 containment. All levels and areas of the containment were found to be free of loose debris. Additionally, in the areas inspected, all stowed material was found to be properly secured. The inspector noted several minor deficiencies including a 6 inch tear in the ventilation duct sleeve in the containment annular air circulation ring and a minor leak on an instrument fitting on Accumulator 2-1 level transmitter (LT-950). The licensee initiated Action Requests (ARs) to document these deficiencies and track corrective actions.

Material not authorized for storage in containment during power operations was noted to have been removed prior to the performance of NRC containment inspections. The licensee's procedure for ensuring removal of loose debris from containment appeared to have been effectively implemented.

2.4 Unit 2 Fuel Defect

On November 8, 1994, the licensee noted a significant increase in the concentration of dose equivalent Iodine 131 (I-131) in the reactor coolant. Over the next several days, dose equivalent I-131 concentrations increased to approximately 0.24 microcuries per gram. The increase of I-131 activity, in the order of magnitude noted, is indicative of defects in the fuel cladding. The increase in coolant activity was first indicated by increased background radiation levels in the vicinity of the radiation monitor (RE-19) which caused a steam generator (SG) blowdown isolation signal. After detecting a significant increase in I-131 activity, reactor coolant system (RCS) sampling periodicity was increased to detect any further changes in activity levels. The licensee initiated actions in accordance with Procedure TS 6.ID1, Revision 0A, "Failed Fuel Prevention and Mitigation Program," to closely track the development of the cladding defect and plan contingency actions should the RCS activity continue to increase.

Diablo Canyon TS 3.4.8 limits the specific activity of the reactor coolant to less than or equal to 1 microcurie per gram dose equivalent I-131. Current activity levels are significantly below the established limit.

Since this issue will continue for the duration of operation with failed fuel, the licensee's implementation of TS 6.ID1 will be evaluated during the normal course of NRC inspection. The resident inspector will continue to closely follow this issue.

3 PLANT MAINTENANCE (62703)

During the inspection period, the inspector observed and reviewed selected documentation associated with the 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

- LCV-107 Diagnostic Testing

Unit 2

- Replacement of Safety Injection (SI) Accumulator 2-4 Valve SI-2-8952D

- Investigation and repair of Nuclear Instrument Inverter IY-22 input rectifier failure

- Replace Loop 1 Wide Range T-Hot resistance temperature detector (TE-413) with spare resistance temeprature detector

3.1 LCV-107 Diagnostic Testing and Troubleshooting

The inspector observed corrective maintenance and testing for AFW Valve FW-1-LCV-107. LCV-107 is the level control globe valve to SG 1-2 which controls the discharge flow of the turbine driven AFW Pump 1-1. The valve was a Consolidated Controls Inc. drag valve with a Limitorque SMB-000-5 actuator.

The inspector reviewed the following related documents:

- AR A0339484, dated 5/4/94

- WO C0132606

- Procedure MP E-53.10V, Valve Operation Test and Evaluation System Test (VOTES) Procedure

- Procedure MP E-53.10P, SMB-000 Maintenance

AR A0339484 identified, in May, 1994, a licensee concern for the potential installation of an improper spring pack in the actuator for the motor-operated valve (MOV). Although this was found to be a problem in three other actuators procured by the licensee under the same purchase order, the licensee had originally not considered the problem to affect LCV-107 because diagnostic test results at the time had indicated that the proper spring pack was installed. As part of their investigation, the licensee had decided to confirm the diagnostic test results during scheduled surveillance testing of LCV-107.

During initial attempts to perform the confirmatory diagnostic testing in November 1994, the primary thrust transducers used during earlier testing had not been functional. Neither the VOTES sensor (yoke mounted strain gage) nor the QTS sensors (stem mounted strain gages) were functional. As a result, during this testing, the licensee used a clamp on calibration device in accordance with their procedure to measure thrust during a short closing stroke to verify that the torque switch was adjusted to meet the required setpoint.

The static diagnostic testing was finally performed in November 1994, under Work Order (WO) C0132606. Using the alternate transducer, the as-found thrust setting for the torque switch was measured to be only 1600 lbf rather than 3600 lbf, as previously measured. The minimum required thrust for the valve, documented in design calculation J-31, is 2656 lbf. The licensee considered the use of the alternate transducer to be of equivalent accuracy to the transducers previously used. The licensee found no evidence of degradation which might explain a decrease from the previous thrust value. The licensee concluded that the low thrust measurement was consistent with an extra light spring pack being mistakenly installed in the actuator, as previously suspected. The licensee took immediate action to replace the spring pack. The licensee initiated nonconformance report (NCR) NRC-DC1-94-EM-N054 to investigate the cause of the discrepant test results.

