# Diablo Canyon 2

## Initiating Events

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

## Mitigating Systems

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

**Significance:** G May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation

**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet

Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : April 01, 2002

# Diablo Canyon 2

## Initiating Events

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

## Mitigating Systems

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50,

Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system

would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000

Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously

guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective

action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : April 01, 2002

# Diablo Canyon 2

## Initiating Events

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

## Mitigating Systems

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process,

was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation

**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

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## Public Radiation Safety

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**

Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : March 29, 2002

# Diablo Canyon 2

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## Initiating Events

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

## Mitigating Systems

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of

the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had

dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

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## Public Radiation Safety

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**

Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : March 28, 2002

# Diablo Canyon 2

## Initiating Events

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

## Mitigating Systems

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this

issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000
Identified By: NRC

Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining

the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.

Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

Last modified : March 28, 2002

# Diablo Canyon 2

## Initiating Events

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

## Mitigating Systems

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating

system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.

Inspection Report# : 2001006*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000

Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and

controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective

action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

Last modified : March 27, 2002

# Diablo Canyon 2

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## Initiating Events

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

## Mitigating Systems

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be

impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** **G** Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** **G** Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green

Inspection Report# : 2000010*(pdf)*

**Significance:** May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** May 12, 2000

Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and

controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective

action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : March 26, 2002

# Diablo Canyon 2

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## Initiating Events

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

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## Mitigating Systems

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**

The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.

Inspection Report# : 2000014*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor

Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : March 01, 2002

# Diablo Canyon 2

## Initiating Events

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation

are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

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# Mitigating Systems

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel

engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to limit the proximity of transient equipment near safety-related systems due to seismic interaction concerns**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to adequately limit the proximity of transient equipment from safety-related systems that may be required during a seismic event. Technical Specification 5.4.1.a requires that written procedures be implemented for equipment control. Procedure AD4.ID3, "SISIP Housekeeping Activities," Revision 4A, Section 5.1.1, required that transient equipment not create a potential seismically induced system interaction. Contrary to the above, on January 14, 2002, the inspectors discovered an unsecured portable welding machine staged approximately 8 inches from the normal and Class 1 air supply lines for Unit 2 atmospheric dump Valve MS-2-PCV-21. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0547478. This violation was more than minor because there was a credible impact on safety because the atmospheric dump valve could not be remotely operated due to loss of air supply in a seismic event. This issue was determined to be of very low safety significance because the other three atmospheric dump valves on the steam generators could be used to adequately cool the reactor coolant system.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Exceeding the licensed power limit due to a failure to follow procedures**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Procedures OP L-4, "Normal Operation at Power," Revision 39, Section 5.4 and OP B-9:I, "Primary Sampling System - Make Available and Place in Service," Revision 7, stated, in part, that when pressurizer steam space sampling to the volume control tank was initiated, two backup pressurizer heaters were to be placed in service. On December 28, 2001, operators initiated pressurizer steam space sampling to the volume control tank without placing two backup pressurizer heaters into service. This resulted in a dilution of the volume control tank that increased reactor power above 100 percent for approximately 2½ hours. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0546623. This violation was more than minor because it had credible impact on safety due to the unplanned change in reactivity. This issue was determined to be of very low safety significance (Green) because the reactivity addition was not of an appreciable amount to challenge the safety systems or operating limits, and operators were able to return reactor power to desired levels in a controlled manner.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform adequate postmaintenance test of a reactor protection system analog input card**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33 lists procedures for surveillance tests. Procedure STP I-33, "Reactor Trip and Engineered Safety Feature Response Time Test," Revision 6, partially implemented this requirement and stated in Section 3.3.3.b that replacement of an Eagle-21 card required time response testing of the appropriate channels. Contrary to the above, the licensee replaced Card 2 of Rack 13 of the Unit 2 Eagle 21 system on September 18, 2001, but did not perform time response testing as a postmaintenance test and returned the component to service. This card affected reactor trip and safety injection setpoints for Loop 3 reactor coolant system temperature, pressurizer pressure, and pressurizer level. Upon discovery, the time response test was successfully performed on March 7, 2002. This event is described in the licensee's corrective action program, reference Action Request A0550656. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation was more than minor because it had credible impact on safety due to the card affecting several mitigating systems and actuations. This issue was determined to be of very low safety significance (Green) because when the post maintenance testing was conducted, the

applicable channels passed.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Reactor operation with the steam generator level-low low trips and engineered safety actuation instrumentation inoperable for both unit 1 and 2.**
The licensee failed to identify that indications on three safety-related steam generator water level instruments were not as expected during a loss of feedwater to steam generator 2-4, and that an expected engineered safety features (ESF) actuation and automatic reactor trip actuation did not occur when required. The scope of the review was narrow, and focused on explaining wide range instrument indications when narrow range instruments were the problem. Tentative explanations were accepted, when comparing the results with other available indications would have demonstrated this theory was wrong. As a result, missed the opportunity to question the operability of those instruments, and restarted and operated the plant with required instrumentation inoperable. The licensee's failure to promptly recognize inoperable trip and actuation functions and comply with Technical Specification requirements had a credible impact on safety, because required trip and ESF actuation would have delayed responses and reduced the water mass available for the heat sink function in the affected steam generator(s). This issue had a credible impact on the operability of the steam generator water level low-low trip and ESF actuation, as well as having the potential for affecting the integrity of the reactor coolant system boundary. A Phase 2 Significance Determination Process review resulted in this issue screening as Green (very low safety significance) because simulations showed that the reactor was protected by secondary automatic trips, and the auxiliary feedwater actuation would have functioned following a trip.
Inspection Report# : 2002007*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment

Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**

The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007 *(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

**Significance:** May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007 *(pdf)*

**Significance:** Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007 *(pdf)*

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# Occupational Radiation Safety

**Significance:** Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009 *(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 12, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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## Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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## Miscellaneous

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of

2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : July 22, 2002

# Diablo Canyon 2

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## Initiating Events

**Significance:** Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant

specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

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# Mitigating Systems

**Significance:** G May 31, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**The failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786.**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786. The licensee, contrary to the procedural requirements, placed the issue in a process to validate the initial perception that diesel fuel oil tank capacity would meet design requirements. The licensee documented on July 19, 2001, that Calculation M-786 had not been updated with regard to changes that would affect diesel fuel usage in the Technical Specifications, Design Criteria Memorandum, the Final Safety Analysis Report Update, and the Emergency Operating Procedures. The licensee determined that such changes could have an adverse impact on the design and licensing basis related to adequate diesel fuel oil storage. The issue was determined to be of very low risk significance during Phase 1 of the NRC Significance Determination Process, because the Calculation M-786 was found to be conservative with respect to diesel generator loads and, therefore, the diesels remained operable. The failure to adequately address operability of potentially nonconforming conditions, if left uncorrected, could become a more significant safety concern, therefore, the issue was determined to be more than minor. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0553285. (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance: TBD** May 31, 2002

Identified By: NRC
Item Type: FIN Finding
**The installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box.**
The inspectors identified a finding with respect to the placement of ventilation louvers on 12 kV grounding transformer fuse boxes. On August 4, 2001, Units 1 and 2 experienced a loss of startup power as a result of multiple electrical faults in Startup Transformer 1-1 Grounding Transformer Fuse Box. Nonconformance Report N0002130, "Loss of Unit 1 and 2 Startup Power," determined the primary cause of the electrical faults to be condensation inside the fuse box. The contributory cause of the event was the ventilation louver, which allowed outside (salty) air to be drawn into the fuse box. The inspectors' Phase 2 evaluation of this issue using the Significance Determination Process indicated a condition that was potentially greater than green. The inspectors determined that the installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box represented an actual impact on safety since the preferred offsite power was momentarily lost from both units. Subsequently, auxiliary power continued to supply power to plant loads during the loss of startup power, and diesel generators were also available to supply power to safety-related equipment. This issue will remain as an unresolved issue (URI 50-275; 323/2002-02-01) pending completion of the significance determination process (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Exceeding the licensed power limit due to a failure to follow procedures**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Procedures OP L-4, "Normal Operation at Power," Revision 39, Section 5.4 and OP B-9:I, "Primary Sampling System - Make Available and Place in Service," Revision 7, stated, in part, that when pressurizer steam space sampling to the volume control tank was initiated, two backup pressurizer heaters were to be placed in service. On December 28, 2001, operators initiated pressurizer steam space sampling to the volume control tank without placing two backup pressurizer heaters into service. This resulted in a dilution of the volume control tank that increased reactor power above 100 percent for approximately 2½ hours. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0546623. This violation was more than minor because it had credible impact on safety due to the unplanned change in reactivity. This issue was determined to be of very low safety significance (Green) because the reactivity addition was not of an appreciable amount to challenge the safety systems or operating limits, and operators were able to return reactor power to desired levels in a controlled manner.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform adequate postmaintenance test of a reactor protection system analog input card**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33 lists procedures for surveillance tests. Procedure STP I-33, "Reactor Trip and Engineered Safety Feature Response Time Test," Revision 6, partially implemented this requirement and stated in Section 3.3.3.b that replacement of an Eagle-21 card required time response testing of the appropriate channels. Contrary to the above, the licensee replaced Card 2 of Rack 13 of the Unit 2 Eagle 21 system on September 18, 2001, but did not perform time response testing as a postmaintenance test and returned the component to service. This card affected reactor trip and safety injection setpoints for Loop 3 reactor coolant system temperature, pressurizer pressure, and pressurizer level. Upon discovery, the time response test was successfully performed on March 7, 2002. This event is described in the

