U.S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos: 50-275/93-07 and 50-323/93-07

Docket Nos: 50-275 and 50-323

License Nos: DPR-80 and DPR-82

Licensee: Pacific Gas and Electric Company
Nuclear Power Generation, B14A
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P. O. Box 770000
San Francisco, California 94177

Facility Name: Diablo Canyon Units 1 and 2

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

Inspection Conducted: March 9 through April 12, 1993

Inspectors: M. Miller, Senior Resident Inspector
F. Gee, Resident Inspector

Approved by: for 5-19-93
P. Johnson, Chief — Date Signed
Reactor Projects Section 1

Summary:

Inspection from March 9 through April 12, 1993 (Report Nos. 50-275/93-07 and 50-323/93-07)

Areas Inspected: The inspection included routine inspections of plant operations; maintenance and surveillance activities; followup of onsite events, open items, and licensee event reports (LERs); and selected independent inspection activities. Inspection Procedures 40500, 41701, 60705, 61715, 61726, 62703, 71707, 86700, 90712, 92700, 92701, and 93702 were used as guidance during this inspection.

Results

General Conclusions on Strengths and Weaknesses

Strengths:

Members of the licensee's staff, both the line and quality assurance organizations, promptly identified several problems to management and to the NRC. Examples included improper closure of the containment equipment hatch (paragraph 4) incorrect dowel dimensions in safety-related check valves (paragraph 9), and two observations discussed in paragraph 18.

Weaknesses:

A weakness was identified in the licensee's failure to completely close the Unit 2 containment equipment hatch in preparation for defueling, resulting in a 1/2 inch gap at the top of the hatch during fuel movement (paragraph 4).

Significant Safety Matters:

None

Summary of Violations:

Three non-cited violations were noted. One violation, identified and corrected by the licensee, involved the failure to fully secure the containment equipment hatch (paragraph 4). The others involved a gas bottle secured to safety-related emergency diesel generator air start piping (paragraph 20) and omission of actions required by a diesel generator test procedure (paragraph 8).

Open Items Summary:

Six items were opened, three enforcement items were closed, one followup item was closed, and nine LERs were closed.

DETAILS

1. Persons Contacted

Pacific Gas and Electric Company

G. M. Rueger, Senior Vice President and General Manager, Nuclear Power Generation Business Unit
J. D. Townsend, Vice President and Plant Manager, Diablo Canyon Operations
W. H. Fujimoto, Vice President, Nuclear Technical Services
*D. B. Miklush, Manager, Operations Services
B. W. Giffin, Manager, Maintenance Services
*W. G. Crockett, Manager, Technical Services
*J. E. Molden, Director, Instrumentation and Controls
R. P. Powers, Manager, Support Services
T. L. Grebel, Supervisor, Regulatory Compliance
*J. S. Bard, Director, Mechanical Maintenance
*H. J. Phillips, Director, Electrical Maintenance
*J. A. Shoulders, Projects Engineer, Onsite Project Engineering Group
D. A. Taggart, Director, Quality Performance and Administration
S. R. Fridley, Director, Operations
T. A. Moulia, Assistant to Vice President, Diablo Canyon Operations
M. R. Tresler, Manager, Nuclear Engineering Services
*K. A. Hubbard, Senior Engineer, Regulatory Compliance
*L. R. Collins, Senior Supervisor, Quality Assurance
D. R. Lampert, Coordinator, Outage Management
M. O. Somerville, Senior Engineer, Radiation Protection
J. J. Griffin, Group Leader, Onsite Engineering
J. E. Fields, Lead Engineer, Quality Control
*W. T. Rapp, Chairman, Onsite Safety Review Group
*M. Burgess, Director, Technical Services, System Engineering
R. Gray, Director, Radiation Protection
J. M. Gisclon, Manager, Nuclear Operations Support
*R. L. Kelmenson, Senior Licensing Engineer, Nuclear Regulatory Services
*D. Bell, Quality Control Supervisor, Nuclear Construction Services
*M. E. Leppke, Assistant Manager, Technical Services

*Denotes those attending the exit interview.

The inspectors interviewed other licensee employees including shift supervisors, shift foremen, reactor and auxiliary operators, plant technicians and engineers, and maintenance and quality assurance personnel.

2. Operational Status of Diablo Canyon Units 1 and 2

During this inspection report period, Unit 1 operated at 100% power, except for March 12, when the 12 KV power cable for circulating water pump (CWP) 1-2 failed. At that time, operators reduced unit power to 50% and tripped CWP 1-2. Portions of the failed cable, as well as other Unit 1 CWP power cables, were replaced. The unit was returned to full power on March 16. This issue was discussed in NRC Inspection Report No. 93-03, and is discussed further in paragraph 4 of this report.