The preliminary results of the licensee investigation indicated that the cause of the discrepant test data was an isolated case of a gross calibration error during previous diagnostic testing. The cause of the error, which is undetermined at this time, will be investigated by the licensee. The inspector was concerned that the previous VOTES measurements appeared to be substantially in error and may be indicative of a programmatic deficiency in the licensee's conduct of diagnostic testing. At the time of the inspection, the licensee was considering the need for additional diagnostic testing to determine the scope of the potential problem.

The inspector observed corrective maintenance and testing for AFW Valve FW-1-LCV-107.

The licensee performed an assessment of LCV-107 operability based on the as-found condition of the valve. Conservatism in the design analysis, which considered degradation of stem lubrication, torque switch setting repeatability, and the dynamic forces in the valve, were eliminated in the assessment. The results of the analysis indicated that LCV-107 was capable of performing it's required safety function during the time period with the incorrect spring pack was installed.

The licensee investigation of the vendor information, indicating that an incorrectly sized spring pack may have been installed in LCV-107, was not completed in a timely manner. Sufficient information existed which called into question the initial test data indicating the correct spring pack was installed. Review of the results of this issue, documented in NCR-DR1-94-EM-N054, will be an inspection followup item (275/94-27-01).

4 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 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

- Surveillance Test Procedure (STP) I-38-A.1, Revision 1, Solid State Protection System (SSPS) Train A Actuation Logic Test in Modes 1, 2, 3, or 4
- STP P-7B, Revision 26A, Routine Surveillance Test of Auxiliary Saltwater Pump 1-1
- STP M-8A1, Revision 1, Overall Leak Rate Testing of the Personnel Air Lock

Unit 2

- STP I-2D, Revision 3, Nuclear Power Range Incore/Excore Calibration
- STP I-2B, Revision 22, Nuclear Power Range Channel Analog Channel Operational Test
- STP R-31, Revision 4, Rod Worth Measurements Using Rod Swap Method
- STP R-30, Revision 8, Reload Cycle Initial Criticality
- STP I-36-S3F, Revision 0, Protection Set III RCS Narrow Range Hot Leg Streaming Factors Update
- STP M-13B3, Revision 4, Emergency Diesel Safeguard Auto Timer Setting Verification for Loads Actuated by SSPS Train A Slave Relay K609
- STP M-13B4, Revision 4, Emergency Diesel Safeguard Auto Timer Setting Verification for Loads Actuated by SSPS Train B Slave Relay K609
- STP M-13G, Revision 9, 4KV Bus G Non-SI Auto-Transfer Test
- STP M-13H, Revision 7, 4KV Bus H Non-SI Auto-Transfer Test

4.1 STP I-2D; Nuclear Power Range Incore/Excore Calibration

Instrument and Control (I&C) technicians performed power range instrument incore/excore calibrations (STP I-2D) for Unit 2 on October 31 through November 1, 1994. The surveillance procedure accomplished the calibration of the delta flux indications, and the delta current penalty input to the OTDT reactor trip setpoint and power range ion current A&B module.

The inspectors observed selected portions of the STP I-2D which performed the calibration of power range channels N44 and N43. The I&C technicians were familiar with the calibration procedure and appeared to possess adequate knowledge and skills to conduct the calibrations. During the calibrations of loop three temperature channels, the licensee discovered that it failed to place the inoperable channels in the tripped condition within 6 hours after placing the channels in bypass. TS 3.3, Table 3.3-1, Action 6, Step a. requires that inoperable channels be placed in the tripped condition within 6 hours of becoming inoperable. The channels were tripped approximately 3 minutes after exceeding the 6-hour time limit.

The licensee initiated an AR and a NCR to document the missed TS requirement. Miscommunication between the I&C and Operations personnel regarding the time the temperature channels were placed in bypass resulted in exceeding the TS time limit.

TS actions require placing the channel in the tripped condition within 6 hours to preclude extended periods with the channel being out of service for maintenance or surveillance testing. Out-of-service times are based on maintaining the appropriate level of reliability of the Reactor Protection system. The 3 minute time period beyond the 6 hour TS time limit, during which the protection function was not properly controlled, did not significantly affect plant safety.

The failure of the shift foreman to communicate and control the performance of I&C surveillance testing on reactor plant instrumentation is considered to be a weakness. The licensee has committed to establish a 5-hour administrative time limit on the use of Eagle 21 channel bypass features. The licensee's failure to place the channel in a tripped condition within 6 hours of the channel becoming inoperable is a violation of TS 3.3.1, Table 3.3-1, Action 6, Step a. Since the violation is of very low safety significance and the inspectors are satisfied with the adequacy of the corrective actions, in accordance with Section VII.B.(2) of the Enforcement Policy, this violation was not cited (323/94-27-02).

