

U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos: 50-275/91-24 and 50-323/91-24

Docket Nos: 50-275 and 50-323

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 Units 1 and 2

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

Inspection Conducted: July 21 through September 1, 1991

Inspectors: P. P. Narbut, Senior Resident Inspector
M. Miller, Resident Inspector
A. Hon, Resident Inspector, San Onofre

Approved by:

P. J. Morrill
P. J. Morrill, Chief, Reactor Projects Section I

9/20/91
Date Signed

Summary:

Inspection from July 21 through September 1, 1991 (Report Nos. 50-275/91-24 and 50-323/91-24)

Areas Inspected: The inspection included routine inspections of plant operations, maintenance and surveillance activities, follow-up of onsite events, open items, and licensee event reports (LERs), as well as selected independent inspection activities. Inspection Procedures 30703, 61726, 62703, 64704, 71707, 71710, 90712, 92700, and 93702 were used as guidance during this inspection.

Safety Issues Management System (SIMS) Items: None

Results:

General Conclusions on Strength and Weaknesses:

The licensee management and operators demonstrated a cautious approach to problems which developed in the Unit 2 charging and letdown systems. The reactor coolant system leakage was carefully monitored on an increased frequency and potential effects on system functions were carefully assessed and fully documented. The unit was shutdown before Technical Specification leakage limits were reached based on trends and pre-established limits.



Unit 2 broke a world record in achieving a continuous period of 482 days at power.

The licensee has not yet issued a procedure to require that rationale be documented for complex operability decisions. The licensee has been carrying this commitment forward since early 1991. The need for documentation of operability was one of the issues associated with a failure to call a main steam valve inoperable after it was set with a faulty test device. The licensee's rationale for initially declaring the valve to be operable was not recorded and proved to be incorrect by a subsequent retest.

Significant Safety Matters: None.

Summary of Violations and Deviations:

One non-cited violation was identified.

Open Items Summary:

No items were closed. Three items were opened.



bcc w/enclosure:
 docket file
 G. Cook
 B. Faulkenberry
 J. Martin
 R. Zimmerman
 K. Perkins
 Project Inspector
 Resident Inspector

bcc w/o enclosure:
 J. Zollicoffer
 M. Smith
 J. Bianchi

REGION <i>PJM</i> V/dot	<i>PJM</i> M Miller	<i>PJM</i> A Hon	<i>PJM</i> PJMorrill	<i>SAR</i> SARichards
PNarbut 9/20/91	9/22/91	9/22/91	9/22/91	9/25/91

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DETAILS

1. Persons Contacted

- *J. D. Townsend, Vice President, Nuclear Power Generation & Plant Manager
Diablo Canyon Power Plant
- *D. B. Miklush, Manager, Operations Services
- *M. J. Angus, Manager, Technical Services
- *B. W. Giffin, Manager, Maintenance Services
- *W. G. Crockett, Instrumentation and Controls Director
- D. H. Oatley, Manager, Support Services
- W. D. Barkhuff, Quality Control Director
- R. Powers, Mechanical Maintenance Director
- *D. A. Taggart, Director Quality Support
- *T. L. Grebel, Regulatory Compliance Supervisor
- *H. J. Phillips, Electrical Maintenance Director
- J. S. Bard, Work Planning Director
- *J. A. Shoulders, Onsite Project Engineering Group Manager
- M. G. Burgess, System Engineering Director
- *S. R. Fridley, Operations Director
- R. Gray, Radiation Protection Director
- J. J. Griffin, Senior Engineer Regulatory Compliance
- R. W. Hess, Assistant Onsite Project Engineer
- R. P. Flohaug, Senior Quality Assurance Supervisor
- *J. B. Hoch, Manager, Nuclear Safety and Regulatory Affairs

The inspectors interviewed several other licensee employees including shift foremen (SFM), reactor and auxiliary operators, maintenance personnel, plant technicians and engineers, quality assurance personnel, and general construction/startup personnel.

*Denotes those attending the exit interview on September 5, 1991.

2. Operational Status of Diablo Canyon Units 1 and 2

The reporting period began with both units at 100% power. Unit 1 was at power for the entire period. Unit 2 was at power during the majority of the reporting period but commenced a power coast down towards the end of the period and then shutdown for its fourth refueling outage on August 31, 1991.

Unit 2 shutdown about one week earlier than the scheduled shutdown date of September 9, 1991. The early shutdown was due to an unisolable leak in the charging system which had been monitored since it first became apparent on August 13, 1991. The leak rate increased on August 31, 1991, and licensee management decided to shut down before the technical specification limit of 1 gpm unidentified leakage was achieved.