Unit 2 was shut down for its fifth refueling outage during the inspection report period. At the end of the period, Unit 2 was in Mode 5.

3. Operational Safety Verification (71707)

a. General

During the inspection period, the inspectors observed and examined activities to verify the operational safety of the licensee's facility. The observations and examinations of those activities were conducted on a daily, weekly or monthly basis.

On a daily basis, the inspectors observed control room activities to verify compliance with selected Limiting Conditions for Operation (LCOs) as prescribed in the facility Technical Specifications (TS). Logs, instrumentation, recorder traces, and other operational records were examined to obtain information on plant conditions and to evaluate trends. This operational information was then evaluated to determine whether regulatory requirements were satisfied. Shift turnovers were observed on a sampling basis to verify that all pertinent information on plant status was relayed to the oncoming crew. During each week, the inspectors toured accessible areas of the facility to observe the following:

(1) General plant and equipment conditions

(2) Fire hazards and fire fighting equipment

(3) Conduct of selected activities for compliance with the licensee's administrative controls and approved procedures

(4) Interiors of electrical and control panels

(5) Plant housekeeping and cleanliness

(6) Engineered safety features equipment alignment and conditions

(7) Storage of pressurized gas bottles

The inspectors talked with control room operators and other plant personnel. The discussions centered on pertinent topics of general plant conditions, procedures, security, training, and other aspects of the work activities.

b. Radiological Protection

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 verified that health physics supervisors and professionals conducted frequent plant tours to observe activities in progress and were aware of significant plant activities, particularly those related to radio-

logical conditions and/or challenges. ALARA considerations were found to be an integral part of each RWP (Radiation Work Permit).

c. Physical Security

Security activities were observed for conformance with regulatory requirements, the site security plan, and administrative procedures, including vehicle and personnel access screening, personnel badging, site security force manning, compensatory measures, and protected and vital area integrity. Exterior lighting was checked during backshift inspections.

d. Leakage of RHR valve in Sample Line: The inspector noted boric acid crystals on an RHR sample valve, indicating a recent packing leak. This was not expected since the valve was isolated by two other valves in the sample lines. Investigation by the licensee identified that leakage of reactor coolant was occurring through two sample isolation valves. The licensee evaluated the rate of leakage with respect to potential post-accident radiation exposure, and determined that exposures would not be significant. The licensee concluded that repair of the valves during the next scheduled outage would be appropriate since the leakage appeared to be very small.

No violations or deviations were identified.

4. Onsite Event Follow-up (93702)

a. Degradation of Circulating Water Pump 12 KV Electrical Cable Jacket

Background: Failures of buried 12 KV electrical cables leading to Unit 1 circulating water pumps (CWPs) were experienced on February 5 and March 12, 1993. The March 12 failure occurred during this report period. The licensee had also previously experienced three failures of buried 4 KV electrical cables, two associated with safety-related auxiliary salt water (ASW) pumps, and the third associated with non-safety-related 4 KV electrical components in the intake structure. The licensee reported these failures to the NRC in voluntary LER 50-275/93-05. Both of the Unit 1 CWP 12 KV cable failures involved severe jacket degradation. These cable failures were previously discussed in NRC Inspection Report No. 50-375/93-03, paragraph 4.

Licensee's Actions: As a result of the CWP failure on March 12, 1993, the licensee replaced the local section of the three remaining CWP circuits (one circuit for CWP 1-1 and both circuits for CWP 1-2). The section replaced was the portion of the cables from the switchgear in the turbine building to the second pull-box over the discharge structure.

During the 1993 Unit 2 refueling outage, the licensee also replaced the same section of the safety-related auxiliary salt water (ASW) pump 1-1 cables as a precautionary measure. As an additional diagnostic action, the licensee replaced the entire length of Circuit A of Unit 2 CWP 2-1, a length of approximately 1400 feet.

The licensee sent samples of all cables to three independent laboratories for analysis. The licensee documented these and other investigative and corrective actions in non-conformance report (NCR) DC1-93-EM-N010.

Summary: The inspectors saw no observable jacket degradation for either the 12 KV cables from Circuit A of CWP 2-1 or the 4 KV cable sections from ASW pump 1-1 that were replaced. The licensee is continuing to pursue the root causes of the failures. The licensees laboratory analyses have established that the failure mechanism for the 12 KV cables is chemical attack. The 4 KV cable analyses have established that the contaminants found in the 12 KV cable jacket are not present in any of the 4 KV cables. NRC inspectors will continue to follow licensee actions and examine information as it becomes available. Followup item 50-275/93-03-01 and LER 50-275/93-05 therefore remain open.

b. Failure to Fully Close Containment Equipment Hatch During Fuel Handling Operations

During Unit 2 core offload operations on March 12, 1993, the senior reactor operator (SRO) conducting defueling operations observed an unusual bolting pattern on the containment equipment hatch. The operator checked if the hatch was sealed, and found that a 1/2-inch gap existed at the top of the hatch for about 90 degrees of arc. Defueling operations were suspended until the hatch was sealed. The licensee reported the failure to fully close the hatch in a four-hour non-emergency report to the NRC, and issued LER 50-323/93-03.

