Diablo Canyon 2
3Q/2003 Plant Inspection Findings

Initiating Events

Significance: Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement Correct Revision of Procedure Results in Power-Operated Relief Valve Opening
A self-revealing, NCV of Technical Specification 5.4.1.a was identified for the failure to use the latest revision of a surveillance procedure. This finding resulted in pressurizer power-operated relief Valve RCS-2-PCV-456, opening during a channel operability test. Maintenance personnel failed to verify the correct procedure revision was being used prior to performing work.

The finding is greater than minor because it had an actual impact of opening the pressurized power-operated relief valve, which is a precursor to a nonsignificant event (i.e., relief valve stuck open). Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Screening Worksheet, this finding is considered a primary system loss-of-coolant-accident initiator, requiring a Significance Determination Process Phase 2 analysis. Using Significance Determination Process Phase 2 notebook, "Risk-Informed Inspection Notebook For Diablo Canyon Power Plant – Units 1 and 2," Revision 1, the deficiency is assumed to impact the "Stuck-Open Power Operated Relief Valve" accident initiator only. The condition existed for less than 3 days. The inspectors considered that performance of the surveillance test would cause the valve to open and therefore increased the likelihood of the power-operated relief valve sticking open in the Phase 2 analysis. All mitigating equipment, including the power-operated relief valve low pressure interlock and power-operated relief valve block valve, was assumed operable, and operators were able to respond to a potential event. The finding was determined to be of very low safety significance using the Significance Determination Process Phase 2 Analysis and the results were reviewed by an NRC senior reactor analyst.

Inspection Report# : 2003006*(pdf)*

Significance: Mar 29, 2003
Identified By: NRC
Item Type: FIN Finding
Failure to Control Work Activities Resulted in False Reactor Vessel Level Indication Changes
A self-revealing finding was identified for failing to consider the impact of filling the pressurizer relief tank during midloop operations, which resulted in an indicated decrease in reactor vessel level.

The finding was more than minor because it affects an attribute and objective of the Initiating Events Cornerstone in that configuration control of shutdown equipment lineup was inadequate. The inspectors found that the procedural requirements were met in that no initiation of transient conditions were induced while at midloop operation. However, the finding was more than minor because an initiating event cornerstone attribute, involving shutdown instrumentation alignment, was potentially affected. Specifically, two reactor vessel level instrument indications, used to support shutdown cooling in midloop operation, indicated that a prompt decrease in reactor coolant inventory had occurred. NRC Manual Chapter 0609, Appendix G, "Shutdown Operations - Significance Determination Process," dated February 27, 2001, was utilized to assess the overall safety significance. Table 1 for Reactor Coolant System Open and

Refueling Cavity Level < 23', Section II.B , Inventory Control Guidelines-Procedures/Training, considers that training procedures and administrative controls are implemented to avoid operations that could lead to perturbations in reactor coolant system level control or decay heat removal flow. The issue was determined to involve change in indication only and therefore was assessed as having very low safety significance.
Inspection Report# : 2003005*(pdf)*

Significance: Mar 29, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement Procedure to Control Bucket Truck Next to Single Supply of Offsite Power During Hot Midloop Operations
The inspectors identified a noncited violation of Technical Specification 5.4.1.a for implementation of procedures for operation of offsite power source access. Procedure AD8.DC51 was not followed in that a bucket truck was operated next to Startup Transformer 2-2 that could have directly or indirectly affected the single source of offsite power without approval of the shift foreman and a detailed schedule review having been performed by the outage organization.

The finding was more than minor since a significant outage work activity was not approved by the shift foreman during Unit 2 midloop operation, a period when industry experience has demonstrated the potential for significant events to occur. NRC Manual Chapter 0609, Appendix G, Shutdown Operations - Significance Determination Process," dated February 27, 2001, was utilized to assess the overall safety significance. Table 1 for Reactor Coolant System Open and Refueling Cavity Level< 23', Section III.A , Power Availability Guidelines-Procedures/Training/Administrative Controls, considers that work activities do not have significant potential to affect existing operable power supplies and that there is control over switchyard and transformer yard activities. The finding is of very low risk significance since a safety assessment was performed prior to the outage work activity beginning and the shift foreman subsequently approved the work to continue without revision to the work safety assessment or the work plan.
Inspection Report# : 2003005*(pdf)*

Mitigating Systems

Significance: Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Instructions and Acceptance Criteria During Diesel Engine Generator Automatic Voltage Regulator Card Inspections
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, for Pacific Gas & Electric Company's failure to utilize acceptance criteria and instructions for Diesel Engine Generator 2-2 auto-voltage regulator card inspection. This failure would have left degraded solder joints on the auto-voltage regulator card. This condition resulted in slow voltage rise times on Diesel Engine Generator 1-3.

