U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report No. 50-275/89-33 and 50-323/89-33

License Nos. DPR-80 and DPR-82

Licensee: Pacific Gas and Electric Company
77 Beale Street
Room 1451
San Francisco, California 94106

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

Inspection at: Diablo Canyon Power Plant, Units 1 and 2

Inspection Conducted: December 4, 1989 to January 17, 1990

Inspectors: P. P. Narbut, Senior Resident Inspector
M. F. Miller, Reactor Inspector
C. B. Ramsey, Reactor Inspector

Approved by: F. R. Huey, Chief, Engineering Section 2/6/90 Date Signed

Summary:

Inspection During the Period of December 4, 1989 through January 17, 1990 (Report No. 50-275/89-33 and 50-323/89-33)

Areas Inspected: The inspection included follow-up of open items and a review of licensee actions in response to those open items. NRC Inspection Procedures 30703, 64704, TI-2515/87 and 92701 were used during this inspection.

Results of Inspection and General Conclusions:

1. A violation was identified concerning a lack of administrative controls requiring operability of the positive displacement charging pump, which is required for safe shutdown in the event of a fire. NRC guidance in the form of Generic Letter 81-12 was available, and would have helped the licensee to avoid this violation, had it been carefully reviewed and implemented.

2. 53 discrepancies between the as-built plant and the approved fire protection program were identified by the licensee, some as early as 1982. The program should have been changed to resolve these discrepancies according to the license condition, which requires evaluation pursuant to 10 CFR 50.59. The licensee stated that NRC guidelines in Generic Letter 86-10 were unclear, and this was given as the reason for the lack of resolution of this issue. However, the license condition takes precedence over Generic Letter 86-10.

In conclusion, the inspectors found that the licensee failed to properly implement the requirements of the fire protection program in two instances.

Summary of Violations or Deviations:

Two deficiencies identified during this inspection appear to involve violations of NRC requirements.

1. Technical Specifications 6.8.1 requires procedural implementation of Regulatory Guide (RG) 1.33 items, one of which is the fire protection program. The fire protection program requires the positive displacement charging pump (PDP) to be operable so it can provide charging for safe shutdown in the event a fire disables both centrifugal charging pumps. Contrary to these requirements, there were no administrative controls requiring the PDP to be operable to ensure safe shutdown. This violation is documented in the attached Notice of Violation.

2. Technical Specifications 6.8.1 requires procedural implementation of RG 1.33 items, one of which is instrument calibration. Licensee procedure AP C-450 "Preventative Maintenance Program" requires that RG 1.97 instrumentation be entered in the Recurring Task Scheduler and calibrated within the required interval. Contrary to these requirements, RG 1.97 instruments for battery 11, 12, 13, 21, 22, and 23 voltage and current indicators were past the three year calibration interval. Also, Unit 2 4kV bus F and H voltage and Unit 2 battery 21, 22, and 23 voltage and current were not documented in the Recurring Task Scheduler. This violation is documented as a non-cited violation in this inspection report.

Open Items Summary:

Thirteen open items were reviewed. Nine were closed, four remain open.

DETAILS

1. Persons Contacted

PG&E

*J. Townsend, Plant Manager
*M. Angus, Assistant Plant Manager, Technical Services
*W. Kelly, Regulatory Compliance Engineering
*T. Allen, NECS, Project Engineer
*W. Crockett, APM, Supervisor Services
*B. Rinkacs, Regulatory Compliance
*P. Roller, System Engineer
*D. Taggart, Director, Quality Support
*T. Rapp, Chairman, Onsite Safety Review Group
*G. Giffin, Assistant Plant Manager, Main Services
*D. Mikliesh, Assistant Plant Manager, Operations Services
*E. Connell, NECS, Project Engineer
*J. Shoulders, NECS, OPEG Project Engineer
*W. D. Barkhuff, Acting QC Manager
W. Yap, System Engineer
B. Smith, NECS Project Engineer
R. Washington, Manager, I&C Maintenance
R. Johansen, I&C Maintenance
D. Baur, Supervisor, Electrical Maintenance
R. Panero, NOS, Fire Protection Engineer
R. Kohout, Supervisor, Emergency/Safety Services
H. Iyer, Design Change Engineer
S. Hamilton, Design Change Engineer
A. Nicholson, Nuclear Safety and Regulatory Affairs
J. Fuhriman, Quality Control
M. Spoutz, System Engineer
R. Clark, Supervisor, Mechanical Engineering
U. Farradj, Engineering, Mechanical
P. Kao, Lead, Safety Related Mechanical