licensee's corrective action program, reference Action Request A0550656. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation was more than minor because it had credible impact on safety due to the card affecting several mitigating systems and actuations. This issue was determined to be of very low safety significance (Green) because when the post maintenance testing was conducted, the applicable channels passed.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to limit the proximity of transient equipment near safety-related systems due to seismic interaction concerns**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to adequately limit the proximity of transient equipment from safety-related systems that may be required during a seismic event. Technical Specification 5.4.1.a requires that written procedures be implemented for equipment control. Procedure AD4.ID3, "SISIP Housekeeping Activities," Revision 4A, Section 5.1.1, required that transient equipment not create a potential seismically induced system interaction. Contrary to the above, on January 14, 2002, the inspectors discovered an unsecured portable welding machine staged approximately 8 inches from the normal and Class 1 air supply lines for Unit 2 atmospheric dump Valve MS-2-PCV-21. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0547478. This violation was more than minor because there was a credible impact on safety because the atmospheric dump valve could not be remotely operated due to loss of air supply in a seismic event. This issue was determined to be of very low safety significance because the other three atmospheric dump valves on the steam generators could be used to adequately cool the reactor coolant system.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Licensee Restarted Unit 2 Before Recognizing Reactor Trip and Engineered Safety Features Actuation Associated with Lo-lo Steam Generator Water Level was Inoperable**
The failure to promptly identify and correct the steam generator narrow range water level-low low reactor trip system and engineered safety system instrumentation nonconservative setpoint bias following the Unit 2 manual reactor trip on February 9, 2002, is a violation of 10 CFR Part 50, Criterion XVI. The licensee's event review failed to recognize that an engineered safety feature, including a reactor trip, failed to actuate when required during a loss of feedwater event to Steam Generator 2-4. This failure resulted in the licensee restarting Unit 2 with the reactor trip and engineered safety system instrumentation inoperable, and in the operation of both units with the same instrumentation inoperable, in violation of Technical Specification 3.3.1. This issue is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (50-275; 323/2002-07-01). The licensee documented this deficiency in Action Request A0549031. The failure to promptly recognize inoperable trip and actuation functions and comply with Technical Specification requirements had a credible impact on safety. The resulting delays in an automatic reactor trip and engineered safety features actuations would have delayed the plant's response to a loss of feedwater event and reduced the water mass available for the heat sink function in the affected steam generator(s). Further, this deficiency had the potential to affect the integrity of the reactor coolant system boundary. A Phase 3 Significance Determination Process evaluation concluded that the issue had very low safety significance (Green). The finding represents a condition that existed for 5-days. The significance of the steam generator narrow range water level-low low setpoint offset (bias) is reduced if feedwater flow is lost to two or more steam generators. Based on the short duration from the time a single steam generator would dryout (the limiting initiator is a loss of feedwater to a single generator) and actuation of

auxiliary feedwater, the condition does not result in an appreciable increase in the probability of a steam generator tube rupture occurring. The licensee's analysis using the plant specific simulator showed that the engineered safety feature actuation and reactor trip on steam generator water level-low low would have initiated at or before steam generator dryout would occur. The reactor coolant system physical over pressure protective features (safety relief and power operated relief valves) should not be challenged and there were other protective trips in place (over temperature-delta temperature and over pressure delta-temperature) in place that would have protected the reactor coolant system and fuel integrity in the event a manual reactor trip is not initiated on a loss of feedwater flow to a steam generator [Sections 4OA2.a.(2) and 5].
Inspection Report# : 2002007*(pdf)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G Jul 11, 2002
Identified By: NRC
Item Type: FIN Finding
**Grounding resistor vulnerability**
The plant electrical distribution consisted of a design where the three redundant 4160 V safety buses and a non-safety bus were supplied from a common transformer winding during both normal and emergency operation. The 4160 V buses were interconnected by conductors so that a voltage disturbance on any part of the system would affect the entire system. The system had a high resistance grounding design to limit the magnitude of ground faults and to enable continued operation of a faulted load. The grounding resistor admits sufficient fault current to prevent severe over-voltages that could occur. However, if the grounding resistor developed an open circuit, the entire system would be susceptible to over-voltage. The licensee was periodically checking the continuity, but not the actual resistance of the grounding resistors and, thus, assumptions in the design were not being verified. The licensee issued Action Request A0561002 to evaluate the preventive maintenance program of the high resistance grounding program. This issue did not involve a violation of NRC requirements, but was considered to be a finding because it revealed a vulnerability in the licensee's design and maintenance that could result in a safety problem. However, the finding was determined to be of very low safety significance because there was no evidence that the grounding resistor had ever been degraded and that the probability of a grounding resistor failure in combination with a sparking ground fault was very small.
Inspection Report# : 2002006*(pdf)*

**Significance:** May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined,

along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."

Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour

fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

**Significance:** G May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey**
On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**
Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.
Inspection Report# : 2000014*(pdf)*

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# Public Radiation Safety

**Significance:** G Jan 12, 2001

Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to control radioactive materials**
Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.
Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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# Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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# Miscellaneous

**Significance: N/A** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The licensee was effective at identifying problems and placing them into the corrective action program with one exception in the area of operability determinations. Occasionally an operability determination being reviewed by engineering was not timely. For example, the licensee failed to identify and evaluate how differential pressure affected steam generator instrumentation and its affect on operability prior to starting the plant following a trip with unusual steam generator level indications. The licensee appropriately determined the extent of evaluation of individual problems and prioritized the schedule for implementation of corrective actions to address the safety significant issues. In general, corrective actions, when specified, were effective and were implemented in a timely manner. The licensee performed effective audits and assessments. Based on the interviews conducted during this inspection, workers at the site felt free to input safety issues into the problem identification and resolution program.
Inspection Report# : 2002002*(pdf)*

**Significance: N/A** Apr 05, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The team determined that a critical opportunity was missed to promptly identify and correct a risk significant condition adverse to quality involving a nonconservative safety features set point. The licensee's post trip event review process did not ensure that the Unit 2 plant response to a loss of feedwater flow to Steam Generator 2-4 was appropriate in that the steam generator level lo-lo engineered safety features and automatic reactor trip actuations did not occur when required.
Inspection Report# : 2002007*(pdf)*

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC

Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : August 29, 2002

# Diablo Canyon 2

## Initiating Events

**Significance:** G Oct 06, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform a prompt operability assessment for an atmospheric dump valve**
The inspectors identified a violation for the licensee's failure to promptly initiate an operability assessment for a broken bonnet stud on the Unit 2 Atmospheric Dump Valve PCV-21. Procedure OM7.ID12, "Operability Determination," Revision 4C, Section 2.4.3, required the licensee to perform a prompt operability assessment within 72 hours of identifying a degraded condition. In this case the licensee identified the broken stud on August 31; however, the licensee failed to evaluate operability of Valve PCV-21 or the other seven atmospheric dump valves (Units 1 and 2) until September 6 (approximately 160 hours later). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0542300. The inspectors also expressed concern with the effectiveness of the corrective action program in this instance. Personnel failed to recognize a significant condition adverse to quality and have it promptly corrected. The inspectors evaluated this issue 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 failure of the degraded studs could result in a loss of the main steam boundary and a direct release path following a postulated steam generator tube rupture. Subsequently, 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 failure of Valve PCV-21. No immediate operability concerns were identified for the other 7 atmospheric dump valves. Based on the determination that the valve body and bonnet would not have separated, the inspectors concluded this issue had very low safety significance (Section 1R13).
Inspection Report# : 2001007*(pdf)*

**Significance:** G Jul 22, 2001
Identified By: NRC
Item Type: FIN Finding
**Licensee did not consider surveillance activities that placed reactor trip system bistables in trip as reactor trip risks**
The inspectors identified that the licensee had not included surveillance activities, which required placing the reactor trip system bistables in the tripped condition, in their maintenance activity risk evaluations. The licensee failed to categorize any surveillances that included tripping of reactor protection system bistables as trip risk significant on a programmatic basis, despite plant specific and industry events in which reactor trips occurred partially because of a reactor protection channel being in the tripped condition. The licensee's risk management procedure prohibited performing high trip risk evolutions concurrently with removing trip mitigation systems from service. This item was placed in the corrective action system as Action Request A0539532. The inspectors evaluated this finding using the significance determination process. The Phase 1 screening identified that Item 2 under Initiating Event was potentially impacted for a finding that contributed to the likelihood of a reactor trip and mitigating systems not being available. The inspectors noted that the finding did not lend itself to evaluation using Phase 2 of the significance determination process. This finding was evaluated by the inspectors, along with a senior reactor analyst, using the licensee's plant specific probabilistic risk assessment and determined that the risk increase of this finding was below the moderately risk significant threshold (by approximately a factor of 10). The inspectors determined, along with the senior reactor analyst, that the overall significance of this finding was very low (Section 1R13).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Nov 10, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Two examples of failure to follow procedures for working on the wrong unit**
Technical Specification 5.4.1.a requires that procedures be implemented for those procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Appendix A recommends procedures for shutdown of offsite power sources and surveillance procedures. Procedures OP J-2:III (Unit 1), "Startup Bank-Shutdown and Clearing," Revision 10A, and STP I-19-L62 (Unit 1), "Reactor Cavity Sump Level Channel LT-62 Calibration," Revision 2, partially implemented this requirement. Procedure OP J-2:III, step 6.1.2 required the user to open Unit 1 Switch 211-1, however, on October 23, 2000, the operator opened Switch 211-2, which inadvertently resulted in the loss of the startup transformer for Unit 2. Procedure STP I-19-L62, Step 8.4.1 required lifting the lead at Unit 1 Panel POCV1, TB-35, but on October 22, the technician lifted a lead in Unit 2 Panel POCV2, causing an inadvertent loss of the reactor coolant system leakage detection system in Unit 2. These examples of violation are described in the corrective action program as ARs A0517849 and A0517720.
Inspection Report# : 2000014*(pdf)*