The finding impacted the mitigating system cornerstone and was more than minor when assessed using Inspection Manual Chapter 0612, Appendix E, Example 4.a. Similar to Example 4.a, the subsequent solder work on the Diesel Engine Generator 2-2 auto-voltage regulator card revealed degraded solder joints similar to those on Diesel Engine Generator 1-3, which was the apparent cause for its slow voltage rise time. The finding is of very low safety significance since there was no loss of actual safety function, no loss of a safety-related train for greater than the diesel engine generator Technical Specification allowed outage time, and the finding is not potentially risk significant due to a

seismic, fire, flooding, or severe weather initiating event.
Inspection Report# : 2003007*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Promptly Identify and Correct a Degraded Mechanical Governor
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI for the failure to promptly identify and correct a degraded mechanical governor on Diesel Engine Generator 2-2. This failure caused the degraded governor to remain in service for over 6 months, which resulted in increasing difficulty by operators to maintain the required load on the diesel engine generator.

The finding impacted the mitigating systems cornerstone and was more than minor when assessed using Inspection Manual Chapter 0612, Appendix E, Example 4.g. In Example 4.g, the failure to correct a condition adverse to quality was more than minor unless the condition had little or no safety impact. Following the March 20, 2003, surveillance test, the ability of Diesel Engine Generator 2-2 to complete its mission time of 7 days was questionable. Therefore, the degraded governor had more than minor impact on safety. The finding is of very low safety significance since there was no loss of an actual safety function, no loss of a safety-related train for greater than the Diesel Engine Generator 2-2 Technical Specification allowed outage time, and the finding is not potentially risk significant due to a seismic, fire, flooding, or severe weather initiating event.
Inspection Report# : 2003007*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Two Examples of a Violation of Technical Specification 5.4.1.d for Inadequate Fire Protection Implementation
The inspectors identified two examples of a violation of Technical Specification Section 5.4.1.d, for failure to establish, implement, and maintain adequate procedures covering fire protection program implementation.

Example 1: The licensee failed to adequately implement fire protection program requirements specified in Calculation M-944 "10 CFR 50 Appendix R, Alternate Shutdown Methodology Time and Manpower Study/Safe Shutdown System Considerations." Specifically, in a control room fire scenario requiring control room evacuation and remote shutdown, operators failed to complete actions required for achieving safe shutdown specified in Procedure OP AP-8A, "Control Room Inaccessibility Hot Standby," within the times assumed in Calculation M-944.

This finding was of greater than minor significance because it impacted the mitigating systems cornerstone and adversely affected the ability of the licensee to manually operate certain components required for safe shutdown within the analyzed times. Specifically, in a simulated field walkdown, operators were not able to establish auxiliary feedwater within 30 minutes as required by analysis nor close a stuck open power operated relief valve within 5 minutes. The inspectors used Appendix F of Manual Chapter 0609 and determined that the inability to perform the safe shutdown procedures required a Phase 2 and Phase 3 analysis in the significance determination process. The Phase 2 and 3 analysis of the ignition frequencies and the potential heatup of the core in this degraded condition, revealed that this finding was of very low safety significance.

Example 2: The licensee failed to adequately implement fire protection program requirements for a fire in the control room requiring control room evacuation and remote shutdown. Specifically, the licensee failed to provide adequate information in procedure OP AP-8A, "Control Room Inaccessibility Hot Standby," or on the Unit 2 hot shutdown panel concerning the correct hot shutdown panel switch positions of certain components required for safe shutdown.

Consequently, in stepping through procedure OP AP-8A, operators failed to transfer control of the auxiliary feedwater throttle valves and steam generator atmospheric dump valves from the control room to the hot shutdown panel.

This finding was of greater than minor significance because it impacted the mitigating systems cornerstone and adversely affected the ability of the licensee to take control of certain components required for safe shutdown. Specifically, information identifying the correct hot shutdown panel switch positions for the auxiliary feedwater throttle valves and steam generator atmospheric dump valves were not provided to the operators. During a control room fire and remote shutdown, if not placed in the correct positions, these components would have remained vulnerable to fire damage that could cause spurious operation. The inspectors used Appendix F of Manual Chapter 0609 and determined that the inability to perform the safe shutdown procedures required a Phase 2 and Phase 3 analysis in the significance determination process. The Phase 2 and 3 analysis of the ignition frequencies and the potential heatup of the core in this degraded condition, revealed that this finding was of very low safety significance.
Inspection Report# : 2003007*(pdf)*

Significance: Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Identify and Prevent Check Valve Problems
A self-revealing, NCV of 10 CFR Part 50, Appendix B, Criterion XVI was identified for the failure to promptly identify and correct a leak in Check Valve FW-2-370 and the backward installation of Check Valve FW-2-377 disk. This finding resulted in minor backflow of feedwater to Auxiliary Feedwater Pump 2-2.