Licensee Investigation: The licensee investigated the root cause of the occurrence, and found that the "tailboard" (i.e., the discussion which takes place before a job is performed) had not been performed properly. The foreman conducting the tailboard had perceived a need for haste. He inadvertently addressed only the first two, rather than all three, of the applicable pages of the hatch closure procedure, MP M-45.1, "Containment Hatch Closure." The foreman had also conducted the tailboard before the individual who installed the bolts arrived. Neither the individuals attending the tailboard, nor the individual who installed the bolts, had been informed of the requirement to check from outside containment to determine if the hatch was sealed, since this requirement was listed on the third page of the procedure.

The licensee found that contributing factors in the failure to close the hatch were failure of the individual signing the work order verification to understand the content of the requirements in detail, and removal of the requirement for independent Quality Control verification. The Quality Control verification requirement had been removed based on several successful hatch closures in the past. Another contributor was that mechanics had been reluctant to climb the bolting equipment on the hatch perimeter to reach the bolts on the top of the hatch, since this did not provide secure footing.

Safety Significance: The licensee evaluated the potential release of radioactivity as a result of a fuel handling accident while the equipment hatch was improperly secured. As documented in Calculation RA-93-02 dated March 18, 1993, and in Action Request A0298388, the licensee determined that the effect of the 1/2 inch gap in the top of the equipment hatch would result in off-site doses less than the FSAR design basis doses for a fuel handling accident, and a dose increase of much less than 10 percent of the 10 CFR 100 limits for off-site exposure.

Licensee Corrective Actions: The licensee reemphasized to maintenance personnel the need to perform adequate tailboards, and encouraged personnel to have a questioning attitude such as that evidenced by the refueling SRO who identified the problem. A mandatory Quality Control verification for hatch closure was added to the procedure. The licensee also installed permanent ladders and a safety restraint system around the perimeter of the equipment hatch to provide for a higher level of personnel safety, and changed the procedure to require 12 vice 4 bolts to be secured for fuel handling operations.

NRC Action: The NRC considered this to be a violation of the Limiting Condition for Operation of Technical Specification 3.9.4, in that the requirements of licensee procedure M-45.1 to properly secure and check the hatch were not followed prior to irradiated fuel movement. This violation was not cited since the criteria of the NRC Enforcement Policy for non-cited violations (NCVs) were satisfied (NCV 50-323/93-07-01, Closed). LER 50-323/93-03, Revision 0 is closed based on evaluation and verification of the licensee's corrective actions.

One non-cited violation was noted.

5. Maintenance (62703)

The inspectors observed portions of, and reviewed records on, selected maintenance activities to assure compliance with approved procedures, Technical Specifications, and appropriate industry codes and standards. Furthermore, the inspectors verified that maintenance activities were performed by qualified personnel, in accordance with fire protection and housekeeping controls, and that replacement parts were appropriately certified. These activities included:

- Work Order Maintenance Procedure MP-M-7-7A, Reactor Core Barrel Lift
- Work Order C0113477, Replacement of Unit 2 Residual Heat Removal (RHR) Pump Safety Injection Timer
- Temporary Procedure TP-TD-9210, Emergency Diesel Generator (EDG) Response to Step Load Increases
- Post Modification Test 21.19, EDG 2-3, Acceptance of Design Plant Loads

- Work Order R0094999, Diesel Engine Pre-Lube Oil Pump Maintenance, Emergency Diesel 2-1, Maintenance Procedure MP-M21.10
- Work Order R0094300, Diesel Engine Generator Inspection (18 month interval), Emergency Diesel 2-1, Maintenance Procedure STP-M81-A
- Work Order R0098961, Diesel Engine Generator Inspection (54 month interval), Emergency Diesel 2-1, Maintenance Procedure STP-M81-C
- Work Order C0107133, Emergency Diesel 2-1, Maintenance Procedures:
  - DCP-113, Field Storage Areas
  - DCP-114, Solder Connections
  - DCP-301, Wire and Cable Installation
  - DCP-302, Electrical Equipment Installation
  - DCP-303, Splices, Repair, and Determinations

No violations or deviations were identified.