*Attended the Exit Meeting on December 8, 1989.

2. Licensee Action on Previously Identified Items

A. (Open) Open Item 50-275/87-27-04, Resolution of Differences between Plant Configuration and Approved Fire Protection Program. In an earlier report, an inspector noted that the licensee identified several (50) specific discrepancies between the NRC approved fire protection program and the as-built plant configuration. More significant examples are: 30 ft rather than the approved 50 ft separation between trains, 2 hour penetration seals which reduce fire protection of the approved 3 hour fire barriers in which they are installed, and some safe-shutdown circuits without the approved 2 hour fire rated barriers. The licensee had documented evaluations of each of these discrepancies and considered each to be satisfactorily resolved because

compensating fire protection (detection and supression) is provided and because the licensee considers the existing fire protection adequate. Based on these evaluations, the licensee concludes that the level of fire protection in the plant is acceptable.

Since discrepancies exist between the approved program and the plant, the licensee should change the program in accordance with the applicable license condition. This condition requires that changes to features of the approved program be evaluated according to the requirements of 10 CFR 50.59; and where these changes do not conform to the criteria of 10 CFR 50.59, the evaluation should be submitted to the NRC staff for review and approval. Concerning the evaluations, the inspectors noted the following:

(1) The evaluations did not specifically address the criteria of 10 CFR 50.59.

(2) The licensee's assumptions concerning fire hazards, transient loading, fire barriers, and propagation of smoke and hot gasses did not appear to be consistent with the guidance of Branch Technical Position (BTP) CMEB 9.5.1, which provides 10 CFR 50 Appendix R guidance concerning evaluation of fire protection. In this regard, the licensee stated that transient and in-situ combustible loading was computed using the licensee's Fire Protection Database System (FPDS). This is a database compiled from plant walkdowns. It also computes values for combustible loading and duration of design basis fires. During discussions the licensee agreed to compare the criteria for fire hazard evaluation listed by Branch Technical Position with the criteria used by the licensee and submit this comparison to the NRC with the evaluations of changes to the program discussed above.

(3) Some evaluations did not provide sufficient information to allow review by a person who is not familiar with the plant.

To give a specific example, Fire Protection Program Change Evaluation FHARE No. 25 discusses the issue of separation between redundant trains for the centrifugal charging pumps. The NRC granted an exemption request to not provide a 3 hour fire barrier between the pumps because it was stated there was a 50 foot separation between undampered duct penetrations between the fire areas. The licensee later determined that the distance was actually about 30 linear feet. The evaluation does not specifically address the issue that the probability of malfunction of equipment important to safety may be increased. The evaluation concludes that the configuration is acceptable as is based on available fire suppression and detection, and combustible loading (fire severity) less than 30 minutes and 60 minutes for the areas. The inspector considers that if a fire initiates in this area, there is a greater likelihood of loss for a 30 foot vice a 50 foot separation between trains. The shorter distance allows less dissipation of heat and smoke, which can increase the probability of a malfunction of safety equipment. The significance of this increased probability

of malfunction will be assessed by detailed NRC review. However, it is an increased probability of a safety system malfunction and should be addressed according to 10 CFR 50.59. In addition, the assumption of 30 and 60 minute fires is based on licensee computer database calculations which appear to have different assumptions than BTP CMED 9.5.1 for combustible loading. Detailed NRC review will assess the acceptability of the licensee combustible loading calculations with respect to evaluations of fire hazards.