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# Mitigating Systems

**Significance:** G Jul 11, 2002
Identified By: NRC
Item Type: FIN Finding
**Grounding resistor vulnerability**
The plant electrical distribution consisted of a design where the three redundant 4160 V safety buses and a non-safety bus were supplied from a common transformer winding during both normal and emergency operation. The 4160 V buses were interconnected by conductors so that a voltage disturbance on any part of the system would affect the entire system. The system had a high resistance grounding design to limit the magnitude of ground faults and to enable continued operation of a faulted load. The grounding resistor admits sufficient fault current to prevent severe over-voltages that could occur. However, if the grounding resistor developed an open circuit, the entire system would be susceptible to over-voltage. The licensee was periodically checking the continuity, but not the actual resistance of the grounding resistors and, thus, assumptions in the design were not being verified. The licensee issued Action Request A0561002 to evaluate the preventive maintenance program of the high resistance grounding program. This issue did not involve a violation of NRC requirements, but was considered to be a finding because it revealed a vulnerability in the licensee's design and maintenance that could result in a safety problem. However, the finding was determined to be of very low safety significance because there was no evidence that the grounding resistor had ever been degraded and that the probability of a grounding resistor failure in combination with a sparking ground fault was very small.
Inspection Report# : 2002006*(pdf)*

**Significance:** G May 31, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**The failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786.**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786. The licensee, contrary to the procedural requirements, placed the issue in a process to validate the initial perception that diesel fuel oil tank capacity would meet design requirements. The licensee documented on July 19, 2001, that Calculation M-786 had not been updated with regard to changes that would affect diesel fuel usage in the Technical Specifications, Design Criteria Memorandum, the Final Safety Analysis Report

Update, and the Emergency Operating Procedures. The licensee determined that such changes could have an adverse impact on the design and licensing basis related to adequate diesel fuel oil storage. The issue was determined to be of very low risk significance during Phase 1 of the NRC Significance Determination Process, because the Calculation M-786 was found to be conservative with respect to diesel generator loads and, therefore, the diesels remained operable. The failure to adequately address operability of potentially nonconforming conditions, if left uncorrected, could become a more significant safety concern, therefore, the issue was determined to be more than minor. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0553285. (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance: TBD** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**The installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box.**
The inspectors identified a finding with respect to the placement of ventilation louvers on 12 kV grounding transformer fuse boxes. On August 4, 2001, Units 1 and 2 experienced a loss of startup power as a result of multiple electrical faults in Startup Transformer 1-1 Grounding Transformer Fuse Box. Nonconformance Report N0002130, "Loss of Unit 1 and 2 Startup Power," determined the primary cause of the electrical faults to be condensation inside the fuse box. The contributory cause of the event was the ventilation louver, which allowed outside (salty) air to be drawn into the fuse box. The inspectors' Phase 2 evaluation of this issue using the Significance Determination Process indicated a condition that was potentially greater than green. The inspectors determined that the installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box represented an actual impact on safety since the preferred offsite power was momentarily lost from both units. Subsequently, auxiliary power continued to supply power to plant loads during the loss of startup power, and diesel generators were also available to supply power to safety-related equipment. This issue will remain as an unresolved issue (URI 50-275; 323/2002-02-01) pending completion of the significance determination process (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Exceeding the licensed power limit due to a failure to follow procedures**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Procedures OP L-4, "Normal Operation at Power," Revision 39, Section 5.4 and OP B-9:I, "Primary Sampling System - Make Available and Place in Service," Revision 7, stated, in part, that when pressurizer steam space sampling to the volume control tank was initiated, two backup pressurizer heaters were to be placed in service. On December 28, 2001, operators initiated pressurizer steam space sampling to the volume control tank without placing two backup pressurizer heaters into service. This resulted in a dilution of the volume control tank that increased reactor power above 100 percent for approximately 2½ hours. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0546623. This violation was more than minor because it had credible impact on safety due to the unplanned change in reactivity. This issue was determined to be of very low safety significance (Green) because the reactivity addition was not of an appreciable amount to challenge the safety systems or operating limits, and operators were able to return reactor power to desired levels in a controlled manner.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform adequate postmaintenance test of a reactor protection system analog input card**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix

A. Regulatory Guide 1.33 lists procedures for surveillance tests. Procedure STP I-33, "Reactor Trip and Engineered Safety Feature Response Time Test," Revision 6, partially implemented this requirement and stated in Section 3.3.3.b that replacement of an Eagle-21 card required time response testing of the appropriate channels. Contrary to the above, the licensee replaced Card 2 of Rack 13 of the Unit 2 Eagle 21 system on September 18, 2001, but did not perform time response testing as a postmaintenance test and returned the component to service. This card affected reactor trip and safety injection setpoints for Loop 3 reactor coolant system temperature, pressurizer pressure, and pressurizer level. Upon discovery, the time response test was successfully performed on March 7, 2002. This event is described in the licensee's corrective action program, reference Action Request A0550656. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation was more than minor because it had credible impact on safety due to the card affecting several mitigating systems and actuations. This issue was determined to be of very low safety significance (Green) because when the post maintenance testing was conducted, the applicable channels passed.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to limit the proximity of transient equipment near safety-related systems due to seismic interaction concerns**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to adequately limit the proximity of transient equipment from safety-related systems that may be required during a seismic event. Technical Specification 5.4.1.a requires that written procedures be implemented for equipment control. Procedure AD4.ID3, "SISIP Housekeeping Activities," Revision 4A, Section 5.1.1, required that transient equipment not create a potential seismically induced system interaction. Contrary to the above, on January 14, 2002, the inspectors discovered an unsecured portable welding machine staged approximately 8 inches from the normal and Class 1 air supply lines for Unit 2 atmospheric dump Valve MS-2-PCV-21. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0547478. This violation was more than minor because there was a credible impact on safety because the atmospheric dump valve could not be remotely operated due to loss of air supply in a seismic event. This issue was determined to be of very low safety significance because the other three atmospheric dump valves on the steam generators could be used to adequately cool the reactor coolant system.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Licensee Restarted Unit 2 Before Recognizing Reactor Trip and Engineered Safety Features Actuation Associated with Lo-lo Steam Generator Water Level was Inoperable**
The failure to promptly identify and correct the steam generator narrow range water level-low low reactor trip system and engineered safety system instrumentation nonconservative setpoint bias following the Unit 2 manual reactor trip on February 9, 2002, is a violation of 10 CFR Part 50, Criterion XVI. The licensee's event review failed to recognize that an engineered safety feature, including a reactor trip, failed to actuate when required during a loss of feedwater event to Steam Generator 2-4. This failure resulted in the licensee restarting Unit 2 with the reactor trip and engineered safety system instrumentation inoperable, and in the operation of both units with the same instrumentation inoperable, in violation of Technical Specification 3.3.1. This issue is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (50-275; 323/2002-07-01). The licensee documented this deficiency in Action Request A0549031. The failure to promptly recognize inoperable trip and actuation functions and comply with Technical Specification requirements had a credible impact on safety. The resulting delays in an automatic reactor trip and engineered safety features actuations would have delayed the plant's response to a loss of feedwater event and reduced the water mass available for the heat sink function in the affected steam generator(s). Further, this deficiency had the potential to affect the integrity of the reactor coolant system boundary. A Phase 3 Significance Determination Process evaluation concluded that the issue had very low safety significance (Green). The finding represents a condition that

existed for 5-days. The significance of the steam generator narrow range water level-low low setpoint offset (bias) is reduced if feedwater flow is lost to two or more steam generators. Based on the short duration from the time a single steam generator would dryout (the limiting initiator is a loss of feedwater to a single generator) and actuation of auxiliary feedwater, the condition does not result in an appreciable increase in the probability of a steam generator tube rupture occurring. The licensee's analysis using the plant specific simulator showed that the engineered safety feature actuation and reactor trip on steam generator water level-low low would have initiated at or before steam generator dryout would occur. The reactor coolant system physical over pressure protective features (safety relief and power operated relief valves) should not be challenged and there were other protective trips in place (over temperature-delta temperature and over pressure delta-temperature) in place that would have protected the reactor coolant system and fuel integrity in the event a manual reactor trip is not initiated on a loss of feedwater flow to a steam generator [Sections 4OA2.a.(2) and 5].
Inspection Report# : 2002007*(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)*

**Significance:** G Aug 25, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Violation of 10 CFR 50 Appendix B, Criterion III for failure to implement design control measures for changes that impacted diesel fuel oil capacity calculations (Section 4OA7)**
Green. The licensee identified a failure to implement design control measures for changes to postaccident operations as described in the Final Safety Analysis Report Update. The licensee changed the loading sequence of the diesel engine generators as described in the Final Safety Analysis Report for several items but did not input these changes into the diesel fuel oil storage capacity calculations. This issue required significant revisions to the calculations to resolve the fuel oil storage requirement. The inspectors determined this to be a violation of 10 CFR 50, Appendix, Criterion III for failure to implement design control measures to changes to postaccident operations. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was entered into the corrective action program as AR A0540317. This issue could become a more significant safety concern if not corrected based on less than the required amount of diesel fuel oil onsite if additional revisions to the loading sequence occurred

without input to the fuel oil storage capacity requirements. The inspectors evaluated the issue using the Significance Determination Process Phase 1 worksheet. Each of the questions related to mitigating systems was answered no resulting in the issue screening out as having very low safety significance.
Inspection Report# : 2001006*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite power and a diesel engine generator inoperable simultaneously for approximately 7 hours, but did not take the required actions. Because of inadequate planning and procedure guidance, operators placed the load tap changer for Unit 2 Startup Transformer 2-1 to an inappropriate tap setting, but did not declare Startup Transformer 2-1 inoperable. These actions, coupled with 500 kV auxiliary power inoperable for breaker cubicle inspections, and Diesel Generator 2-2 inoperable because of degraded wiring, rendered all three emergency power sources for Vital Bus H inoperable in excess of the Technical Specification 3.0.3 allowed outage time of 1 hour. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. This item was placed in the corrective action program as Action Request A0528007. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that this finding had potential impact because a total loss of Unit 2 Vital Bus H would have resulted from several initiating events, including a reactor trip. (Vital Busses F and G and their associated diesel engines remained operable.) This finding involved three mitigating systems, the 500 kV Auxiliary Transformer, the 230 kV Startup Transformer, and Diesel Engine Generator 2-2. Using Phase 1 of the Significance Determination Process, this item could be considered an item in which systems were unavailable in excess of the Technical Specification action statement (3.8.1.1), requiring a Phase 2 Significance Determination Process evaluation. However, the inspector noted that although Startup Transformer 2-1 was inoperable as defined by its Technical Specification 3.8.1.1 function to automatically pick up loads following a loss of 500 kV offsite power, operators could have easily recovered Startup Transformer 2-1 and returned the load tap changer to automatic control. Thus, Startup Transformer 2-1 is considered available for most accident sequences (except those involving loss of the startup transformer). Auxiliary power and Diesel Engine Generator 2-2 were readily recoverable. This violation was determined to be of very low risk significance, as evaluated under the transient and loss of offsite power Significance Determination Process worksheets and as independently verified by an NRC senior reactor analyst (Green) (Section 1R13).
Inspection Report# : 2001003*(pdf)*