6. Surveillance (61726)

By direct observation and record review of selected surveillance testing, the inspectors checked compliance with TS requirements and plant procedures. The inspectors verified that test equipment was calibrated, and that test results met acceptance criteria or were appropriately dispositioned. These tests included:

- STP V-2X, Exercising Containment Hydrogen Purge Isolation Valves
- STP V-3T3, Exercising Containment Hydrogen Sample and Recombiner Valves
- STP V-11, Component Cooling Water Valves Response to Actuation Signal
- STP M-9A, Monthly EDG 1-Hour Surveillance
- STP M-15, Verification of Safety Injection Relay Timers
- STP M-13 B1, Verification of RHR Pump Safety Injection Timer

During the inspector's observation of Unit 2 integrated ECCS timing relay testing, a safety-related 120 V AC inverter, IY22, abruptly failed to produce output voltage. This failure occurred a second time during the testing, and appeared to have been the result of an AC input voltage spike, which caused the AC input breaker to trip on overvoltage. This inverter provides 120 V AC power to several safety-related component indications. The licensee was continuing to monitor and troubleshoot the failure (action request A0302482 and corrective maintenance order C113516), but had not been able to duplicate the failure as of the end of

the inspection period. Licensee troubleshooting indicated that the startup of a battery charger may have contributed to the inverter trip. This issue will be followed by NRC open item 50-323/93-07-02.

No violations or deviations were identified.

7. Observation of Licensed Operator Training (41701)

On April 1 and April 8, 1993, the inspectors observed licensed operator training in the simulator (Lesson LR927S3). The training addressed design basis events and security responses. Skills exercised and discussed included understanding of plant equipment configurations and accident responses, individual diagnostics skills, team communications, and team diagnostic skills. The lesson consisted of review of abnormal plant conditions, plant response to a non-standard turbine control configuration, a loss of coolant accident, and a security event. The exercise included a loss of offsite power, inadequate core cooling, and implementation of the emergency plan. The training included communications with security personnel involved in the training. Operator actions appeared appropriate and procedures were followed. The inspector also observed the licensee's critique of the simulator exercise, which appeared appropriately probing and critical.

No violations or deviations were identified.

8. Emergency Diesel Generator Testing (61726)

Diesel Generator Separation: During the 2R5 outage, the licensee completed installation of emergency diesel generator 2-3. The licensee performed Construction Test Procedure TP E-45581-02A, "DG 13 Control Panel Appendix R Modifications (DC1-EE-47600) & Separation of DG 1-3 from Unit 2 (DC1-EE-45581)," Revision 0, and Post Modification Test 21.18, "Diesel Generator 1-3 Operational Test Following Separation from Unit 2."

The inspectors reviewed these construction and post modification testing procedures for the separation of Unit 2 control and annunciation functions from emergency diesel generator 1-3 and for the connection of emergency diesel generator 2-3 to Unit 2. Separation of the diesel generators appeared to be adequate.

Inattention to Detail in Procedure Writing and Diesel Generator Testing: In performing Steps 9.21.2 and 9.21.3 of Construction Test Procedure TP E-45581-02A, the licensee's construction testing personnel did not follow the test procedure. Test personnel signed off the steps without completing the second action of the steps. The second action for Steps 9.21.2 and 9.21.3 was to place the 4 KV Diesel Generator 1-3 differential relay and loss of field relay switches, respectively, in the "Cut Out" position and to hang a "Caution Tag." At Step 9.21.18, the test directed shorting of the currents at the load side of the Diesel Generator 1-3 differential current test block. When this was performed during a test run, Diesel Generator 1-3 tripped. The licensee's followup investigation determined that the second action of each of the steps had not been performed, and that omission of the second action of Step 9.21.3 had resulted in the inadvertent trip.

The inspector reviewed the procedure steps. The steps were clear, and there was no apparent ambiguity as to step performance. The steps were a two-action step with one sign-off. The two-action step with one sign-off and the inattention of test personnel in following the procedure apparently led to the diesel generator trip. The licensee acknowledged the inspector's observation that each action should require one sign-off. Since the criteria of the NRC Enforcement Policy were satisfied, this violation of TP E-45581-02A was not cited. (NCV 50-275/93-07-03, Closed)

Inadequate Control of Foreign Material in Emergency Diesel Local Panel: On March 19, 1993, diesel generator 1-3 failed to start on a simulated safety injection signal (SIS) during performance of Post Modification Test 21.18. The licensee identified that failure of the diesel to start was due to a piece of wire insulation, left from the modification work, lodged in a contact of an open relay housing, JWPR1, in the diesel generator local panel (GQD 13). The licensee removed the insulation, and the test was completed satisfactorily.

Upon further visual inspection of Panel GQD 13, the licensee found another piece of wire insulation in contact 17 of relay JWPR1A. The insulation was removed immediately. The licensee initiated corrective action to alert personnel performing modification and maintenance work in the five emergency diesel local panels where relays with open housings were located and performed additional inspections of panels. Workers were then required to use containers to control insulation pieces removed from the wire.

One non-cited violation was noted.