Although an acceptable level of fire protection may be the preliminary NRC conclusion, a detailed NRC fire protection review is required to confirm this conclusion. In a meeting on January 17, 1990 with NRC Region V, the licensee stated that each of the descrepancies would be evaluated according to Nuclear Safety Analysis Center guidelines for 10 CFR 50.59 evaluations. These evaluations will be submitted to the NRC for review. The licensee stated these evaluations would be submitted by April 15, 1990. After reviewing the submittal, the NRC will issue a Safety Evaluation Report. In this manner, a detailed NRC evaluation will be completed, and the approved fire protection program will become consistent with the as-built plant.

According to the license condition, changes to approved program features which decrease the level of fire protection should be made with an application for license amendment pursuant to 10 CFR 50.90. At least 15 of these discrepancies appear to reduce the level of fire protection provided by features documented in the NRC approved program. Because fire supression, fire watches and other compensating measures are provided, the inspectors preliminarily conclude that safety in the plant has not been reduced to an unacceptable level. However, these types of changes appear to reduce the margin of safety and increase the probabilistic risk to the plant.

The compensating fire protection measures (fire watches) which the licensee is implementing are allowed by NRC fire protection regulations, and are considered by the NRC to ensure acceptable fire protection to compensate for the descrepancies between the approved program and as built plant. However, the NRC considers these measures acceptable interim corrective measures only. Since these discrepancies have been identified by the licensee as early as 1982, these compensating measures appear to have been implemented over the longer term. Therefore, this issue should be resolved promptly since the licensee should not continue to rely on short term measures to fulfill long term requirements.

The length of time between initial identification of this issue by the NRC (6/10/87) and this inspection (12/4/89) is significant. The licensee has been aware the NRC is concerned that this issue has not been resolved. The inspector reviewed 5 internal PG&E memoranda issued over a 15 month period in which PG&E fire protection personnel requested PG&E engineering and licensing staff to determine appropriate resolution and notification of the NRC. These memoranda stated that regulatory issues may be involved, and that

prompt resolution was desired. However, no communication was made with the NRC to resolve this issue. The timeliness of corrective action will be evaluated after technical resolution of this item is complete.

B. (Closed) Unresolved Item 50-275/89-17-02, Out of Service Positive Displacement Charging Pump Which is Redundant to Centrifugal Charging Pumps.

The FSAR requires the positive displacement charging pump (PDP) to be available for safe shutdown in the event a fire renders the centrifugal charging pumps inoperable. Inspection report 89-17 identified that the PDPs in both units had been out of service for extended periods during the previous two years. Furthermore, inspection report 89-17 noted that the pumps had each been continuously out of service for over three weeks during calendar year 1989.

(1) As corrective action for lack of PDP operability controls, the inspector verified that the licensee implemented administrative controls to ensure that the PDP was returned to operability in one week, and that the centrifugal charging pump rooms had operable fire detection and supression systems, fire watches, limited fire loading and limited hot work. Justification for Continued Operation 89-13 was issued which increased restrictions on operation and maintenance as a result of PDP inoperability.

The lack of administrative controls to ensure operability of the PDP, which is required for safe shutdown, is considered a violation of Technical Specification 6.8.1, which requires implementation and use of procedures for RG 1.33, Appendix A items, which includes the fire protection program. This is considered a violation (50-275/89-33-01). As a corrective action, the licensee changed the charging system operating procedure to require a 7 day limit on the time the PDP can be out of service during plant operation, after which, plant shutdown must be initiated. The inspector verified licensee corrective action for this violation, and considered it adequate.

