U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos. 50-275/85-35, 50-323/85-33

Docket Nos. 50-275, 50-323

License Nos. DPR-80, DPR-81

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

Facility Name: Diablo Canyon Units 1 and 2

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

Inspection conducted: October 7-November 1, 1985

Inspector: J. F. Burdoin 11/22/85
J. F. Burdoin, Reactor Inspector Date Signed

Approved by: R. T. Dodds 11/22/85
R. T. Dodds, Chief, Reactor Project Section 1 Date Signed

Summary:

Inspection during period of October 7-November 1, 1985 (Report Nos. 50-275/85-35 and 50-323/85-33.

Areas Inspected: Unannounced inspection by one regional inspector of open items consisting of followup inspection items, Part 21 Reports, Generic Letters and IE Notices and followup of allegations by NRC contractors. Inspection procedures numbers 92700, 92701, 92704, and 92705 were used as guidance for the inspection. The inspection involved 78 inspection hours by one inspector and approximately 120 hours by contract personnel.

Results: No items of noncompliance or deviations were identified.

DETAILS

1. INDIVIDUALS CONTACTED

Pacific Gas and Electric Company (PG&E)

*R. C. Thornberry, Plant Manager
R. Paterson, Assistant Plant Manager, Plant Superintendent
G. R. Vincent, OC Inspector
J. R. Harris, QA Auditor
R. Johnson, Licensing Representative
D. R. Bell, QC Supervisor
N. M. Norm, Field Construction Manager
W. E. Coley, Lead Startup Engineer
J. R. Bratton, Rate Case Coordinator
*R. W. Taylor, QA Engineer
W. J. Kelly, Licensing Representative
C. E. Johnson, Fire Marshall
R. P. Kohovt, Emergency Safety Services Supervisor
*T. L. Grebel, Regulatory Compliance Supervisor
M. P. Hanrahan, Senior I&C Supervisor
*D. A. Taggart, Acting Director, Quality Support, QA
R. W. Cook, Rate Case Group Coordinator
J. D. Mc Clintock, Site Fire Protection Engineer

Various other engineering and QC personnel.

*Denotes attendees at exit management meeting on November 1, 1985.

In addition, NRC Resident Inspectors attended the exit management meeting.

2. AREA INSPECTION

An independent inspection was conducted in Units 1 and 2 auxiliary buildings. The equipment spaces inspected for both units included six battery rooms, charging pump areas, component cooling water pump areas, auxiliary feed pump areas, safety injection pump areas, and RHR pump areas (Unit 1 only). Only minor housekeeping problems such as debris on floor and some paper tags attached to equipment were found, everything appeared to be in order.

No violations of NRC requirements were identified.

3. FOLLOWUP ON PREVIOUS IDENTIFIED INSPECTION ITEMS

a. (Closed) Item 275/323/82-40-14 Heat Shields

A concern developed by an inspector for possible damage to pressurizer sensing lines LT 460/461/462 and PT 456/457/458A/458B resulting from the potential for ignition and burning of overflowed lube oil from the reactor coolant pump collection tanks.

This concern was addressed in Sections Nine of Supplemental Safety Evaluation Report (SSER) 23 for Unit 1 and SSER 31 for Unit 2. The inspector reviewed these SSERs and inspected the oil collection pans and automatic sprinkler systems at the RCPs. Everything appeared to be in order.

This item is closed for Units 1 and 2.

b. <u>(Closed) Item 275/83-39-05 Fire Barrier Penetration</u>

During walkdown of facility the inspector found two unsealed penetrations in areas containing safety-related equipment. The concern was that unsealed penetrations were not always entered into the licensee computer penetration tracking system.

The inspector examined the seals installed in two conduit penetration in the 4-B-1 area at the 85 foot elevation and found them to be sealed with silicone foam. The inspector reviewed the work order (E-21) under which the work was accomplished and noted the sign-off for completion of the work

The inspector examined administrative procedure C-113, which describes the requirements initiating proper tracking of unsealed penetrations, and reviewed the particular concern of the original inspector with the licensee. Clearance request forms are required to be filled out for unsealed penetrations which are the source of entry for computer penetration tracking system. There is, however, a slight delay from the time a clearance request form has been filed and the entry made in the computer. This may have been the source of the original inspector's concern. Everything appeared to be in order.