9. Incorrect Dowel Dimension in Safety-Related Check Valves (62703)

During a routine inspection of Unit 2 check valves, the licensee identified that two dowel pins used to maintain relative spacing between the valve body and the disk hinge ring were too short to perform their function. The licensee also identified that the vendor's scale drawing (Anchor Darling 1027-3E) showed a proper dowel length (1.5 inches), but the dowel dimensions recorded in the material list (item 15) were too short (1.2 inches). A later revision of the drawing listed the correct dimension of the dowel.

Safety Significance: The function of this dowel pin is to maintain relative spacing between the valve body and the hinge assembly. During a seismic event, the shorter pin would allow the hinge ring, and thus the internals assembly, to rotate a small amount relative to the valve seat, potentially preventing the check valve from fully closing. The licensee also identified that the same inconsistency in dowel length documentation had occurred in a similar 4-inch valve (Anchor Darling Model 2787-5, drawing 1026-3E). Inspections identified other dimensional inconsistencies, such as hinge ring locking pin length and gasket groove dimensions outside of drawing tolerances.

The licensee identified six 8-inch valves in each unit, and eight 4-inch valves in each unit which were potentially affected. A total of 16 valves are installed in safety-related applications. These valves are

installed in the following systems: reactor·coolant, safety injection, chemical and volume control, containment spray and spent fuel pool systems.

To determine the potential adverse effects of a seismic event, the licensee reviewed the systems required to be used after a seismic event, and determined that misalignment of these particular valves as a result of a seismic event would not have adverse consequences, since these specific check valve installations would not be required to perform during or after a design basis earthquake.

Licensee Actions: The licensee performed routine inspections of all of the applicable Unit 2 valves during the outage, except for the valves in the spent fuel pool system, which cannot be cleared during an outage. The spent fuel pool valves in Unit 1 were inspected. The licensee identified that dowels were too short in two of the six valves inspected to date. The inspections of Unit 2 spent fuel pool valves will be performed after the outage.

The licensee performed an operability assessment (OE 93-04) of the valves which had not been inspected, and concluded that they were operable. Calculations showed that less than 1.0 G acceleration in the vertical direction would be experienced at the affected valve locations during a seismic event of 0.2 G acceleration. Less than 1.0 G acceleration in the vertical direction does not affect operability of the valves. Plant procedures require that the plant be placed in cold shutdown if a seismic event of 0.2 G or greater acceleration should occur, at which time an inspection of the affected valves would be performed. The design basis also does not require the licensee to consider a loss of coolant accident concurrent with an earthquake. The licensee will inspect all Unit 1 valves during the next scheduled outage, or during the next outage of sufficient duration.

Root Cause Assessment: The licensee identified that these valves had been manufactured by Anchor Darling in the 1970's, for Westinghouse. The dedication of these valves for nuclear service was performed at Westinghouse. Further investigation is ongoing.

NRC Action: The NRC will follow licensee actions to continue valve inspections and other actions. Further actions by the NRC will address areas such as the basis for 0.2 G acceleration as a threshold for inspection, the lack of current valve drawings, and the lack of identified inconsistencies during inspections performed in earlier outages. These actions will be followed by open item 50-323/93-07-04.

No violations or deviations were identified.

10. Cracking of Safety Injection Accumulator Cladding and Nozzles (62703)

During routine inservice inspection, the licensee identified that several nozzles and the stainless steel cladding on several weld seams of the ECCS accumulators evidenced cracking. The licensee replaced the nozzles which had cracks, and evaluated the cracking at the weld seams. The licensee performed non-destructive testing of the cracks, and determined

that many of the cracks had propagated through the cladding, but had stopped at the carbon-steel base material. The licensee identified that this type of cracking had been observed in the accumulators at the Watts Bar Nuclear Power Plant, reported in a Westinghouse 10 CFR Part 21 report, and was considered to be a symptom of the heat treatment method used by the vendor, Delta Southern. The licensee re-evaluated the design-basis of the accumulator and determined that the remaining vessel wall thickness, without including the cladding, exceeded code requirements and had a significant margin to safety. On April 22, 1993, PG&E received interim NRC acceptance to use the accumulators, without repairing the cracking. A concern of corrosion of the carbon steel base metal by boric acid still exists. This issue is still under review by the NRC, however, the NRC concurs with the licensee's evaluation that any degradation due to this type of corrosion would occur slowly. The licensee has agreed to monitor the cracks in accordance with the code, thereby providing assurance of safety by allowing the licensee to take action should any growth or corrosion occur.

The NRC noted that, as a basis for acceptance of the cracks without repair, the licensee had relied upon supplemental analysis performed pursuant to the ASME Code. The licensee did not initially recognize that this approach needed NRC approval before the accumulators were returned to service. This weakness in the licensee's program, along with the licensee's final resolution of this issue, will be reviewed during a future inspection. (Followup Item 93-07-05)

No violations or deviations were identified.