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: FIN Finding
**Insufficient integration of training and new instrumentation for Mid-loop operations**
The inspectors identified that the licensee had not properly integrated the instrumentation, training and procedures relied on for mid-loop operation. Specifically, the inspectors noted that: several issues occurred with respect to instrumentation that resulted in operator distractions during mid-loop operations; the licensee did not perform full dynamic simulator training on mid-loop operations; and, mid-loop procedures were not enhanced to address the newly installed reactor vessel level instrumentation and associated alarms. The failure to adequately address instrumentation, training and procedures for the monitoring of mid-loop operations was determined to be a cross-cutting issue. The inspectors evaluated this finding using the significance determination process. Specifically, Manual Chapter 0609, Appendix G, Shutdown Operations Significance Determination Process, was considered. The finding did not result in a loss of control as defined by Appendix G, TABLE 1, Losses of Control for Loss of Thermal Margin or Loss of Level PWRs. The inspectors, along with a senior reactor analyst reviewed PWR Hot Shutdown operation with a time to core boiling less than 2 hours. The core heat removal guidelines and inventory control guidelines were considered. Item II of the Core Heat Removal Guidelines, A. Instrumentation specifying 2 independent pressurizer level instruments with a Hi/Lo alarm or level deviation annunciator was determined to be impacted requiring a Phase 2 evaluation. The senior reactor analyst reviewed the actual conditions, observed the control room and plant simulator instrumentation and

discussed the finding with the cognizant inspectors who observed the mid-loop operation. The inspectors determined, along with the senior reactor analyst, that adequate reactor vessel level was available such that the overall significance of this finding was very low (Section 1R20.1).
Inspection Report# : 2001006*(pdf)*

**Significance:** G Jan 26, 2001
Identified By: NRC
Item Type: FIN Finding
**Failure to properly evaluate a maintenance preventable functional failure because of incorrectly set corrective action system defaults**
The corrective action system defaults were incorrectly applied such that maintenance rule reviews to determine if a maintenance preventable functional failure occurred would be bypassed. The inspectors identified that the maintenance preventable functional failure review did not occur when Unit 2 Startup Transformer 2-1 was inadvertently de-energized for maintenance, instead of Unit 1 Startup Transformer 1-1, and the action request was closed. The licensee subsequently determined that a maintenance preventable functional failure had occurred; however, the system would not be placed into goal setting following a human performance error. The inspectors evaluated this issue using the Significance Determination Process. The inspectors noted that Startup Transformer 2-1 remained inoperable for less than 1 hour and the Unit 2 diesel engine generators started as required. The condition did not result in an increase to an initiating event frequency. The offsite power supply, as a mitigating system, was unavailable for a short period of time with the respective diesel engine generators available. Therefore, adequate sources of power remained available to mitigate a reactor trip or loss of offsite power event. The inspectors determined that this issue had very low risk significance (Green)
Inspection Report# : 2001002*(pdf)*

**Significance: N/A** Aug 24, 2000
Identified By: NRC
Item Type: FIN Finding
**Evaluation of Scrams w/Loss of Normal Heat Removal white performance indicator**
The inspectors performed a supplemental inspection to examine a change from green to white in the Scrams With Loss of Normal Heat Removal performance indicator. This change in performance resulted from Unit 2 experiencing three scrams with loss of normal heat removal over the previous 12 quarters. Following each event, NRC had evaluated operator response, plant and equipment response, and immediate corrective actions. During this supplemental inspection, performed in accordance with Procedure 95001, the inspectors evaluated the adequacy of the root cause evaluation and long-term corrective actions for each individual event. The inspectors also evaluated the effectiveness of the licensee review into the collective events. The inspectors determined that the licensee had performed comprehensive root cause evaluations and corrective actions for each individual scram and the events collectively. The licensee determined that one scram occurred because condensate/feedwater flow problems were exacerbated by a control circuit problem (poor design and dirty slide wire) in Valve TCV-23, generator hydrogen cold gas temperature control, combined with throttling Valve CND-2-165, steam jet air ejector outlet isolation. The licensee did not identify a definite root cause for the event initiator. Operators initiated the other two scrams because debris in the circulating water system intake had increased the differential pressure across the traveling screens above the setpoint that required them to be secured prior to being damaged. The licensee determined that the onset of ocean storms, combined with the end of the growing season (peak amounts of marine growth), established conditions that exceeded the ability of the traveling screens to remove the marine growth and remain within acceptable operating parameters. The licensee established plans to upgrade the traveling screens, formalized their process for predicting conditions affecting the ability of the intake components to remove marine growth, and initiated efforts to raise the turbine trip/reactor trip setpoint to optimize withstanding this condition yet conducting an orderly shutdown of the plants. The inspectors concluded that the licensee addressed the Scrams With Loss of Normal Heat Removal for Unit 2 in an acceptable manner. No further evaluations are required. This is in accordance with the guidance in IMC 0305, "Operating Reactor Assessment Program."
Inspection Report# : 2000013*(pdf)*