This item is closed.

c. <u>(Closed) Item 275/323/83-39-06 Fire Doors Exemption</u>

During a walkdown of the facility an inspector noted a number of fire doors remained unlabeled following an inspection by Underwriters Laboratory (UL) to qualify and label fire doors.

Concerning the fire doors left unlabeled by UL, the inspector was informed that the licensee had applied to NRR for an exemption on fire doors in their letter dated July 5, 1983. The request for the exemption was evaluated and approved, with certain modifications (options) identified in Sections 9 of SSER 23 for Unit 1 and SSER 31 for Unit 2.

The inspector examined the following doors to verify that they had been modified in accordance with SSER 23 and licensee letters DCL-84-185 and 259; B-17, B-18, B-348, B-364, B-503, B-508, B-511-2, B-560 and B-567. These doors appeared to have been modified as required.

This item is closed for Units 1 and 2.

d. (Closed) Item 275/85-05-06 Penetration Above Door B-21

During a walkdown of the facility, an inspector found a penetration seal above Door B-21, around the sprinkler pipe, which was no longer in place. The pyrocrete seal had apparently been damaged due to construction activity in the area. The inspector examined the repair to the seal of the pipe penetration in the field. The work order and QC completion sign-off were also examined. Everything appeared to be in order.

This Item is closed.

A second and independent aspect of this item was that the original inspector also observed a number of doors which failed to close and latch due to the ventilation flow through the door while in the open position. The failure of these doors to close and latch was not a violation, because they were in an area where the hourly fire tours are conducted. This aspect, fire door latching, will be reviewed during a future inspection (50-275/85-35-01).

4. PART 21 ITEMS

a. (Closed) 275/323/TY-85-06, Part 21 Fire Damper Closure Under Air Flow, Ruskin Fire Damper Model Nos IBD-21, B-23 and NIBD-23.

On November 6, 1984, the NRC received a 10 CFR 21 report from the Ruskin Manufacturing Company which indicated that Ruskin fire dampers of a type installed in the plant would not close under certain conditions as described, the fire dampers may not close because of:

1) the interference of conduit for the electro-thermal link on the vertical dampers, and

2) insufficiently strong "negator" springs on the horizontal dampers.

By letter dated January 29, 1985, the licensee stated that such dampers were being tested and committed to modify the dampers for Unit 2 by removing the conduits in both the vertical and horizontal dampers, and by providing new negator springs and modified locking mechanisms if modifications were deemed necessary.

This issue of Ruskin fire dampers was addressed in licensee's letter DCL-85-092 for Unit 1 and Section 9 of SSER 31 for Unit 2. The licensee issued DCN's; DC1-EH-29731 RO and DC2-EH-30731.RO for testing fire dampers and making the necessary modifications recommended by the manufacture.

The inspector reviewed these DCNs and the respective work requests BM-712 and BM-711. The inspector also examined the write-offs for the completion of the work requests and the QC inspections for the completed modifications. It appears that the appropriate

modifications have been completed satisfactorily for the Ruskin fire dampers.

This item is closed for Units 1 and 2.

b. (Closed) 275/323/85-15-P, Part 21 Technology for Energy Corporation (Tec) Value Flow Monitor Module, Model 914-1

TEC has found a quality deviation in three TEC Model 914-1 Valve Flow Monitor Modules which was reported to the Nuclear Regulatory Commission on July 18, 1985. This deviation results in a failure of the module to reset after indicating full flow through the valve. Note that the monitor does indicate properly when the valve is opened, thus performing its safety-related function. This quality deviation (Bar Graph "Latch-Up") is caused by a defective U5 (a Texas Instrument TL4900CN Analog Level Detector).

A list of the serial numbers of all affected TEC 914-1 modules was supplied. TEC recommended that all TEL 914-1 modules be tested to identify defective U5s. Modules with defective U5s should be replaced. In interim, if U5 latches up an operator can reset by cycling power off then on again.