Unit 2's shutdown on August 31, 1991, occurred after 482 days of continuous operation at power. This broke the world record for continuous operation of nuclear reactors.



The NRC Chairman visited the site on August 26, 1991. Additionally, a management meeting was held at the site on August 27, 1991, to discuss the results of the NRC's Systematic Assessment of Licensee Performance (SALP) for Diablo Canyon.

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 Operations (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 if regulatory requirements were satisfied. Shift turnovers were observed on a sample basis to verify that all pertinent information of plant status was relayed to the oncoming crew. During each week, the inspectors toured the accessible areas of the facility to observe the following:

- (a) General plant and equipment conditions.
- (b) Fire hazards and fire fighting equipment.
- (c) Conduct of selected activities for compliance with the licensee's administrative controls and approved procedures.
- (d) Interiors of electrical and control panels.
- (e) Plant housekeeping and cleanliness.
- (f) Engineered safety feature equipment alignment and conditions.
- (g) Storage of pressurized gas bottles.

The inspectors talked with operators in the control room 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 radiological conditions and/or challenges. ALARA considerations were found to be an integral part of each RWP (Radiation Work Permit).

c. Physical Security (71707)

Security activities were observed for conformance with regulatory requirements, implementation of 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. Operability Evaluation (OE) Review

The inspector reviewed licensee operability evaluation OE 91-02R0, "Unit 1 Containment Fan Cooler Units with Degraded Class II Exhaust Ductwork," which addressed the impact of the degraded conditions on Unit 1 operation. The licensee believed the cause of the ductwork degradation was due to vibration caused by high air flow. The air flow was increased in 1987 to compensate for a temperature increase which was due to dust and moisture fouling of the heat exchanger. The heat exchanger was cleaned in 1991, but the air flow dampers were not readjusted to decrease the air flow. The inspectors noted that the licensee had not documented an evaluation of the impact of the dust on the heat removal capability during accident conditions prior to the 1991 heat exchanger cleaning. The licensee committed to perform an evaluation of operability prior to the end of 1991. Furthermore, the inspectors noted the licensee did not have a procedure describing how to prepare an OE. The licensee stated that OEs were a new concept and that this OE was only the second to be written. The licensee stated that a procedure was being developed for preparation of OEs and Justifications for Continued Operation (JCOs). The 10 CFR 50.59 and NSAC 125 methodology for evaluations would be included in the procedure.

e. Observation of Operation Rounds (71707)

The inspector observed daily and weekly operator actions for the main control board walkdown, turbine building, and plant electrical equipment rounds. The operators appeared knowledgeable in performance of these rounds, and status of observed plant equipment appeared to have been controlled appropriately.

f. Fire Hazards Concerns (64704)

On August 11, 1991, the inspector and three control room staff members observed flammables stored in temporary trailers in a manner inconsistent with prudent practice. For example, open buckets of paint brush cleaner were observed. On August 13, and subsequent days, the licensee conducted fire prevention inspections of other temporary trailers and issued several action requests. In addition, due to the broad scope of fire hazard concerns, NCR DCO-91-SS-N071 was issued on August 27, 1991. Because of ongoing licensee actions,



this concern will be followed in the normal course of future inspections.

No violations or deviations were identified.

4. Onsite Event Follow-up (93702)

a. Flooding Caused by Failure to Follow Work Order Instructions

On August 2, 1991, flooding of the component cooling water (CCW) heat exchanger area occurred as a result of maintenance workers' failure to follow clearance instructions. The work order provided for cleaning the shell (seawater) side of a CCW heat exchanger. Clearance instructions required that the auxiliary sea water (ASW) inlet valve to the shell side be gagged shut before opening the shell side manway of the heat exchanger. Instead, maintenance workers removed the manway cover with the inlet valve shut but not gagged. Subsequently, during the attempt to gag the valve, the valve opened. This caused flooding of the heat exchanger area through the manway. This flooding would have presented a significant personnel safety hazard had an individual still been in the shell side of the heat exchanger when the inlet valve opened.

Control room operators saw the indication of a change of valve position, promptly isolated the source of flooding, and sent personnel to investigate.

The failure to follow clearance instructions appeared to be a violation of Technical Specification 6.8.1, which requires that work shall be implemented by procedures. This issue will be addressed as Unresolved Item 50-275/91-24-01, pending followup of the safety significance of the event and the licensee's corrective actions.

b. Unit 2 Charging and Letdown Problems

On August 13, 1991, after valve testing in the Chemical and Volume Control System (CVCS), the unit developed a reactor coolant system unidentified leakage rate in excess of Technical Specification limits. Also, the letdown system relief valve had lifted and would not reset.