11. <u>Inadequate Weld of Feedwater Isolation Valve Disk Guide Rail (62703)</u>

During routine outage inspection, the licensee identified that one of the main feedwater isolation valves, FCV-438, an 18-inch gate valve, had been incorrectly fabricated. The rail which guides the disk had not been correctly welded on one side. On the lower end of one of the rails, the weld attachment to the valve was incomplete. On one side, about seven inches of the rail was not attached. The other side of the disk, the lower eight inches of the rail were not attached. The rail did not appear to have been displaced from the original installed position.

<u>Licensee Action:</u> The licensee inspected the other Unit 2 feedwater isolation valves and found that all rails were properly attached. The licensee had inspected all Unit 1 valves during the previous outage, and found no rails improperly welded. The licensee discussed the proposed repair of the valve with the vendor, and performed the repair.

<u>Root Cause Investigation:</u> The licensee and vendor were unable to determine the root cause of the incorrect welding. The licensee concluded that the welding involved error by the vendor during initial fabrication. The licensee concluded that the problem was not reportable in accordance with 10 CFR 21, but issued an INPO notification to alert the industry.

<u>Safety Evaluation:</u> The licensee determined that the valve would have performed its safety function. This conclusion was based on the previous performance of the valve, during surveillance tests which tested closure

times, and the lack of markings on the rails which would indicate misalignment during closure.

No violations or deviations were identified.

12. <u>Unit 2 Component Cooling Water (CCW) Heat Exchanger Tube Fretting (62703)</u>

During routine outage eddy current inspection, the licensee identified that several tubes in both CCW heat exchangers evidenced signs of fretting due to contact with the support plates in the upper area of the heat exchanger. Fretting was observed at several of the support plates along the lengths of the affected tubes. The depth of the damage varied, with the deepest damage about 25 percent of the tube wall thickness of 50 mils. The licensee removed one of the damaged tubes to perform a root cause assessment and to calibrate the inspection probe precisely to this type of damage. The licensee plugged all tubes (10) with damage greater than 20 percent though wall.

<u>Background:</u> Each unit has two YUBA CCW heat exchangers which have straight tubes, about 35 feet long. Auxiliary salt water (ASW) flows inside the 90-10 copper nickel tubes, with component cooling water (CCW) on the shell side. Six 5/8-inch thick support plates are spaced about every four feet. The 1,237 tubes in each heat exchanger are about one inch in diameter.

<u>Root Cause Evaluation:</u> The licensee's investigation concluded that flow-induced vibration had caused about 50 of the tubes in the upper portion of the heat exchanger to contact the support plate holes, resulting in circumferential wearing of the tube at the support plate locations. Since wear sites occur at several support plates, vibration had occurred along a significant fraction of the length of the affected tubes. Although investigation is continuing, the source of the vibration has not been determined. Two likely sources contributing to the amplitude of the vibration have been identified. First, flow perpendicular to the tubes occurs near the CCW inlet and outlet of the heat exchanger, located at either end of the top centerline of the vessel. Secondly, the inspector noted that flashing occurs at the top of the discharge head of the heat exchanger due to the combined affects of heat exhanger fouling, and the vertical drop of ASW effluent to the discharge structure. Therefore, flashing may occur inside the tubes in the top of the heat exchanger, in the general area of damage. This could significantly change the moment of inertia of these tubes, and thus affect vibration resonance. The licensee was evaluating these effects.

<u>Safety Significance:</u> The licensee stated that the plugging of tubes still allowed sufficient margin for the CCW heat exchanger to perform all normal operational and safety functions. The licensee also concluded that the fretting mechanism would not cause additional damage. During the last Unit 1 outage, no damage was observed in the Unit 1 CCW heat exchanger tubes. Unit 1 tubes will be inspected again during the next scheduled outage. The vendor, YUBA, stated that up to 40 percent wall thinning was acceptable, and stress calculations allow as much as 50 percent wall degradation. The vendor stated that there was a 2 percent margin in heat transfer coefficient, and that the current tube plugging

would not be significant to heat exchanger performance.

Inspector Concerns: The inspector examined the conclusions of a recent licensee analysis which concluded that there was a potential for overheating of the CCW system during a loss of coolant accident. A reduction of CCW system heat removal capability could be significant to this analysis. The licensee responded that the plugging of 10 tubes still left sufficient margin for heat removal, and that another 10 tubes could be plugged without concern for heat exchanger performance. Further examination of the assumptions of CCW heat removal analysis appears warranted. This issue will be followed by existing NRC Unresolved Item 50-275/92-16-04.

13. Missing Seismic Supports in Breaker Cabinet (40500)

During an audit of safety system outage modifications, a licensee Quality Assurance inspector noticed inconsistencies in structural supports while comparing the redundant 4KV vital switchgear. In response to this issue, the licensee conducted further investigation which concluded that the differences between the switchgear structural supports was not safety significant.