The inspector reviewed the status of this item with the licensee. Their response to the Part 21 included initiating "Action Requests" which required testing the installed and spare 914-1 modules to verify the pass/fail status of these modules.

It was determined that replacement Modules (914-2) were required. The replacements were back ordered and have been received on site. The installation of these replacements will be scheduled during a plant shutdown of sufficient duration to allow retesting the systems following the installation of the new modules.

This item is closed for Units 1 and 2.

5. LICENSEE ACTION ON IE NOTICES

a. (Closed) IE Notice 85-49 Relay Calibration Problem

The notice alerted licensees to a significant error in the calibration of Agastat series E-7000 time-delay relays if calibrated in other than field mounted (vertical) position.

This item was reviewed with the licensee. The licensee has initiated action request (AR) A 0007451 revising Maintenance Procedure (MP) E-50.30 Rev. 4 to include the requirements that the Agastat relays be bench calibrated in the same positions as they are mounted in the field. This action request was reviewed and approved by the Plant Safety Review Committee September 12, 1985. The inspector verified the revision to the maintenance procedure.

This item is closed for Units 1 and 2.

[illegible]

[illegible]

[illegible]

[illegible]

b. (Closed) IE Notice 85-74 Station Battery Problems

The notice alerted licensees to problems that have occurred with lead-acid station batteries at several nuclear power plants. These problems concerned battery testing, charging, and standard operating practices.

The inspector reviewed this item with the licensee. The licensee demonstrated that the concerns of this notice have been fully addressed by the following procedures:

- STP M-11A, Measurement of Station Battery Pilot Cell Voltage and Specific Gravity (Revision 4)
- STP M-11B, Measurement of Station Battery Voltage and Specific Gravity (Revision 7)
- STP M-12A, Battery Performance Test (Revision 3)
- STP M-12C, Station Battery Service Test (Revision 2)
- OP J-9 IV, Placing a Battery on an Equalizing Charge (Revision2)
- OP J-9 II, Operating the Battery Chargers (Revision 4)
- MP E-55.3, Maintenance of Plant Storage Batteries and Racks (Revision 7)

The inspector reviewed the above procedures and verified that they do address the concerns of Notice 85-74. Also a related issue with regard to installation, operation, maintenance and measurement of pilot cell voltage/specific gravity for station batteries was addressed in inspection report 50-275/85-08.

This item is closed for Units 1 and 2.

6. LICENSEE EVENT REPORT FOLLOWUP (UNIT 2)

LER 85-008 Containment Ventilation Isolation

At 1628 PDT, September 13, 1985, while the Unit was in Mode 5 (cold shutdown), an automatic isolation of the Unit 2 containment Ventilation System (CVS) (JM) occurred. The CVS is an Engineered Safety Feature (ESF). All automatic closures responded as designed.

The containment ventilation isolation (CVI) was caused by a spurious spike in the gaseous radiation monitor (GRM) (IL) (MON) RM14A. The RM14A alarm was reset and the GRM isolation valves (VA) (ISV) were returned to their normally open position.

A second CVI occurred at 2044 PDT during additional switching in the 500 kV yard. The spurious spikes were initiated by electromagnetic signals generated during switching in the 500 kV yard.

Eight spurious or inadvertent Unit 1 and Unit 2 CVIs have occurred within a two month period. The other six events have been reported as Unit 1 LERs 85-023, 85-025 and 85-027, and Unit 2 LERs 85-001, 85-003, and 85-005.

The inspector conferred with the licensee to review the permanent solution to this repetitious occurrence. These recurring containment ventilation isolations appear to be caused by spurious spikes on the instrument system resulting from electromagnetic signals generated during switching in the 500 kV yard and from spikes generated during switching of gaseous radiation monitors (GRMs).

