Diablo Canyon 1
1Q/2004 Plant Inspection Findings

Initiating Events

Significance: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Maintain Design Drawings Lead to Inadvertent Inward Rod Motion
A self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion V occurred because of a failure to accurately reflect wiring changes in drawings following a design modification. Subsequently, deficient drawings were used by maintenance personnel in another design modification and contributed to inadvertent, inward control rod motion that reduced reactor power by approximately 2 percent.

Using Example 4.b of Inspection Manual Chapter 0612, Appendix E, the finding is greater than minor since maintenance personnel performed activities with the deficient drawings and without verifying the function of the leads, caused a small plant transient. The finding, which is under the initiating events cornerstone, was of very low safety significance since operators performed in a timely, appropriate manner. Also, the transient was not severe enough to challenge the capability of the plant's mitigating equipment. The finding was reviewed against the initiating event screening criteria documented in Inspection Manual Chapter 0609, Appendix A, Attachment 1, Significance Determination Process Phase 1 Screening Worksheet for Initiating Event, Mitigating Systems and Barrier Cornerstones. The finding is of very low safety significance because the condition did not contribute to a loss of coolant initiator, would not contribute to the likelihood a mitigating system would not be available and did not involve an external event initiator. In addition, the operators responded in a timely and appropriate manner. Plant mitigating equipment was not challenged by the transient..
Inspection Report# : 2003006*(pdf)*

Mitigating Systems

Significance: TBD Dec 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Train Operations Responders in Support of the Fire Brigade
The inspectors identified a violation of Technical Specification 5.4.1.d which requires written procedures be established, implemented and maintained covering the Fire Protection Program implementation. Specifically, PG&E failed to adequately establish and implement procedural changes that provided for senior control operators, licensed control operators and non-licensed, level 8 nuclear operators to serve in the operator responder position. The inspectors noted that the applicable attachment to the procedure for conduct of the operations response position was not established until after training had been provided on implementing the procedure. Operations responders supporting the fire brigades exhibited a knowledge weakness in activities such as communications with the control room, manual actuation of fire suppression equipment, and providing information to the fire brigade regarding safe shutdown equipment.

The finding is unresolved pending completion of a significance determination. The finding is greater than minor because it affects the mitigating system cornerstone objective by degrading fire brigade effectiveness, which is a fire protection defense-in-depth element.
Inspection Report# : 2003008*(pdf)*

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Adequately Monitor Auxiliary Feedwater System According to 10 CFR 50.65(a)(2)
The inspectors identified a noncited violation for the failure to adequately monitor the performance of the Unit 1 auxiliary feedwater system in accordance with 10 CFR 50.65(a)(2). Specifically, the unavailability time performance criteria for the auxiliary feedwater system had been exceeded during its monitoring period, but the system was not monitored per 10 CFR 50.65(a)(1).

The finding impacted the mitigating systems cornerstone objective to ensure the availability and reliability of the auxiliary feedwater system to respond to initiating events. The finding is greater than minor using Example 1.f of Inspection Manual Chapter 0612, Appendix E. Similar to the example, the inspectors identified that Pacific Gas and Electric did not consider unavailability time for the Unit 1 auxiliary feedwater system, although the unavailability time was due to prior poor maintenance practices on Valve FW-1-FCV-437. If the unavailability time was considered, the 10 CFR 50.65(a)(2) evaluation would be invalid. Using the Significance Determination Process Phase I worksheet in Inspection

Manual Chapter 0609, Appendix A, 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 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# : 2003008*(pdf)*

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Provide Adequate Technical Bases for Core Exit Thermocouple Radial Temperature Measurement
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, when Pacific Gas and Electric personnel failed to adequately evaluate the capability of core exit thermocouples to measure the radial temperature gradient for Quadrant 1 of the Unit 1 reactor core. Specifically, maintenance personnel inadvertently swapped core exit thermocouples at a connection, leaving only 3 operable thermocouples per Trains A and B for Quadrant 1. When questioned by the inspectors, engineering personnel could not provide an adequate technical bases for how measurement of radial temperature gradient could be accomplished.