Quality Assurance Concerns: The Quality Assurance organization plans to perform a further evaluation of the differences between the switchgear, since the original seismic analysis was sufficiently complicated to require that one of the vital 4 KV switchgear be removed from the plant and seismically tested as a unit. Therefore, the Quality Assurance organization is continuing to evaluate the safety significance along with the licensee's line organization. The evaluation of the safety significance of the inconsistent structural supports will be followed by NRC open item 50-323/93-07-06.

No violations or deviations were identified.

14. Failure of Valve to Close Due To Intrusion of Water and Particulate (62703)

On March 12, during valve disassembly and inspection following an incomplete closure of the valve during a surveillance test, the licensee determined that the grease in the valve operator for AFW steam supply valve 2-FCV-37 contained water, particulate matter, and corrosion. This appeared to have been due to lack of installation of quad rings which protect the valve operator from intrusion of rain, spray, and particulate matter. In January 1993, when the valve failed to close during its surveillance test, the licensee's limited inspection concluded that the valve stem had been sticking, since later lubrication of the valve stem allowed the valve to pass its surveillance test. After inspection of the valve in March, during the Unit 2 outage, the licensee concluded that the valve may not have been able to function upon demand with full differential pressure across the disk, as required by NRC Generic Letter 89-10.

The licensee cleaned and reassembled the valve, and inspected two feedwater isolation valves nearby. The internals of 2-FCV-439 were also found to have heavy rust and moisture. Quality evaluation Q0010397 and

NCR DC2-93-EM-N014 were initiated to track the above valve issues. The frequency of stem lubrication was increased to quarterly for valves associated with the AFW pump steam supply.

Safety Significance: FCV-37, a remote manual containment isolation valve, is installed in the steam supply to the auxiliary feedwater system turbine driven pump. To perform its safety function, FCV-37 must be closed either manually or by remote actuation to isolate the contaminated steam source from the AFW turbine driven pump in the event of a steam generator tube rupture.

Regarding a possible steam line break, a Westinghouse analysis concluded that this valve is not required to isolate a downstream steam line break, since this type of line break does not initiate a plant trip, and main feedwater can be used to support plant operation until the break is isolated manually. Therefore, the licensee concluded that no adverse effect would have occurred in the event this valve failed to function with remote operation.

NRC action: The inspector will follow up on the following aspects of this occurrence:

a. The validity of the licensee's assumptions of habitability in the local area of FCV-37 and areas required to access FCV-37 in the event of a steam line break and required manual operation of FCV-37.

b. The length of time assumed during station blackout for unavailability of steam to the turbine driven AFW pump; specifically regarding assumption of a single failure of FCV-38 (the redundant isolation valve in the AFW pump steam supply) concurrent with mispositioning of FCV-37.

c. The scope and depth of corrective actions to ensure that components are appropriately identified and protected from weather effects, specifically regarding installation of valve quad rings which should exclude water and particulate matter from valve internals.

d. The safety significance of and corrective actions for rust and moisture in FCV-439.

e. The maintenance procedure steps which ensure appropriate protection of component internals from weather effects.

f. A more detailed examination of the reportability of the inoperable valve as a result of the failed surveillance test in January 1993.

These items will be tracked by NRC open item 50-323/93-07-07.

No violations or deviations were identified.

15. Spent Fuel Pool Activities (86700)

The inspector observed fuel handling operations in the spent fuel pools. The licensee appeared to have followed all Technical Specification and

administrative requirements, with the following exception. In Unit 2, doors were opened in response to a hydrazine spill, resulting in insufficient negative pressure for operability of the ventilation system. This was documented by LER 50-323/93-04, Revision 0. The licensee's corrective actions appeared appropriate.

No violations or deviations were identified.

16. Preparation for Refueling (60705)

To evaluate preparations for refueling, the inspector observed several refueling preparations and interviewed operators. The Technical Specifications and surveillance test procedure requirements appeared to have been met for the areas inspected, with one exception. Testing of the charcoal filters in the ventilation system was performed before fuel movement. However, the results of the analysis were not available before the refueling. The results of the analysis, received after refueling activities had begun, concluded that the efficiency of the charcoal in the operable ventilation train was slightly below the Technical Specification requirements. The licensee's safety analysis of a design basis fuel handling accident concluded that the adsorption capability of the charcoal was still sufficient to have mitigated an accident within design basis requirements. This conclusion appeared appropriate. This occurrence will be documented in a Licensee Event Report which has not yet been issued. The licensee's corrective actions, as of the end of this reporting period, appeared appropriate. This item will be followed with inspection activities to close the LER (Followup Item 50-323/93-07-08).