These spurious actuations were being studied by the newly formed noise reduction task force in an effort to determine the exact causes and effective corrective actions to prevent recurrence. Some of the immediate solutions presently being reviewed to reduce the noise (spikes) on the instrument system include: a) using crimped connectors versus solder connectors, b) replacing cables with low-noise cables, c) using low noise pre-amps. and d) providing a separate instrument grounding system. A long term approximately two years solution under serious consideration includes replacing the existing analog radiation monitoring system with a digital system. It appears that a permanent solution to the reptitious containment ventilation isolations was in process. Administrative procedure (AP) C-11 S2 has been revised to exclude reporting <u>expected</u> actuation of containment isolation during 500 kV system switching and spurious spiking during switching of the radiation monitoring systems.

These LER's are considered closed.

7. <u>IE TEMPORARY INSTRUCTION 2512/12</u>

This temporary instruction requested that Regional inspectors determine the quality of construction activities performed by Reactor Controls Incorporated.

This item was reviewed with the licensee; and it was determined that no construction work was performed by this company at the Diablo Canyon Power Plant.

This item is closed for Unit 2.

8. <u>FOLLOWUP OF VARIOUS ALLEGATIONS PERTAINING TO PLANT CONSTRUCTION</u>

<u>Note:</u> The allegation characterization statements contained in this report are either a paraphrasing of the staff's understanding of the allegers concern or statements taken from the allegation source document. The characterization statements do not represent a staff assessment, conclusion or position.

a. Task: Allegation or Concern No. 572 (ATS No: RV-84-A-052)

1) Characterization

PG&E has stifled Pullman Power Power Products (PPP) inspectors reports of faulty Bosten-Bergen and American Bridge Welds, by directing them not to issue discrepancy reports.

2) Implied Significance to Design, Construction, or Operation

Faulty welds in components received from vendors, if uncorrected, could possibly result in failure of safety-related systems.

3) Assessment of Safety Significance:

Staff review of source document, GAP 2/2/84 letter, #102-104 indicates that PPP inspectors were issued memoranda to stop issuing Discrepancy Reports (DRs) on "shop" welds. The terminology "shop" is used to describe purchased components that were welded by outside vendors. The allegation cites a PG&E memo dated April 3, 1980, authorized by Marvin (SIC) Leppke as a basis for the conclusion that PG&E stifled PPP inspectors. The allegation further states that, in 1982, PG&E instructed Pullman to delete those welds from the formal walkdown program.

The NRC staff reviewed the April 3, 1980 memo issued by Mr. M. R. Leppke. The staff was unable to find any direction that PPP should stop issuing discrepancy reports on shop welds. The context of this letter indicates that an extensive program had been underway to investigate, evaluate and repair rupture restraint welds. The letter states that sufficient shop weld data had been obtained to allow the engineering department to review the data and include its conclusions in the final rupture restraint report.

4) Conclusions and Staff Positon

The allegation that PG&E stifled attempts by PPP inspectors to report and correct faulty vendor welds cannot be substantiated.

5) Action Required:

None.

b. Task: Allegation or Condern No. 993 (ATS No: RV-84-A-076)

1) Characterization

(Similar to allegations 353 through 359 for field welds 197-212, concerning welding QC at Diablo Canyon.) It was alleged that several college students working as QC Inspectors

were unqualified. It was further alleged that weld procedure code 200 was "grossly inadequate".

(2) Implied Signifiance to Design, Safety or Construction

Unqualified inspectors or inadequate weld procedures could possibly result in unsatisfactory construction of safety-related systems or components.

3) Assessment of Safety Significance

The source documents were reviewed. The allegers testimoney only provided generalities such as "the quality controls at Diablo Canyon leave a lot to be desired" and"it is a real horrendous mess". This document lacks the substance or details necessary to conduct a viable investigation.

Subsequent testimony identifies three individuals who were allegedly not qualified to be inspectors. The staff researched Pullman Power Products Company training and certification records for these three inspectors. Their records indicate that these inspectors held certificates of qualification that were issued upon completion of training and successfully passed written examinations for nuclear pipe welding and visual inspection.

The testimoney further alleges that another indivdual wrote welding procedures for Pullman Power Products Company and that "in no way was he qualified for this task". A search of Pullman Power Products Company records failed to identify this person as ever having been employed at Diablo Canyon. The only person with the same name employed at Diablo Canyon that was even remotely connected with welding was qualified as a visual welding inspector.