(2) The inspector reviewed the licensee's 10 CFR 50.59 evaluation for the inoperable PDP, and found it did not completely address the question of whether or not there is a possibility of a malfunction of a type different than that evaluated in the safety analysis report. The licensee stated "There is no change in the configuration of either Unit that would create the possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report." However, the inspector found that the FSAR fire hazards analysis (Section 9) requires the PDP as a backup in the event of a fire in the charging pump room. This indicates that the associated analysis assumed the PDP would be available to maintain RCS inventory. Because the FSAR fire hazard analysis

requires use of the PDP as a backup, shutdown of the plant during a fire without use of a charging pump was not addressed in the FSAR fire hazard analysis. In response to this particular question, the licensee should have noted that operation without PDP to maintain RCS inventory may be outside the requirements of the Fire Hazard Safety Analysis. The licensee did not address maintaining RCS inventory in the 50.59 evaluation. However, in the associated justification for continued operation, the licensee stated that shutdown, if required, would be implemented according to the plant procedure "Loss of All Charging". The inspector considers that the procedure "Loss of All Charging" meets the safety analysis requirement concerning RCS inventory since, according to the licensee, the procedure should allow a safe shutdown while maintaining RCS inventory such that pressurizer level remains in the indicating band as required by 10 CFR 50, Appendix R, Section L.2.b. The licensee (P. Kao) agreed to attach this 50.59 review to the April 15 submittal for further NRC evaluation to determine if the evaluation properly addressed the criteria of 10 CFR 50.59.

(3) Generic Letter (GL) 81-12, which specifically addressed operability requirements for safe shutdown equipment, was addressed to plants licensed before January 1, 1979. Diablo Canyon was not licensed at that time, but had been issued a construction permit, had received GL 81-12, and was participating in resolution of NRC fire protection requirements in other areas. A cursory review of this letter would have highlighted these weakness in the Diablo Canyon program. However, except for equipment already addressed in Technical Specifications for other purposes, the licensee has not provided these administrative controls. The licensee apparently did not recognize the current lack of administrative controls covering safe shutdown equipment operability. To date, there are administrative controls to ensure equipment operability only for equipment already controlled by Technical Specifications (for reasons other than safe shutdown during a fire), and for the PDP as a result of this specific issue.

(4) As one of the corrective actions associated with the lack of administrative controls for safe shutdown equipment, the licensee is evaluating the type of additional administrative controls required for safe shutdown equipment. The inspector reviewed the applicable NCR reports and minutes of Technical Review Group meetings, and attended a Technical Review Group meeting concerning this issue.

The licensee has characterized safe shutdown equipment operability controls as follows: equipment controlled by Technical Specifications, safety related equipment, and non safety related equipment.

(a.) Technical Specification Equipment

The licensee concluded that equipment controlled by Technical Specifications requires no additional controls. For example, Technical Specifications require operability of the source range nuclear instrumentation in all modes but Mode 1. Based on equipment history, the licensee found that, although significant periods of maintenance have occurred, at least one source range instrument has been operable during Mode 1. This operability is prudent, since the change from Mode 1 to Mode 3 can be unscheduled. The licensee assessment appears to be adequate.

(b.) Safety Related Equipment

The licensee found that operability of some safety related equipment is not directly controlled, however, administrative maintenance procedures place a high priority on returning safety related equipment to service. Based on a review of equipment history which showed that all inoperable safety related equipment was promptly returned to service, the licensee concluded that the existing adminstrative controls requiring a high priority for maintenance of safety related equipment already ensure acceptable operability of safety related equipment required for safe shutdown. This assessment appears to be adequate.

(c.) Non Safety Related Equipment

The licensee determined that several items required for safe shutdown were non safety related. The operability records of the specific valves in the ASW and CCW systems were reviewed, and the licensee determined that there had been no significant periods of inoperability. In addition, the safe shutdown function of several valves is to remain shut and not change position. Therefore, if the valve is made inoperable for maintenance on the operator, its safe shutdown function is not impaired. The licensee concluded, however, that based on the lower maintenance priority of non safety related equipment and experience with the PDP, operability controls may be required for non safety related equipment.