The inspector observed that the licensee's administrative requirements during refueling and outage operations for both the operability of plant systems, and availability of electrical power supplies, appeared to be more comprehensive and conservative than the requirements of the Technical Specifications. During the course of the defueling and refueling, the inspector observed that the licensee followed these more restrictive requirements.

No violations or deviations were identified.

17. Verification of Containment Integrity (61715)

As discussed in paragraph 4 above, the licensee identified that the containment equipment hatch had not been properly closed during refueling operations. The corrective actions and root causes were also discussed.

The inspector toured Unit 2 containment to evaluate whether containment closure was properly set. The inspector sampled several penetrations and hatches, which all appeared to have been properly controlled. The inspector also reviewed and inspected the licensee's implementation of surveillance procedure M-45B, Containment Integrity, which implements containment integrity requirements.

No violations or deviations were identified.

18. Review of Independent Safety Group Activities (40500)

The inspector reviewed routine actions by the independent safety groups and the QA organization. These included activities of the Plant Staff Review Committee (PSRC), Onsite Safety Review Group (OSRG), and Nuclear Safety Oversight Committee (NSOC), and QA audits and surveillances.

Based on this review, the inspector noted that the independent safety groups had made and documented several notable findings. Examples included the following:

- A member of the OSRG noticed that an isolation valve had been recently installed in the relief valve discharge for the positive displacement charging pump (PDP) in each unit, contrary to the ASME piping code. The licensee stated that this isolation valve had been installed recently as a design change in order to isolate the PDP, so that modifications to the PDP piping (to control piping vibration) could be made. The licensee requested a code exception, stating that the safety significance was negligible since the valve had been included in the controlled valve list and locked open during plant operation as part of the design change. On April 15, 1993, PG&E received interim NRC acceptance of the isolation valve installation, provided that the valve be administratively controlled in the locked open position when the PDP is operable.

- A QA inspector identified that the licensee planned to isolate compressed air and backup nitrogen during a pending containment integrated leak rate test (CILRT). This compressed air and nitrogen provide the motive force for the power operated relief valves (PORVs), required by the TS for low temperature overpressure (LTOP) protection of the reactor coolant system. Licensee management agreed that the finding was a concern, and performed the CILRT with the reactor coolant system vented through a blocked-open PORV.

Based on the reviews conducted, the inspector noted that the activities of the independent safety groups appeared appropriate and well directed. This inspection procedure remains open for additional inspection during future inspections.

No violations or deviations were identified.

19. Licensee Event Report (LER) Followup (90712, 92701)

The following LERs were reviewed and closed based on the licensee's root cause determination and corrective actions:

Unit 1: 92-007, Revision 0, Missed Fire Watch and Manual Engineered Safety Feature Actuation from Chemical Spill

92-009, Revision 0, Dose Limits Potentially Exceeded from Chemical and Volume Control System Valve Diaphragm Leakage Due to Thermally Induced Degradation

92-012, Revision 1, Single Failure Vulnerability of the Auxiliary Building Ventilation System, Resulting in Entry into Technical Specification 3.0.3

92-29, Revision 0, Noncompliance with Technical Specifications Due to Misinterpretation of Technical Specification Requirements

93-002, Revision 0, Containment Isolation Valve Not Isolated in Accordance with Technical Specification 3.6.3

Unit 2: 92-03, Revision 1, Unit Shutdown due to Inoperable Main Turbine Stop Valve

93-02, Revision 0, Entry into Technical Specification 3.0.3 due to Inadequate Work Instructions for Auxiliary Building Ventilation System

93-04, Revision 0, Fuel Handling Building Ventilation System Inoperable During Fuel Movement

Unit 2 Special Report 93-01, Steam Generator Tube Plugging.

No violations or deviations were identified.

20. Review of Open Items (92700)

Gas Bottle Secured to Diesel Air Start Piping (Open Item 50-275/93-03-02), Closed

During an earlier inspection, the inspector observed a compressed gas bottle which was secured to safety-related emergency diesel air start piping. The licensee's analysis determined that, during a design basis seismic event, the forces on the piping and supports would be less than 10% of code allowables. The inspector reviewed the calculations and determined that the calculations appeared reasonable. However, securing gas bottles to safety-related equipment is contrary to the requirements of the licensee's Administrative Procedure AP-763, Paragraph 5.4.7 and Technical Specifications 6.8.1. Since the criteria of the NRC Enforcement Policy were satisfied, this violation of NRC requirements was not cited (NCV 50-275/93-07-09, closed based on licensee corrective actions).

21. Exit Meeting

An exit meeting was conducted on April 21, 1993, with the licensee representatives identified in paragraph 1. The inspectors summarized the scope and findings of the inspection as described in this report.

The licensee did not identify as proprietary any of the materials reviewed by or discussed with the inspectors during this inspection.