Using Inspection Manual Chapter 0612, Appendix E, Example 5.b, the finding is more than minor because Auxiliary Feedwater Pump 2-2 was returned to service, prior to the discovery of the leak and the incorrect check valve reassembly, despite auxiliary feedwater system backflow alarms and industry experience on proper assembly of check valves. The finding did not result in sufficient backflow and temperature increase to prevent the pump from providing adequate auxiliary feedwater flow to the steam generators. Therefore, using the Significance Determination Process Phase 1 Worksheet, as described in Inspection Manual Chapter 0609, Appendix A, the finding was determined to be of very low safety significance. Specifically, the finding did not result in a loss of safety function or screen as potentially risk significant from an external event.
Inspection Report# : 2003006*(pdf)*

Significance: Mar 29, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform Testing To Assure Valve Performance
A self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, was identified for failure to verify by testing the ability of Component Cooling Water Valve CCW-2-18 to meet its design basis function of isolating a postulated leak between trains of component cooling water. This valve was credited for ensuring that a single passive failure of the component cooling water system, that resulted in a 200 gallon-per-minute leak, could be isolated within 20 minutes. However, for several years, the valve had a damaged liner that precluded any effective isolation capability that had not been identified because the licensee had not established a leak testing program for the valve.

The failure to provide adequate testing to ensure that Valve CCW-2-18 could meet its design basis function affected the Mitigating Systems Cornerstone and is more than minor because it had an actual impact on safety. Specifically, the lack of a test program allowed the existence of the damaged valve liner for a significant period of time. A Phase 3 significance determination process assessment was performed for a similar condition that occurred on Unit 1 (NRC Integrated Inspection Report 50-275/00-16; 50-323/00-16, Section 1R14.2). The Phase 3 assessment considered that a

passive failure of one train of component cooling water (a low energy system) would have to occur prior to calling upon a comparable valve (to Valve CCW-2-18) to perform its isolation function, a very low probability failure. The assessment also considered that a safety-related 250 gpm makeup source was available to replenish the component cooling water system. Two other nonsafety-related makeup sources were also available. The inspectors noted that although the ability to split the trains was compromised, the component cooling water system could have met its intended safety function despite the condition, with adequate normal and backup makeup systems available. This finding was determined to be of very low safety significance.
Inspection Report# : 2003005*(pdf)*

Significance: G Mar 29, 2003
Identified By: NRC
Item Type: FIN Finding
Ineffective corrective action in placement of ventilation louvers on 12 kv grounding transformer fuse boxes
The inspectors identified a finding involving ineffective corrective action in placement of ventilation louvers on 12 kV grounding transformer fuse boxes. The placement of the louvers introduced a new failure mechanism, which resulted in a recurrence of a previous event. On August 4, 2001, Units 1 and 2 experienced a loss of startup power as a result of multiple electrical faults in the Startup Transformer 1-1 grounding transformer fuse box. Nonconformance Report N0002130, "Loss of Unit 1 and 2 Startup Power," documented that the primary cause of the electrical faults was condensation inside the fuse box. The ventilation louvers contributed to the event by allowing outside (salty) air to be drawn into the fuse box. The ventilation louver was installed as a corrective action after the November 22, 1996, Auxiliary Transformer 1-1 grounding transformer fuse box event.

The SDP Phase 3 analysis was performed by the Office of Nuclear Reactor Regulation Probabilistic Safety Assessment Branch. The analysis indicated that the estimated change in core damage probability for internal and external events probabilities was approximately 6E-7, and the change in large early release probability was approximately 6E-8. The conclusion of the analysis characterized the performance deficiency as an issue of very low safety significance.
Inspection Report# : 2003005*(pdf)*

Significance: G Oct 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
Willful violation of maintenance procedure when torquing atmospheric dump valve nuts.
A violation of Technical Specification 5.4.1.a occurred for failure to follow a maintenance procedure for torquing atmospheric dump Valve PCV-21 bonnet cover bolts. The maintenance procedure required incrementally torquing the studs and nuts using a calibrated torque wrench. However, the mechanics willfully violated the procedure by using a hammer and extender to tighten the bolts, resulting in cracking of 7 out of 8 of the stud and nut combinations. This Severity Level IV violation is being treated as a noncited violation consistent with Section VI.A.1of the NRC Enforcement Policy. Although this violation was willful, the licensee promptly reported the results of the investigation to the NRC, the acts were committed by low level individuals, management was not involved nor was the action due to lack of management oversight, and the licensee took significant remedial action. This violation is in the corrective action program as Nonconformance Report N0002134.