4.2 STP I-2B; Nuclear Power Range Channel Analog Channel Operational Test

The inspectors observed the bistable adjustment per STP I-2B, Section 6.3, for Unit 2 channel N43 on November 1, 1994. STP I-2B verified proper operation of the bistables for power range channels and provided for adjustment of the bistables. During the observations, the inspectors noted the I&C technicians were inconsistently applying correction factors to the meter indications to obtain a reference value to be compared to the actual power indication. In one case, the correction factors were added when they were positive, and in the other case, the correction factors were subtracted when they were positive. The inspectors questioned the I&C personnel present regarding the discrepancy in applying the correction factors, and they stopped work and informed their supervisor. The I&C supervisor reviewed this discrepancy and informed the I&C personnel how to properly apply the correction factors. The correct interpretation was that positive corrections were to be added to the meter indication, while negative corrections were to be subtracted from the meter indication. The I&C supervisor informed the technicians performing the procedure of their mistakes and provided guidance on the correct method for applying the correction factors. During further observation by the inspectors of setting the power range bistables by the I&C personnel, no other deficiencies were noted.

The improper application of correction factors to reference values for meter indications in this situation was noted during the process prior to any supervisory reviews of the surveillance data. Additionally, the magnitude of the corrections were small and would not have had resulted in any of the trip setpoints being improperly adjusted, given the as-found data.

The licensee has reviewed the procedure and considers it to be adequate. The procedure does not contain specific instructions on the method of applying correction factors. The correction factors read from a graph are both positive and negative. The inspector evaluated the licensee's actions and concluded that the correction factors could reasonably be expected to be applied correctly by the I&C technicians without providing additional guidance in the procedure. The improper application of the correction factors is considered to have been an isolated occurrence caused by lack of attention to detail by technicians. The licensee reviewed other calibrations that had been performed using the procedure, no others were found to have improper corrections applied. The actions taken in response to this issue have been evaluated by the inspector and appear to be appropriate.

4.3 STP I-38-A.1, SSPS Train A Actuation Logic Test in Modes 1, 2, 3, or 4

STP I-38-A.1 was performed on November 3, 1994, by the I&C group. The STP provided the instructions for the SSPS Train A Actuation Logic Test, Master Relay Test, and Reactor Trip Breaker A (52/RTA) Trip Actuating Device Operational Test. TS 3.3.1, Table 3.3-1, Item 20, Action Statement 10 in Modes 1 or 2, allowed one Reactor Trip Breaker to be bypassed for up to 2 hours for surveillance testing provided the other channel is operable. The STP provided a caution note and a place to record the time when the bypass Breaker 52/BYA was closed. Later, the procedure recorded the time the bypass Breaker 52/BYA is opened, which stopped the two hour TS time limit. The inspectors observed the entire STP without noting any deficiencies.

4.4 STP I-36-S3F; Protection Set III RCS Narrow Range Hot Leg Streaming Factors Update

The inspector noted several issues during the performance of Protection Set III RCS Narrow Range Hot Leg Streaming Factors Update. Paragraph 8.4.23.e requires that the three streaming factors (S1, S2, S3) entered into the Man-Machine Interface be the same as the values printed in a previous step. The inspector noted that one of the three values had been rounded down after entry. When the I&C technicians were questioned regarding the difference, they stated that this question had been previously resolved and that the small change in the value did not affect the update of the streaming factors. The technicians did not know if the evaluation of this condition had been formally documented. The inspector later verified that this condition had been evaluated and was documented as being an acceptable condition in the Procedure SC-I-36-M, Revision 0, "Eagle 21 Tunable Constants."

During the verification of the streaming factors, accomplished by paragraph 8.6.26, the updated streaming factors, when applied to obtain corrected loop temperatures, yielded temperatures that had a difference of greater than 1°F. Paragraph 8.6.26 establishes a 1°F limit for the difference between corrected temperatures and specifies that if the criteria can not be met that the supervisor's permission must be obtained to proceed. The supervisor was notified and directed the technicians to repeat the procedure for determining the streaming factors. Following the second calculation of streaming factors, differences between corrected temperatures were still greater than 1°F. The system engineer was then contacted and indicated that there was no technical requirement for the 1° difference specified in the procedure and that it was a checkpoint which was added to the procedure to ensure that the previous entries had been made properly. After the technicians determined that the entries had been made correctly, the procedure was continued and completed.

The technicians had a thorough understanding of the equipment being operated. Although the technicians were not cognizant of where the justification for the rounding of the tunable constant was documented, they did know that the rounding had been previously evaluated as being acceptable.