Based on the review of the operability records, the licensee determined that, with the exception of the PDP, there are no indications to date that required safe shutdown equipment would not have been operable. At the time of this report the licensee is considering what type of administrative controls would be most appropriate to implement for non safety related equipment to ensure operability of items required for safe shutdown. Options under consideration are procedural controls, inclusion of requirements in the maintenance of database to automatically include operability requirements in work

planning, and other methods. Based on the licensee's corrective action to date and tracking of this item in the licensee's NCP and Technical Review Group process, this item is closed.

C. (Closed) Unresolved Item 50-275/88-02-05, Calibration of Regulatory Guide 1.97 Instrumentation. An earlier NRC inspection observed that calibration intervals for some Regulatory Guide (RG) 1.97 instrumentation were not documented. Specifically Unit 2 4 kV bus H voltage, Unit 2 bus F voltage, Unit 2 battery 21, 22, and 23 voltage and current had no calibration recorded. Also, calibrations for several other RG 1.97 instruments showed that they were past the three year calibration interval and therefore overdue for calibration; specifically Unit 1 battery 11, 12, and 13 voltage and current.

After a review of all RG 1.97 instruments, the licensee determined that calibration intervals for these and 12 other instruments had not been entered into the Recurring Task Schedule (RTS) after installation. The licensee verified that calibration of battery 11, 12, 13, 21, 22 and 23 voltage and current indicators were overdue.

The battery voltage and current instrumentation calibrations were due 6/30/88 and performed between 7/7/88 and 7/12/88. Five of these instruments were found in tolerance, the sixth was out of tolerance by 5% of scale, which is more than the 2% tolerance allowed.

The licensee entered the missing instrumentation calibration tasks into the RTS and performed calibrations as required. This item is closed.

Since licensee procedure AP C-450 "Preventative Maintenance Program" requires RG 1.97 instrument calibration intervals to be entered in the RTS, the fact that these required preventative maintenance tasks were not included in the RTS is a violation of AP C-450 and therefore a violation of NRC requirements. The violation is considered a non cited violation in accordance with 10 CFR 2, Appendix C, V.G.1 (50-275/89-33-02).

D. (Open) Unresolved Item 50-275/88-02-01, Resistor Networks Used as Isolators. The licensee uses resistor networks instead of isolation amplifiers to isolate some Class IE instrument signals from Class 2 systems. The licensee stated that computer analyses supported satisfactory resistor network performance as isolators, but that no test had been performed on these isolation devices.

Design and analysis information for RCS pressure and steam generator wide range level indication resistor networks has been provided to the NRC staff for further analysis. This item will remain open pending conclusion of the NRC evaluation.

E. (Open) Deviation 50-275/88-02-02, RG 1.97 Requirements for Redundancy and Diversity of Steam Generator Level Indication. A common power inverter (Division IV) supplies power to all four wide range steam

generator level indication channels. Deviation 88-02-02 was issued because a single power source does not meet RG 1.97 diversity requirements. The licensee stated that this configuration met the requirements of RG 1.97 because steam generator narrow range level indication and auxiliary feedwater flow are powered by diverse electrical busses. NRC inspection report 88-02 noted that steam generator level appears to fall below the range of narrow range indication during several accident scenarios.

This issue is under technical review by the NRC staff and will remain open.

F. (Open) Unresolved Item 50-275/88-02-03, Lack of Recorders Used for Post Accident Monitoring for Neutron Flux Indication. Neutron flux is identified as a Type A, Category 1 variable which must be monitored in the post accident phase. RG 1.97 requires that this variable be recorded.

The licensee submittals to the NRC for RG 1.97 variables listed neutron flux as continuously indicated, but not recorded. However, this discrepancy with RG 1.97 was not identified as such. This submittal is under NRC staff review. Therefore, this item remains open.