It was also alleged that the quality of work performed by the production workers at Diablo Canyon was "really poor". This allegation was made by another alleger, whose allegation was resolved on allegation no. 1543, ATS No. RV 84A114.

A May 7, 1984 letter from another alleger to the NRC expresses opinions that weld procedure code 200, Specification P12B-P1-K1-4F-SMAW-6G was "grossly inadequate', that failure analysis conclusions were incomplete, that "significant factors" were not addressed, that weld procedure code 200 "indicated a basic misunderstanding of Preheat theory" and that radiographic examination used by itself was not adequate to verify weld integrity. This allegation appears to be very similar to earlier allegations 353-359 concerning weld procedure code 200 for field welds 197-212. It was concluded previously that weld procedure code 200 and specification P12B-P1-K1-4F-SMAW-6G were written, qualified and approved in accordance with the ASME Code, Section IX, Paragraph QW 200.2,

1983 edition, and Q-10 in the 1971 edition, which is the accepted standard for use in the nuclear industry.

4) Conclusion and Staff Position:

Review of Pullman Power Products Company records indicate that inspectors were indeed trained, tested and certified before they were permitted to perform weld inspection. This conclusion is further supported by the results of allegations 995 and 378, which are also concerned with inspector training and qualification. The allegations can not be substantiated.

5) Action Required:

None.

c. Task: Allegation or Concern No. 1009 (ATS No: RV-84-A-064)

1) Characterization

An individual believes that engineers who questioned suspect assumptions were transferred to Unit 2. Cooperative engineers plus new recruits were assigned to Unit 1.

2) Implied Significance to Design, Construction and Operation

Assignment of engineers based upon attitudes could result in a insufficient level of experience in the groups assigned to one of the units. An inadequate experience level could result in failure to detect safety significant design or installation errors.

3) Assessment of Safety Significance

Review of previous, similar allegations by the same individual led to the conclusion that the allegation pertains to the split up of the Onsite Project Engineering Group (OPEG) small bore pipe support group that occurred in January 1983. This reorganization created Unit 1 and Unit 2 areas within the small bore pipe support group and divided each area in to three squads. Previously there had been no subdivisions within the group.

The reasons for this reorganization and the basis for the individual assignments to the two groups were discussed with the individuals who were, at the time, the assistant onsite project engineer and the small bore pipe support group supervisor. They both indicated that, at the time of the reorganization, a consistent increase in work load was being experienced due to the fact that both units were entering a construction phase that entailed a large amount of small bore pipe support work. The group was reorganized to provide for better management of the increased staffing levels necessary to support the required level of effort. The assignments to the

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

two groups were made by the group supervisor in conjunction with the Unit 1 and Unit 2 area leaders plus the squad leaders who were onsite at the time.

Discussions with the individual who was the group supervisor indicated that roughly the same experience level was required of all group members, however, there were some individuals in the group who had been employed at the Diablo Canyon Project for longer than others, thus these individuals were more familiar with project procedures and personnel. It was his intention that the reorganization result in each squad having a few individuals with Diablo Canyon Project experience.

To determine if the ultimate squad composition reflected this intent, the composition of each squad was reviewed to identify those individuals who had been with the small bore pipe support group since its inception. Each of the squads, both in the Unit 1 and Unit 2 areas, had between two and three engineers who met this criterion.

4) Conclusions and Staff Position

All of the engineers in both the Unit 1 and Unit 2 small bore pipe support areas were required to have basically the same level of industry experience. Review of the individuals assigned in each of the area indicated that the engineers with the most Diablo Canyon experience were divided roughly equally between the Unit 1 and Unit 2 areas. Therefore, no safety significance can be attributed to this allegation.

5) Action Required

None.

d. Task: Allegation or Concern No. 1399 (ATS No. RV-84-A-073)

1) Characterization

Base plates of support members on Unit 2 RHR containment sump recirculating lines to RHR Pump have partial penetration weldments.