The inspector verified that the limit of 1° difference between corrected temperatures was not an absolute limit for acceptability of the streaming factors. In this regard, the 1° limit provided an additional assurance of the validity of results prior to the completion of the surveillance. The criteria for corrected temperature differences is an issue which will be evaluated by the licensee.

4.5 STP-M-13H; 4KV Bus H Non-SI Auto-Transfer Test

On October 19, 1994, during pretest alignment of Unit 2 Containment Fan Cooling Unit 4, per Section 12.8.6 of the subject procedure, test personnel noted that the fan had tripped on thermal overload while running in low speed for approximately 30 minutes. After waiting 30 minutes the overload would not reset. AR A0355270 was issued to investigate the problem. The fan was restarted for testing and WO C0132195 was issued to investigate and repair as required, to prevent recurrence.

Test personnel took prompt actions to address the unexpected fan trip, and complete satisfactory startup testing. Licensee investigation and inspection of the thermal overload did not reveal any abnormal condition in containment fan cooler unit (CFCU) 2-4 breaker cubicle. Infrared thermography of the thermal overload and related components was performed on CFCU 2-4 and CFCU 2-1. Comparison of the results did not reveal any abnormal temperatures during CFCU 2-4 operation. CFCU 2-4 has been operated multiple times and run for extended time periods since the overload trip with no subsequent trips. The licensee investigation identified that the CFCU had been started five times during the shift prior to the thermal overload relay tripping and that the repeated switching of the fan resulted in the tripping of the thermal overload relay.

The inspector reviewed the licensee investigation and resolution of the overload trip of CFCU 2-4 and determined that it appeared to be appropriate.

5 ONSITE ENGINEERING (37551)

5.1 Foreign Objects in Secondary Side of Unit 2 SGs

During inspections performed of the secondary side of the Unit 2 SGs, during Outage 2R6, following sludge lancing of the tubesheets, the licensee found and retrieved the following foreign objects from the areas identified below:

SG 2-2; a brass piece from a flashlight, an insulation clip, two insulation tabs, two battery tops, a flashlight spring, a small wire 1/2 inch long.

SG 2-3; one piece of weld slag.

SG 2-4; A tungsten inert gas electrode (3 inches long, 1/8 inch diameter) sharpened at both ends to a sharp point, two welding rods (9 inches long with a U bend and a 2-inch section 1/16 inch diameter). Attached to the welding rods were two sections of remaining flux (2 1/4 inches long and one 1/3 inch long).

In addition, the strainer which was used during SG sludge lancing contained six small pieces of metal which cannot be identified as coming from any one SG.

The identification and removal of these foreign objects from the inspection areas were documented in ARs A0354471 and A0355103. These ARs had evaluation due dates of November 15, 1994.

The licensee's initial evaluation of the foreign objects found in SG 2-4, noted the following in AR A0355103:

- The weld rod could not have come from any of the work that was performed during the current outage (2R6). The weld rod was identified as a 3/32 inch diameter carbon steel shielded metal arc welding rod. Only 1/8 inch carbon steel weld rod was used inside SG 2-4 during the current outage, for welding the internal hatches.

- Tungsten inert gas tungsten electrodes were used for the SG feed water ring J-tube modification work performed this outage. At the time they were used, all foreign material exclusion (FME) barriers were in place. After the J-tube modification work the area was vacuumed and inspected prior to removal of any FME barriers. Additionally, none of the welders reported losing any of their electrodes. It should also be noted that the tungsten electrode could have been in the SG for many years. The SG environment would have little or no effect on the electrode.

Discussion with the licensee resulted in the following:

- During this 2R6 outage the licensee had completed eddy current inspection of all four Unit 2 SGs. These inspections did not identify any tube damage that could be attributed to the foreign objects found in the secondary side of the SGs.

- The licensee procedures did not require any additional eddy current inspection of the SG tubes, as a result of the identification the subject foreign objects after the initial eddy current inspections.

The licensee had implemented additional actions to evaluate the foreign objects found in Unit 2 SGs. The inspector determined, based upon his review and discussion, that licensee's actions to evaluate the foreign objects found in the Unit 2 SGs during 2R6 appeared appropriate.

5.2 <u>Molded Case Circuit Breaker Setpoints</u>

The inspector reviewed the licensee's design controls for the sizing and testing of molded case circuit breakers (MCCB).

The inspector found that the licensee's Nuclear Engineering Services (NES), Electrical Engineering Group had design responsibility for MCCB sizing.

The inspector reviewed NES Procedure EE-8, Revision 0, "Guides for Selection/Setting/Testing of MCCBs and Sizing of Thermal Overload Heaters." This NES procedure established the licensee's standard and the method for sizing MCCBs. The inspector also reviewed Calculation 195B-DC, Revision 5, "Magnetic Breaker Setting For Class 1E Motors." This calculation evaluated the adequacy of MCCBs for 480 VAC motors for MOVs. Both of these documents discussed in detail the design basis and criteria used by the licensee to establish MCCB settings.