G. (Closed) Follow-up Item 50-275/89-17-01, Inadequate Procedural Guidance to Ensure Fire Protection Design Review. As a result of inadequate design review, two fire protection water supply system valves were not verified operable upon completion of a modification. This was documented in LER No. 323/89-03. The licensee determined the cause to be inadequate procedural guidance and insufficient communication between engineers. The licensee took corrective action to revise or issue procedures, and to train individuals.

To evaluate licensee corrective action, the inspector reviewed procedure AP C-1S8, "Design Change Operability Testing Program", which was issued to specify the review of design changes with respect to operability testing requirements. This procedure required specific interaction between various engineering and operational groups to ensure an adequate scope of review for design change operability tests. This procedure appeared adequate.

The inspector reviewed procedure AP C-1S1, "Onsite Plant Modification Administrative" which was revised to expand the design change sponsor's scope of review and required increased interaction between various engineering disciplines and plant departments. The inspector also observed that design change sponsors are specifically required by procedure 3.60N "Operating Nuclear Power Plant Design Change" instructions and checklist to review fire protection and other interface requirements for design changes.

Based on this review and discussions with plant personnel involved in the design change process, this corrective action appeared adequate. This item is closed.

H. (Closed) Followup Item 50-275/89-17-03, Delayed Notification of Offsite Fire Department. During a wildland fire, notification of the California Department of Forestry was delayed 10 to 15 minutes because the sheriff's office followed "Unusual Event" procedures to notify other state agencies before notifying the California Department of Forestry to respond to the fire. Also, during the event, the shift supervisor incorrectly assumed that the NRC resident inspector would provide updates of the event to the NRC Headquarters Operation Center.

The inspector reviewed the licensee procedure EP M-6 "Emergency Procedure, Non Radiological Fire," which was revised to add a step requiring the shift foreman to "immediately notify the California Department of Forestry, San Luis Obispo County Fire by calling 911 or by using CDF radio/telephone." This resolves the previous lack of prompt, direct notification of offsite fire fighting assistance.

Concerning NRC notifications during events, the inspector observed that this and other emergency procedures require licensee personnel to notify NRC Headquarters directly. The assumption that the NRC resident inspector would provide updates to NRC Headquarters appears to have been an isolated personnel error. Licensee corrective action included training of operators to remind them that NRC residents do not provide updates to NRC Headquarters. Based on these corrective actions, this item is closed.

I. (Closed) Follow-up Item 50-275/89-17-04, Corrosion of the Fire Water System Piping. The licensee identified corrosion products in the turbine building fire water system piping. Immediate tests to identify pipe wall thinning and sprinkler head plugging showed that the corrosion and resulting particulate had not impaired fire protection. These tests and evaluations appeared adequate. The licensee initiated an engineering evaluation to determine the long term effects of this corrosion and possible prevention or mitigation of the corrosion.

The inspector reviewed the licensee analysis and assessment of corrective actions, and considers it adequate. The analysis determined dissolved oxygen to be the cause of corrosion. The licensee is evaluating options of corrosion inhibitors, changes in pipe flushing schedules, and other means of reducing dissolved oxygen concentrations. Based on licensee actions to date and tracking of this item in the licensee NCR and commitment tracking system, this item is closed.

J. (Closed) Followup Item 50-275/89-17-05 Sealing of Fire Barrier Penetration. Generic Letters 88-04, 88-56 and 89-52 require the licensee to evaluate the seals on dampers, doors and sealing materials to identify sealing problems.

The inspector found that the licensee is performing these evaluations and investigating various means to ensure adequate seals. These evaluations range from damper test and observation programs to the option to temporarily shut down ventilation if

dampers can not be closed against air flow. Based on adequate licensee effort to date and the fact that this item is being tracked by licensee commitments, this item is closed.

