Diablo Canyon 2
4Q/2003 Plant Inspection Findings

Initiating Events

Significance: G 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: G 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: G 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: 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 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: 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: 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 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: G 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: 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: Mar 18, 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 procedure

Inspection Report# : 2003002*(pdf)*

Barrier Integrity

Significance: 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: 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)*

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 : March 02, 2004