The inspector found that the licensee's procedure established adequate fault protection capability and conservatively sized the instantaneous magnetic trip point of MCCBs at 234 percent of locked-rotor current to avoid nuisance tripping of the motor during starting.

5.2.1 Adjustable Magnetic Setting

MCCBs incorporate a magnetic element which will trip the breaker "instantaneously" for all currents above a certain value. The design of some MCCBs incorporate an adjustable setting which allows selection of the desired nominal instantaneous trip current. The manufacturer predicts the trip current within a tolerance for each setting of the adjustable magnetic device.

The inspector noted that the licensee's setpoint analysis for adjustable MCCBs was limited to only one of a range of selectable settings for the breaker trip current. Each position setting corresponded to a single nominal trip current value. Licensee Procedure EE-8, paragraph 4.1.2.5, stated a margin allowance was added to the given single nominal trip current value to determine an acceptable trip current range. The margin range was based on the tolerance for expected variations in performance during testing. The licensee used the tolerance of -25 percent and +40 percent as recommended in National Electrical Manufacturers Association (NEMA) Standard AB-2-1980, "Procedures for Verifying the Performance of MCCBs."

As such, the inspector considered that the trip current range was specified in the calculation for component performance testing purposes with the breaker set at the specified setting. The trip current range specified in the calculations did not appear to define a range of functional trip current requirements which could be satisfied by any specific setting of the breaker.

The magnetic trip settings determined from these calculations were included in configuration controlled Drawings 50024/50011 which were the "electrical device lists" used by plant electrical maintenance to obtain appropriate trip setpoints to verify MCCB operability following breaker installation and during periodic testing. The inspector noted that the drawings contained a note which identified that the specified settings were "nominal" and that the actual setting may require slight adjustment during field testing.

The inspector reviewed Electrical Maintenance Procedure MP E-64.1A, Revision 24, "AC and DC Molded Case Circuit Breaker Test Procedure." The licensee used this procedure to install and periodically test MCCBs. The inspector noted that paragraph 2.8 stated:

> "If the breaker fails to trip at the magnetic trip setting, this breaker may be adjusted plus or minus one position, if applicable."

The inspector questioned the licensee's practice of allowing field adjustment of the breaker setting during verification testing and observed that field adjustment was not allowed for the containment penetration protection breakers. The inspector noted that paragraph 2.6 of EE-8 stated:

> "Due to the inability to reproduce controlled factory calibration conditions in the field, manufacturer's tolerances cannot be reproduced. The purpose of this test is to verify operability within reasonable margins of the manufacturer's predicted behavior."

Furthermore, NEMA AB-2-1980, Part 3, "Verification Test Procedures," stated, in part,

> "These tests are based on proven maintenance practices and are aimed at assuring that the circuit breaker is functionally operable. This is in contrast to those tests which are designed to specifically check the accuracy of the manufacturer's published calibration performance curves and which must be done under precisely controlled ambient temperature and electrical conditions."

The inspector found that, if necessary, the licensee used the verification test data to effectively recalibrate the breaker setting to a trip current different from that specified by the manufacturer. The inspector found that there was no analysis in the design calculations to support a range of allowable settings as directed by Procedure MP E-64.1A. Furthermore, allowing an adjustment of "minus one position" did not appear to satisfy the design criteria in the licensee's design Procedure EE-8 to assure that the minimum trip current for a given setting would be greater than the worst case operational current.

The licensee performed a preliminary review of the breaker settings for all safety related MCCBs. According to the licensee, approximately 10 percent of the noncontainment penetration breakers had been adjusted during testing and were left with settings other than as specified in the electrical design calculations.

The licensee acknowledged the inspector's concerns but considered that their practice of recalibration of the breaker during field testing was consistent with the intent of NEMA AB-2-1980. Further, the licensee considered that the trip range specified in their electric equipment list did define the functional trip current requirements.

In addition to fault interrupting capacity evaluation, the licensee stated that the functional requirement to interrupt the expected fault current was evaluated in separate calculations. The maximum trip current of the breaker was evaluated to assure it was less than the minimum fault current. Using this criteria, the functional requirement was satisfied regardless of the setting of the breaker. Furthermore, for containment penetration protection breakers, if the maximum withstand current was more limiting than the NEMA test tolerance, then the trip current range was limited to the withstand current.

The inspector noted that NRC Information Notices 92-51, Supplement 1, and 93-64 dealt with similar deficiencies in the application and testing of MCCBs.