Near the end of July 1991, Unit 2 Reactor Cooling System (RCS) unidentified leakage had increased from a normal rate of about 0.1 gpm to about 0.4 gpm. The unidentified leakage steadied out at this rate until August 13 when the leakage rate was 1.91 gpm, which is in excess of the Technical Specification limit of 1 gpm. The licensee declared an Unusual Event as a result of exceeding the technical specification limit.

Personnel entered the containment and found the leak to be near the check valves on the normal charging line. Upon isolation of the charging line the leak rate reduced to 0.47 gpm, and the licensee terminated the Unusual Event.



As a result of the unseated letdown system relief valve, the licensee isolated the normal letdown system and placed the excess letdown system in operation. Subsequently, as a result of the charging system leak, the normal charging line was isolated and the alternate charging line was placed in operation.

The licensee initiated enhanced and more frequent RCS leakage monitoring after August 13, 1991.

On August 29, the excess letdown flow rate suddenly increased from a normal rate of 25 gpm to 47 gpm. The valve (HCV-123) which controls excess letdown flow, and normally operates at 100% open, was closed to 40% open, and excess letdown flow reduced to 37 gpm. The licensee performed radiography on HCV 123, but could not determine the cause of the increased flow. Analysis concluded that the loss of the lower part of the needle valve (HCV-123) would explain both the increased flow rate and the change in normal valve position. Valve disassembly during the outage validated this conclusion.

On August 31, the RCS unidentified leak rate increased to 0.9 gpm. The licensee then shut the unit down and commenced the refueling outage, originally scheduled to start on September 9, 1991.

On September 1, 1991, after Unit 2 containment entry and removal of insulation, the licensee found that the check valve closest to the reactor coolant system on the normal charging line had a body to bonnet gasket leak and that 2 of 12 bonnet bolts were severed, presumably by steam and/or boric acid corrosion.

Also, the licensee found evidence of gasket leakage, a severed bolt, and several corroded bolts on the similar check valve of the alternate charging line.

Analysis

These facts indicate the leakage was non-isolable but did not represent a "pressure boundary leak" since the leak was from a gasketed joint. Technical Specification definitions state that pressure boundary leakage, which is prohibited, is non-isolable leakage through a pipe or vessel wall. A gasket leak is not considered a through wall leak.

The licensee initiated accelerated monitoring of the leakage when it occurred. Further, the licensee satisfactorily assessed continued operability of the unit in JCO 91-05R0.

The licensee's engineered safety feature equipment was not affected by the charging and letdown problems. The various safety injection flow paths for high, intermediate, and low pressure injection use different piping.



The licensee initiated a root cause investigation and plans for a careful controlled disassembly and parts analysis. Nonconformance Report DC2-91-MM-N069 was written to track these actions.

The licensee also assessed adjunct effects such as plant chemistry changes and reactor coolant activity level increases as a result of isolating the letdown system.

Overall, the inspectors considered that the licensee's actions in reaction to the series of occurrences on the charging and letdown systems were conservative, carefully considered, and adequately executed. Upper licensee management was involved in the decision making and was promoting and exercising a cautious approach.

c. Determination of Main Steam Safety Valve Operability

On August 26, 1991, the licensee started a surveillance test of Unit 2 main steam safety valves. Valve RV-60 was found slightly outside the setpoint acceptance criteria and was adjusted by about 3 flats (per the licensee engineers, each flat nominally represents a setpoint change of 10 psi). The second valve tested, RV-225, was found significantly outside acceptance criteria and was adjusted by 13 flats. Because this amount of adjustment was unusual, the licensee checked the calibration of the test device and found it to be significantly out-of-calibration. A replacement test device could not be obtained for approximately 24 hours. The licensee considered that valve RV-225 was inoperable since the test equipment was found out-of-calibration, and the valve had been reset by 13 flats (roughly 130 psi). However, the licensee decided that valve RV-60 was operable. The licensee stated that this determination was reasonable, based on the valve only being adjusted 3 flats (roughly 30 psi), which is typical for recalibration of this type of valve.

The licensee considered that only one valve was inoperable and promptly entered the applicable Technical Specification action statement to reduce reactor power and reset the high neutron flux trip setpoints to 87% power.

The inspector questioned why both valves were not considered inoperable, since both valves had been reset by test equipment of indeterminate condition. The licensee management stated that RV-60 was considered to be operable based on the fact that a normal amount of adjustment had been required, and the setpoint had been raised upward and would have been in the band of the graduated setpoints of the 20 main steam safety valves.