The inspectors evaluated the as-found condition of the studs and nuts on Atmospheric Dump Valve PCV-21 using the Significance Determination Process. The inspectors determined that the multiple stud and nut failures represented a credible impact on safety in that their failure could have resulted in the body-to-bonnet separation of Valve PCV-21. The failure would have been similar to a failed open atmospheric dump or secondary safety relief valve. The inspectors considered that the failure of the degraded studs would result in a potential loss of the main steam boundary and a direct release path following a postulated Unit 2 Steam Generator 3 tube rupture. Although the condition resulted in a

minor steam leak, the licensee completed a metallurgical analysis that demonstrated the remaining studs and nuts had sufficient strength, along with the stud configuration around the valve bonnet, to prevent catastrophic failure of Valve PCV-21. No immediate operability concerns were identified for any of the other atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded the issue had very low safety significance.
Inspection Report# : 2002004*(pdf)*

Barrier Integrity

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Implement Outage Safety Management Controls to Containment Closure
An NRC-identified noncited violation of Technical Specification 5.4.1.a was determined for the failure to promptly notifying the shift foreman, as required by procedure, when it was ascertained that containment closure could not be established during reduced inventory operations. Containment closure could not be established because of a stuck fuel transfer cart that prevented the fuel transfer tube isolation valve from being closed. Pacific Gas and Electric Company personnel calculated that during the 2.5-hour period the fuel transfer tube could not be isolated, the reactor coolant system could potentially begin boiling within 22 minutes, if shutdown cooling was lost.

The finding is more than minor because it affected the barrier cornerstone objective of providing reasonable assurance that the containment would preclude the release of radionuclides from accidents or events. The inspectors evaluated the safety significance of the finding using Inspection Manual Chapter 0609, Appendix G, Shutdown Operations. Section IV to Containment Control Guidelines was considered and a Significance Determination Process Phase 2 and 3 analysis was determined to be appropriate because of the impact on the ability to isolate the fuel transfer canal. The initial conditions considered for the containment integrity significance determination process were: (1) the condition occurred within 8 days of the outage, (2) the reactor vessel level was less than 23 feet from the top of the reactor vessel flange, (3) the reactor coolant system was vented, (4) a robust mitigation capability was in place and the condition existed for less than 8 hours. Utilizing Table 6.4 , Phase 2 Risk Significance - Type B Findings at Shutdown (For POS 1/TW-E and POS 2/TW-E in which the finding occurs during the first 8 days of the outage) the finding was potentially white. Note 2, to Table 6.4, specifies that for Type B findings (does not effect core damage frequency) that exist for less than 8 hours, then the color of the finding is reduced by an order of magnitude. A senior reactor analyst also reviewed the reactor plant initial conditions, fuel transfer canal configuration and mitigating strategies specified in Pacific Gas and Electric Company's outage plan. Based on Inspection Manual Chapter 0609, Appendix H, "Containment Integrity Significance Determination Process," and an independent Phase 3 review, the NRC staff concluded that the finding was of very low safety significance.
Inspection Report# : 2003006*(pdf)*

Emergency Preparedness

Occupational Radiation Safety

Significance: G Mar 29, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Follow Radiation Work Permit Requirements
On February 13, 2003, the inspectors identified a violation of Technical Specification 5.4.1 for failure to follow radiation work permit requirements. Specifically, radiation workers failed to contact radiation protection personnel prior to working greater than 8 feet above the floor on Safety Injection Valve SI-2-8821B. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy.

The issue was more than minor because the failure to follow radiation work permit requirements has the potential for unplanned or unintended dose which could have been significantly greater as a result of higher radiation or contamination levels. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it did not involve as low as reasonably achievable (ALARA) planning and controls, there was no personnel overexposure, there was no substantial potential for personnel overexposure, and the finding did not compromise the licensee's ability to assess dose.
Inspection Report# : 2003005*(pdf)*

Public Radiation Safety

Physical Protection

Significance: N/A Jan 10, 2003
Identified By: NRC
Item Type: FIN Finding
Verification of Compliance With Interim Compensatory Measures Order
On February 25, 2002, the NRC imposed by Order, Interim Compensatory Measures to enhance physical security. The inspectors determined that, overall, the licensee appropriately incorporated the Interim Compensatory Measures into the site protective strategy and access authorization program; developed and implemented relevant procedures; ensured that the emergency plan could be implemented; and established and effectively coordinated interface agreements with offsite organizations.
Inspection Report# : 2003003*(pdf)*

Miscellaneous

Last modified : December 01, 2003