Although the implementation of licensee Procedure MP E-64.1A apparently deviated from the intent of the NEMA standard during testing, the inspector concluded that the licensee's controls of adjustable MCCBs were adequate. The inspector considered that allowing the single setting deviation from the design setting did not introduce a significant breaker safety concern for inadvertent breaker tripping due to the conservative breaker sizing criteria used by the licensee.

5.2.2 Breaker Preconditioning

The inspector reviewed licensee Procedure STP M-83A, "Penetration Overcurrent Protection," Revision 15. The licensee used this procedure to satisfy the periodic testing requirements of TS 4.8.4.2 for containment penetration protection circuit breakers. For electrical testing of MCCBs, the surveillance procedure directed the performance of electrical maintenance Procedure MP-E-64-1A.

The inspector reviewed licensee Procedure MP-E-64-1A, "AC and DC Molded Case Circuit Breaker Test Procedure," Revision 24. The inspector noted that the licensee performed preventative maintenance on the MCCBs prior to testing. Mounting bolt tightening, pivot point lubrication and manual exercising through three cycles were performed prior to overcurrent trip testing. In addition, the as-found condition was not being recorded for the breakers. The inspector questioned whether the preconditioning of the breaker prior to trip testing was appropriate since the trip testing was intended to periodically demonstrate the functional operability of the breaker and detect any degradation. The inspector was concerned that the test data obtained after exercising may not be representative of the as-found capability of the breaker.

In response to the inspector's concerns, the licensee initiated a review of the adequacy of their MCCB test procedures.

Diablo Canyon TS 4.8.4.2.a.2 required that MCCBs used for containment penetration overcurrent protection shall be demonstrated to be operable every 18 months by performing functional trip current testing to assure that degraded breaker performance will not go undetected. However, the licensee's STPs did not test the as-found condition of the breakers because they performed preventative maintenance prior to the test. This inadequacy in the licensee's procedures used to satisfy TS surveillance requirements was a violation (NRC Inspection Report 50-275/94-27-03, 50-323/94-27-03).

The ability of the circuit breakers to isolate a bus from a faulted load was verified by the licensee by performance of TS 4.8.4.a.2 testing. The licensee practice of performing additional cycling of the breakers, or, in certain instances, other maintenance, prior to the performance of the testing, had the potential to affect circuit breaker test results, and is considered a violation in that the procedure for performing the surveillance test did not determine the as-found condition of the breaker and was, therefore, inappropriate.

The NRC follow-up of the violation will address the impact of the preconditioning on the functional trip current testing performed.

5.3 4 Kv Bus H Voltage Decay During Emergency Diesel Generator (EDG) 2-2 Testing

Initial hot restart testing of EDG 2-2, performed during 2R6, failed to meet the test acceptance citeria. STP M-9G, "Diesel Generator 24-Hour Load Tests," required that EDG 2-2 start and load onto 4 kv bus H within 10 seconds. During the initial performance of the test in 2R6, EDG 2-2 failed to meet the 10-second acceptance criteria, starting and loading onto the bus in 10.5 seconds. The test measured the time from opening of the 4 kv

Bus H auxiliary feeder breaker until the closing of the EDG 2-2 output breaker. Review of test data revealed that the period from opening the auxiliary feeder breaker until the first level undervoltage relay actuated was approximately 3 seconds. The relay was verified to be functioning at the correct voltage. The licensee reviewed the conditions present during the test, including the Bus H voltage profile and determined that the failure to meet the test criteria was caused by minimal bus loading during test performance. The lightly loaded bus resulted in a slower decrease in bus voltage which, in turn, delayed the actuation of the first level undervoltage relay. Subsequent testing, inadvertently performed with residual heat removal pump 2-2 running at the start of the test, yielded acceptable test results. The EDG started and loaded onto 4 kv Bus H in 8.22 seconds. The additional bus load, provided by residual heat removal pump 2-2, resulted in the faster decrease in 4 kv Bus H voltage and consequently the reduction of the time from the opening of the auxiliary feeder breaker to the first level undervoltage relay initiation.

Testing which resulted in hot restart times of greater than 10 seconds was performed with minimal loading of the 4 kv bus. The procedure for performing the test, STP M-9G, directed realignment of 4 kv powered equipment prior to de-energizing the 4 kv bus to ensure required components remained in operation during the test. In this regard the test did not establish conditions which were representative of those which would be encountered during normal operation. During normal operation the loading of the 4 kv bus would have been greater. During testing performed with the 4 kv bus loading representative of normal operation (i.e. greater 4 kv bus loading) EDG 2-2 performed acceptably within the test acceptance criteria.

EDG 2-2 was evaluated by the licensee to respond within the requirements for hot restart testing. The licensee is preparing a procedure revision to STP M-9G to ensure that during hot restart testing there is certain load established on the 4 kv bus. The inspector reviewed the licensee's actions in response to this issue and determined that they appeared appropriate.