On August 27, the licensee received calibrated test equipment and first reset valve RV-225 because it had been adjusted the greatest amount and therefore was more significant to plant safety. The inspector considered this prudent. The valve was found to be out-of-specification and had to be reset by approximately 11 flats in the reverse direction.



Upon testing valve RV-60, which had been judged operable, it lifted at 1105 psig and at 1109 psig. The upper technical specification limit for this valve was 1101 psig. The licensee acknowledged the valve had been outside its technical specification limit by an average of 6 psig, reset the valve to within the limits, and is preparing a licensee event report based on the valve being outside technical specification limits without compliance with the appropriate limiting condition for operation. Valve RV-60 being set above its upper technical specification limit of 1101 psig is considered a potential violation of technical specification requirements. The violation is not being cited because the criteria specified in Section V.A. of the Enforcement Policy were satisfied. The safety significance of 1 valve out of 20 being set 6 psig above its setpoint is considered minimal (Item 50-323/91-24-02).

At the exit interview, the inspector discussed the occurrence with plant management. The inspector recognized the minimal safety significance of the occurrence. The inspector pointed out, however, that the decision to call RV-60 operable was non-conservative and appeared not technically justifiable.

Secondly, the inspector pointed out that even though management was consulted in this case (and concurred), the rationale for the operability was not documented. The failure to document the rationale for operability has been the subject of continuing dialog with plant management. Management has continued to respond that the procedure for documentation of operability is being worked on.

d. NRC Chairman Visits Diablo

On August 26, 1991, Chairman Selin visited and toured the site. At the end of the tour, the Chairman met with the press and representatives of the San Luis Obispo Mothers for Peace.

e. Unit 2 Commenced its Fourth Refueling Outage

On August 31, Unit 2 shut down and began its fourth refueling outage. The unit shutdown earlier than the scheduled date of September 9, 1991, due to an increasing unidentified RCS leak rate. The outage is currently scheduled to last 53 days.

f. Reportable Event due to Technical Specification Violations

On September 1, 1991, with Unit 2 in Mode 4, operators removed electrical power from the containment sump isolation valves and from the containment spray pumps. Both of the items are required to be operable in Mode 4 per technical specification requirements. Preliminary causes established by the licensee indicate that the containment sump isolation valves were depowered due to a procedural error and the pumps were depowered by an operator's failure to follow procedures.



The licensee discovered the situation during control board review, restored the power within 15 minutes of discovery, and properly reported the event.

The inspectors will followup the circumstances of the potential violation of technical specifications (Unresolved item 50-323/91-24-03).

No violations or deviations were identified.

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 maintenance activities were performed by qualified personnel, in accordance with fire protection and housekeeping controls, and replacement parts were appropriately certified. Specifically, maintenance activities associated with ASW heat exchanger cleaning and containment fan cooler heat exchangers were examined. No issues were identified with the exception of the apparent failure to follow work order instructions discussed in paragraph 4.a above.

No violations or deviations were identified.

6. Surveillance (61726)

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

The inspector observed testing in Unit 2 of Surveillance Test Procedure (STP) R-1A "Exercising Full-length Control Rods" and STP M-16P1 "Continuity Testing of Train A/B Slave Relays."

M-16P1 involved an operator reading the STP and another operator manipulating test switches and verifying appropriate status lights. This two-person test team provided concurrent verification. The system engineer was also in the vicinity to provide technical guidance. The inspector observed that occasionally, the procedure reader did not closely look at the other operator's actions but relied on the verbal confirmations. At the debriefing meeting, the licensee management acknowledged the inspector's observation and stated that the need to strengthen "concurrent verification" had been recognized, and an operator training program was being developed.

Additionally, surveillance test activities associated with main steam safety valve testing and primary coolant leak rate surveillances were assessed.

No violations or deviations were identified.



7. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations, or deviations. Unresolved items disclosed during this inspection are discussed in paragraphs 4.a. and 4.f. of this report.

8. ESF Walkdown (71710)

The inspector examined the Unit 1 steam driven AFW system. Procedure OP D-1.I, Attachment 9.2: "Auxiliary Feedwater System - Alignment Verification Checklist," and applicable drawings were used. No misalignments were found. The inspector noted that the lighting levels in the pump rooms were low. The licensee acknowledged the observation and stated that a program was already in place to improve the lighting conditions in the plant.

9. Exit (30703)

On September 5, 1991, an exit meeting was conducted with the licensee's representatives identified in paragraph 1. The inspectors summarized the scope and findings of the inspection as described in this report.