2) Implied Significance to Design, Construction or Operation

The implied safety significance is that base plates improperly welded may not support the designed loading.

3) Assessment of Safety Significance

The staff, after investigation, identified two pipe supports which have base plates with sections added and welded as described by the allegation. These were Hangers 22-11R and 413-76R.

Hanger 22-11R & 413-76R are similar in which a 3/8" plate was located between two other base plates and welded with a square groove weld joint, requiring full penetration. The PPP as-built was generated to modify support per Stress Analysis No. G-003-04, Rev. 1 by Gathersburg Power Division (GPD). This new work consisted of gusset plates and other non-relevent modifications.

A field inspection substantiated the allegation. One hanger had the weld wrapping around the plate making it difficult to determine weld type but the other showed a partial penetration weldment. PG&E engineering staff reanalyzed the hangers leaving out the welds and found them to meet the safety factor requirements. Therefore, in these two isolated cases no structural safety significance exists.

The inspector in preparing a FIR (Field Information Request) for GPD mistakenly called out the wrong weld type. To verify that was an isolated case, all the inspectors records were reexamined. The records indicated this individual was a cognizant inspector and aware of details. The individual was at Diablo Canyon for approximately one year. In that time, he performed approximately 450 inspections of which only 22 were of a similar type penetration weld and only 7 of the 22 were structural in nature. These were reinspected and the welds were correctly called out.

4) Staff Position

The situation in which a wrong weld was called appears to be an isolated case of no structural safety significance.

5) Action Required

None.

e. Task: Allegation or Concern No. 1485 (ATS No: RV-84-A-113)

1) Characterization

Unistrut and Thunderbird clamps used to support hydrogen gas tubing do not provide adequate support.

2) Implied Significance to Design, Construction and Operation

Inadequately supported hydrogen lines would present a potential fire hazard if the lack of support resulted in line rupture under conditions that imposed abnormal loads on the tubing (e.g. turbine trip).

3) Assessment of Safety Significance

The hydrogen lines for Unit 2 were inspected and it was confirmed that Unistrut P2026 clamps have been used in some

places to support 1/2 inch pipe and Thunderbird saddle clamps have been used in some places to support instrument tubing. Discussions with PG&E engineering indicated that they have determined that accelerations on the order of 14g would be required to load the Unistrut clamps beyond their design load. This far exceeds the maximum loading that could be imposed by an credible source of dynamic loading. PG&E indicated that similar results would be expected for the Thunderbird clamp applications. Independent assessment of the loads required to exceed design strength of the clamps concluded that PG&E's value of 14 g was conservative.

4) Conclusions and Staff Position

The Thunderbird and Unistrut clamps being used to support main generator hydrogen lines will withstand significantly greater loads than they will ever be expected to experience. Therefore, this allegation has no safety significance.

5) Action Required

None.

f. Task: Allegation or Concern No. 1493 (ATS No: RV-84-A-114)

1) Characterization

Pullman used pipe welding procedures on structural steel and when the problem was identified Pullman wrote a memo which revised the ESD which legalized the existing practice.

2) Implied Significance to Design, Construction or Operation

Welds made utilizing the wrong weld procedure may not meet the designed load.

3) Assessment of Safety Significance:

The alleger stated that full penetration single bevel welds performed on structural steel utilized a 37-1/2° bevel which was feasible for pipe. When this was brought to the attention of Pullman Q.A. a memo was written to revise the weld procedure.

ASME Section IX, Welding and Brazing Qualifications, does not consider a change on weld joint angle for SMAW or GTAW as an essential variable, therefore, requalification of the weld procedure if an angle change is made is not required.

4) Staff Position:

The staff concludes that the weld joint angle change made by Pullman Q.A. was performed correctly as per the governing Code.

5) Action Required:

None.

9. EXIT MEETINGS

The inspector conducted exit meetings on October 11 (an interim meeting) and November 1 with the Plant Manager, Plant Superintendent, and other members of the plant staff. During these meetings, the inspector summarized the scope of the inspection activities and reviewed the inspection findings as described in the report. The licensee acknowledged the concerns identified in the report.