5.4 Design Change Package (DCP) N-4874

DCP N-4874, Revision 1, and associated Design Change Notices (DCN) DCN2-EC-4874 with Field Change C-18020; DCN2-EE-4874; DCN2-EJ-4874; and DCN2-EN-4874 with Field Changes M-18099 and M-17988 were reviewed by the inspectors. This design change replaced the existing drive mechanism of the personnel airlock with a more reliable industry proven design having less maintenance requirements. The inspectors found that a number of changes had been made in this DCP with "white out." The changes were made to the listing of attachments on the cover sheet, in the summary section and later text in the DCP. The changes were typographical in nature and had no impact on the technical content of the design change.

However, this was a violation of Procedure HR2.ID1, "Signatures and Signature Responsibilities," Revision 2, paragraph 5.8. The procedure requirements are for all records to be changed with a single line through the part to be changed (strike out), initials of the person making the change, and date of the change.

Although identified by the NRC, the safety significance of this problem was low since examples noted were typographical in nature and had no impact on the technical content of the design change.

The licensee documented this violation in AR A0351628 after it was identified by the inspectors. This AR was subsequently closed in error and another AR A0359688 was issued. The licensee determined that it was not a common practice, nor an uncommon practice to use "white out." In addition, it was reported by the licensee that "correction tape" was also used. The practice of using both "white out" and "correction tape" has been stopped. The licensee is also establishing the scope of this problem, i.e. how many design documents were changed in this manner. Early results indicated that the problem is limited. The licensee has initiated corrective actions to prevent future changes from being made with white out or correction tape. This violation is not being cited because the requirements of 10 CFR Part 2, Appendix C, Section VII.B.(1) were met (323/94-27-04).

6 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.

6.1 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 observed health physics technicians removing contaminated clothing from the disposal bin in a surface contaminated area adjacent to the Unit 1 containment personnel access hatch. The technician inside of the surface contaminated area was attired in appropriate protective clothing, and removal of contaminated clothing was conducted in a manner to preclude the spread of contamination.

6.2 Unit 2 Increased Radiation Levels

The inspector reviewed the actions being implemented in response to the increased radiation levels within the Radiologically Controlled Area (RCA) with the Radiation Protection Director. Due to the increased amount of activity within the Unit 2 RCS radiation levels in certain areas within the RCA have increased significantly. Areas where significant increases were noted within the RCA were primarily those in the vicinity of the chemical and volume control system piping and components and connected support systems such as the liquid and gaseous radwaste systems.

The inspector noted that the response of the RP organization to the changes in radiological conditions appeared to be directed at minimizing occupational dose and performed in a timely manner.

7 FME CONTROLS (TI 2515/125)

7.1 The NRC has issued a number of generic communications regarding emergency core cooling system strainer clogging due to debris. Inspectors at several nuclear facilities have identified practices which could present the potential problem of introducing foreign materials into systems important to safety. To address these issues, Temporary Instruction 2515/125, "FME Controls," was issued for NRC inspectors to determine whether or not licensees have implemented effective procedures to prevent foreign material from inadvertently entering safety systems during maintenance activities, outages, and routine operations.

7.2 Inspection Observations

The inspector reviewed the procedure which covered foreign materials exclusion at Diablo Canyon: AD4.ID6, "FME Program," Revision 1. AD4.ID6 has specific requirements and goes into detail on how to accomplish foreign materials exclusion controls around the reactor cavity, spent fuel pool, and all other susceptible areas in the plant. It is a general procedure which provides a number of different alternatives for accomplishing foreign materials exclusion controls. For example, there are four appendices to the procedure governing Foreign Materials Exclusion (FME). They include: 7.1) Standard FME Plan; 7.2) High Risk FME Plan; 7.4) Reactor Cavity FME Plan; and 7.4) Spent Fuel Pool FME Plan. The procedure and appendices provided guidelines and responsibilities for monitoring the material condition of the Diablo Canyon facility in the areas of FME.

The inspectors reviewed past NRC inspection reports, interviewed licensee personnel, and reviewed several licensee documents. The documents included ARs, Quality Assurance audit reports, Material Deficiency and Housekeeping Reports, and WOs.

It was evident that the licensee's Quality Assurance group had placed a significant amount of attention in this area. The licensee's report on the performance of foreign materials controls during the 1994 refueling outage noted a general improvement over past outages. Examples of problems which the licensee identified were: (1) log keeping not rigorously maintained and (2) tool accountability was not properly maintained. An extensive Quality Assurance audit was conducted from October 26 to December 1, 1993 (Audit 930431). The inspector reviewed the audit and compared the findings with actions taken by the licensee to address the results of the audit. The inspector noted that the audit findings were incorporated in Revision 1 of AD4.ID6. Interviews with cognizant personnel indicated that they were aware of the changes.