The finding impacts the mitigating system cornerstone through degraded overall availability of the components within a system used to assess and respond to initiating events to prevent undesirable consequences. The finding was greater than minor when compared to Example 3.a of Inspection Manual Chapter 0612, Appendix E. Similar to Example 3.a, Pacific Gas and Electric performed additional work to verify the ability of the core exit thermocouples to measure radial temperature gradient within Quadrant 1 of the Unit 1 reactor core. Using the Significance Determination Process Phase 1 screening worksheet from Inspection Manual Chapter 0609, Appendix A, the finding was determined to be of very low safety significance since the deficiency was confirmed not to result in loss of function per Generic Letter 91-18, Revision .
Inspection Report# : 2003008*(pdf)*

Significance: G Dec 31, 2003
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
Failure to Promptly Identify and Correct Rockwell-Edwards Valves Susceptible to Packing Gland Follower Flange Failures
A self-revealing violation of 10 CFR Part 50, Appendix B, Criterion XVI, was identified for failure to promptly identify and correct a condition adverse to quality. Specifically, in December 2000, Pacific Gas and Electric failed to identify and correct the population of Rockwell-Edwards valves in safety-related and risk-significant systems that were susceptible to failure of the packing gland follower flange from intergranular stress corrosion cracking. Pacific Gas and Electric received an industry notification in December 2000 that Rockwell-Edwards valves were vulnerable for this type of failure, but initiated corrective actions on a very limited population of valves (those involving a trip risk). As a result, on December 3, 2003, the packing gland follower flange for safety injection Valve SI-1-8890A (pressure equalization valve) on the hot leg injection line failed, due to intergranular stress corrosion cracking, resulting in excessive packing gland leakage.

The finding impacted the mitigating systems cornerstone through degraded equipment performance for a system train that responds to initiating events to prevent undesirable consequences. The finding is greater than minor because the finding would become a more significant safety concern if the valve condition was left uncorrected. The amount of leakage from the valve would be significantly greater than a 30 drop per minute leak rate, if the safety injection pumps were fully running in the hot leg injection mode. The Valve SI-1-8890A leak rate is bounded by a residual heat removal pump seal failure. Pacific Gas and Electric concluded the safety injection system was operable but degraded because both safety injection system trains would be available to provide adequate flow if a demand occurs. Using the Significance Determination Process Phase 1 worksheet in Inspection Manual Chapter 0609, Appendix A, the finding was determined to be of very low safety significance, since there is no loss of an actual safety function, no loss of a safety-related train for greater than the 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# : 2003008*(pdf)*

Significance: G Oct 07, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Perform A Prompt Operability Assessment for Multiple Battery Charger Failures
A non-cited violation was identified for inadequate corrective actions for multiple battery charger failures. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part, that significant conditions adverse to quality shall be promptly identified, the cause shall be determined, and corrective action shall be taken to preclude repetition. Additionally, the identification, cause, and corrective actions associated with a significant condition adverse to quality shall be documented and reported to appropriate levels of management. Contrary to the above, the team discovered multiple examples of PG&E's failure to promptly identify, determine the cause, apply corrective action and report to appropriate management the design deficiency and other causes for multiple failures in vital battery chargers between January 1999 and May 2003. The failure to correct the battery charger design deficiency allowed battery charger failures in both units.

This issue was more than minor because it could become more significant safety concern if not corrected because multiple failures could exist simultaneously without being detected, although this did not represent a common mode failure. It affected the Mitigating Systems Cornerstone The issue was of very low safety significance because the primary failure mechanism involved an increased failure rate, but did not constitute a

common cause failure mode. A Phase 3 SDP determined that there was a good likelihood that at least one 125 Vdc bus would have power during design basis conditions, allowing the plant to reach a safe shutdown condition.
Inspection Report# : 2003010*(pdf)*

Significance: G Oct 07, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Multiple Examples of A Violation of 10 CFR Part 50, Appendix B, Criterion XVI, Related to Battery Charger Failures Between 1999 and 2003
The team identified that, in the case of repeated failures of Class 1E battery chargers between January 1999 and May 2003, the licensee's corrective action process was ineffective in a number of ways. The licensee failed to appropriately prioritize and evaluate battery charger failures, individually and collectively. The Action Request Review Team consistently assigned low significance, did not assign any cause investigation, and did not recognize a trend of charger failures existed, even when multiple failures were identified in a short period of time. The licensee inappropriately judged the significance of the charger failures on lack of actual adverse plant consequences rather than the potential consequences of similar failures during a design basis event. Corrective actions were ineffective and limited to component replacement, allowing additional failures to occur. The licensee's Corrective Action Program had little defense-in-depth and no effective feedback mechanisms in the area of determining the significance of an issue and assigning an appropriate type of cause assessment. The licensee did not have a formal program for trending equipment failures. The program did not give adequate consideration to determining the extent of condition or potential for common mode failure.
Inspection Report# : 2003010*(pdf)*

Significance: G Oct 07, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Ten Examples of A Violation of Technical Specification 3.8.4 for Battery Chargers Inoperable Longer Than the AOT.
A non-cited violation of Technical Specification 3.8.4 was identified because various Class 1E DC chargers in both units were incapable of performing their intended safety functions of supplying 125 Vdc loads and recharging the associated battery for longer than permitted by the associate action statements during various times between January 1999 and May 2003. This condition was allowed to occur because the licensee failed to identify the cause and take effective corrective actions from earlier failures. Specifically, multiple, and in some cases repetitive, failures occurred which were undetected until the chargers were fully loaded, as would be the case during performance of its intended safety function.