K. (Closed) Follow-up Item 50-323/89-02-01, Three Annunciator Windows are Necessary for Unambiguous Accident Mitigation System Actuating Circuitry (AMSAC) Indication. The control room annunciator indication for armed AMSAC is an input into the AMSAC trouble window signals. Therefore, the control room operators can not quickly and accurately diagnose an AMSAC trip without relying on other information.

To correct the ambiguous indication, the licensee will add a third AMSAC annunciator window labeled "AMSAC trip," which provides unambiguous indication of an AMSAC trip. The inspector observed that this third window has been installed in Unit 1, and is scheduled to be installed in Unit 2 during outage 2R3 (March, 1990). The third window is scheduled to be installed in the simulator by February, 1990. These schedules are documented in the licensee action requests and commitment tracking systems. Based on installation of the "AMSAC trip" window in Unit 1 and documentation of plans for these installations in Unit 2 and in the simulator, this item is closed.

L. (Closed) Follow-up Item 50-323/89-02-02, Separation of AMSAC from the Reactor Protection System (RPS). The licensee has not committed to meet the electrical separation requirements of RG 1.75. However, the separation between the AMSAC input signal wiring (steam generator low level and turbine impulse pressure) and the Class 1E wiring in the RPS analog process cabinet was determined to be less than desirable. In each RPS channel analog cabinet, Class 2 spare AMSAC wires as well as the Class 2 AMSAC signal wires from the isolation amplifier were bundled with Class IE RPS wiring. The AMSAC spare wires bundled with signal wires from the four RPS cabinets entered the AMSAC logic cabinet, where FSAR Class 1E separation criteria were followed for the signal wire. However, all four channels of the spare wires, which were taped at both ends, were coiled in the lower section of the AMSAC cabinet. In this configuration, a failure in the AMSAC non-class 1E cabinet could negate protective actions because of lack of physical separation between the output of isolators and Class IE wiring. However, this possibility is small considering that the highest potential in the AMSAC cabinet is 120V AC for processor power supplies, and the rest of the AMSAC circuitry is of much lower energy.

The inspector noted that the Unit 1 AMSAC cabinets had been modified to have no spare signal wires. The AMSAC Class 2 signal wires in the RPS cabinets are bundled with RPS Class 1E signal wiring, however, the AMSAC wiring was wrapped with varglass, which meets the FSAR physical separation criterion. The input wires from all 4 channels were bundled together in the AMSAC logic cabinet, however, the individual input wires were wrapped with varglass. Although 5 inch separation between each channel would be more desirable, the insulation meets the separation criteria of the FSAR. In addition, the low energy of the circuits in the AMSAC cabinets reduces the

likelihood of a problem. The Unit 2 AMSAC cabinets are scheduled to be modified in the same manner during outage 2R3 (March 1990). The licensee is tracking the commitment to complete this design change. This item is closed.

M. (Closed) Follow-up Item 50-323/89-02-03, AMSAC Testing Procedures for AMSAC at power testing and refueling outage end-to-end testing were not finalized. Also, the test requirements had not been incorporated in the licensee Recurring Task Schedule.

The inspector reviewed procedure STP-I-92A, Surveillance Test Procedure AMSAC functional test. It appeared adequate. The inspector found that at power surveillance test procedures for Unit 1 had been approved and scheduled. Unit 2 procedures had been prepared, but could not be scheduled until the end of the outage (April, 1990). Preparation of refueling outage end-to-end testing and entry of test requirements into the Recurring Task Schedule were in progress for Unit 1 and scheduled for Unit 2. The inspector verified that PG&E letter DCL-88-049 committed the licensee to quarterly at power testing of AMSAC, end to end (refueling outage) testing every 18 months, and entry of the surveillances in the Recurring Task Schedule. Based on these observations and licensee commitments, this item is closed.

7. Exit Meeting

An exit meeting was held with the licensee staff on December 8, 1989. The specific concerns addressed in this report were discussed with the licensee during the above meetings and were acknowledged by the licensee.