The inspector reviewed training materials and a test administered to personnel involved with FME activities. The inspector concluded that the training appeared to be adequate. The test was a multiple choice format examination and appeared to validate that examinees had been properly trained. The inspector concluded that the test was adequate to ensure an evaluation of the knowledge level required; however, the examination appeared to be marginal in reinforcing important points.

The inspector noted that AD4.ID6 did not provide for FME controls on activities that are not governed by WOs. Specifically, filling and venting of systems not directed by a WO does not require the use of a standard FME plan. The inspector views this as a potential weakness. However, the inspector recognized that the FME controls associated with filling and venting systems are minimal in scope.

Based upon the inspectors discussions and evaluations, the inspector concluded that the licensee's foreign materials exclusion program had improved over that of the previous year. Although, as noted by the licensee in Quality Assurance Audit 930431, December 22, 1993, some weakness in the administration aspects of the program were noted. Specifically, repeated situations with log errors and tool/material accountability occurred during both 1R5 and 2R5 outages. The auditors characterized this pattern of repeated errors as a precursor to a programmatic breakdown in the area of FME administrative controls. The enhancements made following the audit were evaluated by the inspector and appeared to have addressed the concerns of the auditors. The inspector concluded that the licensee's FME program appeared to be administered in a manner that would prevent the loss of FME integrity.

8 IN-OFFICE REVIEW OF LERs (90712)

The inspectors performed review of the following LERs associated with operating events. Based on the information provided in the report, review of associated documents, and interviews with cognizant licensee personnel, the inspectors concluded that the licensee had met the reporting requirements, had addressed root causes, and had taken appropriate corrective actions. The following LERs are closed:

- 275/93-001, Revision 2 Component Cooling Water System Potentially Outside of Design Basis Due to Nonconservatism in Design Basis Analysis
- 323/94-011, Revision 0 TS 3.3.1 Violation During a Nuclear Instrument Calibration 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
W. H. Fujimoto, Vice President and Plant Manager, Diablo Canyon Operations
R. P. Powers, Manager, Nuclear Quality Services
*J. S. Bard, Director, Mechanical Maintenance
J. R. Becker, Shift Supervisor, Operations
D. H. Behnke, Senior Engineer, Regulatory Compliance
S. G. Chesnut, Supervisor, Reactor Engineering
*W. G. Crockett, Manager, Technical and Support Services
T. F. Fetterman, Group Supervisor, Electrical Engineering
S. J. Foat, Electrical Engineer, Electrical Maintenance
S. R. Fridley, Director, Operations
B. W. Giffin, Manager, Maintenance Services
*T. L. Grebel, Supervisor, Regulatory Compliance
R. Gray, Director, Radiation Protection
*C. R. Groff, Director, Plant Engineering
*J. A. Hayes, Director, Onsite Quality Control
*K. A. Hubbard, Engineer, Regulatory Compliance
*H. S. Iyer, Power Production Engineer, Design Control, Plant Engineering
M. E. Leppke, Assistant Manager, Technical Services
*D. B. Miklush, Manager, Operations Services
*T. A. Moulia, Assistant to the Vice President and Plant Manager
*D. H. Oatley, Director, Material Services
*S. R. Ortore, Director, Electrical Maintenance
*H. J. Phillips, Director, Instrumentation and Control
D. L. Ricca, Maintenance Engineer, Maintenance Engineering
P. G. Sarafian, Senior Engineer, Nuclear Quality Services
*J. A. Shoulders, Director, Technical Support, Nuclear Engineering Services
*D. A. Taggart, Director, Onsite Quality Assurance

1.2 NRC Personnel

*M. Tschiltz, Resident Inspector

*Denotes those attending the exit meeting on December 2, 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 December 2, 1994. During this meeting, the resident inspector 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

AFW	auxiliary feedwater
AR	action request
CFCU	containment fan cooler unit
CVCS	chemical and volume control system
DCN	design change notice
DCP	design change package
EDG	emergency diesel generator
FME	foreign material exclusion
I&C	instrumentation and control
LCV	level control valve
LER	licensee event report
MCCB	molded case circuit breaker
MOV	motor-operated valve
NCR	nonconformance report
NEMA	National Electrical Maintenance Association
NES	Nuclear Engineering Services
OTDT	overtemperature delta temperature
RCA	radiological controlled area
RCS	reactor coolant system
SG	steam generator
SI	safety injection
STP	surveillance test procedure
SSPS	solid state protection system
TS	Technical Specification
VOTES	valve operation test and evaluation system
WO	work order