This issue was more than minor because it could become more significant safety concern if not corrected because multiple failures could exist simultaneously without being detected, although this did not represent a common mode failure. It affected the Mitigating Systems Cornerstone. The issue was of very low safety significance because the primary failure mechanism involved an increased failure rate, but did not constitute a common cause failure mode. A Phase 3 SDP determined that there was a good likelihood that at least one 125 Vdc bus would have power during design basis conditions, allowing the plant to reach a safe shutdown condition.
Inspection Report# : 2003010*(pdf)*

Significance: G Sep 27, 2003
Identified By: NRC
Item Type: NCV NonCited Violation
Failure to Promptly Identify and Correct Diesel Engine Generator Lube Oil Carbonization
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI for Pacific Gas & Electric Company's failure to promptly identify and correct lube oil carbonization in diesel engine generators. This failure resulted in increased lube oil filter differential pressures and partial clogging of precirculation lube oil lines for Diesel Engine Generators 1-3 and 2-2.

The finding impacted the mitigating systems cornerstone and was more than minor when assessed using Inspection Manual Chapter 0612, Appendix E, Example 3.b. In Example 3.b, a discrepancy between an actual condition and the design was more than minor if the operation of the system was adversely affected. With respect to this finding, the carbonized oil clogged the precirculation lube oil line requiring Diesel Engine Generator 1-3 unavailability to clean the line. Additionally, the carbonized lube oil caused an increase in lube oil filter differential pressure. 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 1-3 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: G Jun 28, 2003
Identified By: NRC
Item Type: NCV NonCited Violation

Failure to Promptly Identify and Correct a Faulty Automatic Voltage Regulator Card

An NRC-identified NCV of 10 CFR Part 50, Appendix B, Criterion XVI was determined for the failure to identify and correct a faulty automatic voltage regulator card that resulted in Diesel Engine Generator 1-3 failures. Diesel Engine Generator 1-3 remaining in service for over 6 months with a faulty automatic voltage regulator card. Overall there were three occasions where the diesel engine generator did not achieve its required voltage rise time.

The finding was more than minor when assessed using Inspection Manual Chapter 0612, Appendix E, Example 4.f. Similar to the example, Diesel Engine Generator 1-3 was inoperable from August 31, 2002, to February 23, 2003, which is the time period that the fault in the automatic voltage regulator card was determined to exist. Using the Significance Determination Process Phase 1 Worksheet in Inspection Manual Chapter 0609, the inspectors determined that there was an actual loss of a safety function for greater than the diesel engine generator Technical Specification allowed outage time, which required a Significance Determination Process Phase 2 analysis. The finding was reviewed by senior reactor analysts and an engineer with the Office of Nuclear Reactor Regulation to identify the sequences to be analyzed. Specifically, the sequences involving a loss of offsite power with a large break loss-of-coolant-accident were evaluated since Diesel Engine Generator 1-3 exhibited a slow voltage rise time only. In all other sequences, the emergency alternating current safety function was credited. An additional mitigating factor is the two residual heat removal pumps were located on the other two vital buses. The Significance Determination Process Phase 2 analysis determined that the finding was of very low safety significance.
Inspection Report# : 2003006*(pdf)*

Barrier Integrity

Emergency Preparedness

Occupational Radiation Safety

Significance: G Dec 31, 2003
Identified By: NRC
Item Type: FIN Finding
Failure to Maintain Collective Doses ALARA
A finding was identified because Pacific Gas and Electric failed to maintain collective doses as low as is reasonably achievable. Specifically, work activities associated with Radiation Work Permit 03-2055, "Reactor Coolant Pump (RCP) 2-2, 10 year inspection," exceeded 5 person-rem and the dose estimation by more than 50 percent due to a miscommunication among work groups.

The failure to maintain collective doses as low as is reasonably achievable is a performance deficiency. This finding was more than minor because it is associated with the Occupational Radiation Safety Cornerstone attribute (program and process) and affected the associated cornerstone objective (to ensure adequate protection of workers' health and safety from exposure to radiation). This occurrence involved inadequate planning which resulted inunplanned, unintended occupational collective dose for the work activity. When processed through the Occupational Radiation Safety Significance Determination Process, this finding was found to have no more than very low safety significance because the finding was an as low as is reasonably achievable planning issue and Pacific Gas and Electric Company's 3-year rolling average collective dose was less than 135 person-rem.
Inspection Report# : 2003008*(pdf)*

Public Radiation Safety

Physical Protection

Miscellaneous

Last modified : May 05, 2004