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G May 06, 2000
Identified By: NRC
Item Type: FIN Finding
**Multiple Control Room Light Socket Failures**
Green. On August 1, 1999, the licensee reported a design weakness in the control room lamp sockets in both units resulted in multiple failures during 1998 and 1999. The failure of lamp sockets could have resulted in shorting the control power to affected safety-related components during a seismic event. The affected light sockets were replaced. The licensee performed a detailed risk analysis and concluded that the increased risk was small. Simultaneous failure of multiple sockets in a manner that would result in electrical shorts that prevented function of all of the affected components was considered highly unlikely. An NRC Senior Reactor Analyst reviewed the licensee's seismic risk analysis and concluded that the analysis was adequate to demonstrate that the increased risk (delta core damage and large early release frequencies) was small and of very low risk significance (Closes LER 1/2-99-007)
Inspection Report# : 2000006*(pdf)*

**Significance:** G Apr 07, 2000
Identified By: NRC
Item Type: FIN Finding
**Degraded 1-hour fire-rated ceiling in Fire Area 4A and degraded 2-hour fire-rated barrier between Fire Areas 4A and 4B.**
The team identified that the 1-hour fire-rated ceiling in Fire Area 4A (counting and chemistry laboratory) and the 2-hour fire-rated barrier between Fire Areas 4A and 4B (radiologically controlled area access) were degraded. Specifically, the team identified that the 1-hour fire-rated ceiling in the chemistry laboratory contained holes, non-fire-rated dampers, and gaps around the lighting fixtures. The NRC relied on the 1-hour fire rating of this ceiling as a basis for granting an exemption from the requirement to enclose redundant trains of safe shutdown equipment in a 1-hour fire-rated enclosure as described in 10 CFR Part 50, Appendix R, Section III.G.2.c. In addition, the team observed concrete spalling, holes, and a non-fire-rated penetration in the 2-hour fire-rated barrier between Fire Areas 4A and 4B . Upon further review, the team found that the licensee had previously identified most of these conditions and had taken appropriate compensatory measures. Although the team identified additional minor discrepancies, no additional compensatory measures were warranted. The conditions not previously identified by the licensee were entered into the licensee's corrective action program as Action Requests A05050857, A0505861, and A0505892. This issue was

evaluated using the significance determination process and was determined to be of low risk significance, because barrier degradation was moderate; detection, automatic suppression, and manual suppression met the conditions of the licensing basis for Fire Areas 4A and 4B; and a safe shutdown path remained
Inspection Report# : 2000003*(pdf)*

**Significance:** G Mar 07, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to evaluate/ restrain a portable cart next to safety piping**
The licensee placed a top-heavy portable load center near component cooling water piping and failed to evaluate the condition. The portable load center was not restrained such that it would not strike and potentially damage the component cooling water piping. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. A similar occurrence was discussed in Inspection Report 50-275; 323/9912. This item was placed in the corrective action program as Action Request A0506658. The inspectors assessed the risk significance of this item using the significance determination process. The inspectors determined that this issue was of very low risk significance, and thus was a Green finding. The inspectors used the significance determination process Phase I worksheet for seismic, fire, flooding, and severe weather screening criteria and determined that the portable load center would not damage more than one train of component cooling water, thus the item was screened to Green. The failure to implement a procedure for seismic interaction was a violation of Technical Specification 6.8.1.a..
Inspection Report# : 2000007*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** G May 12, 2000
Identified By: NRC
Item Type: FIN Finding
**Critique failed to identify facility activation not completed in accordance with procedures**
The inspectors identified that the critique process failed to identify that two emergency response facilities were not activated in accordance with the emergency response plan and implementing procedures. The licensee entered the issue into its corrective action system as Action Request A0507922. This finding was determined to have very low risk significance because the affected planning standard was not risk significant (Section 1EP1).
Inspection Report# : 2000007*(pdf)*

**Significance:** G Feb 17, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Unauthorized person reviewed emergency preparedness program (Closes URI 0002-02)**
The inspectors identified that a member of the emergency planning staff inappropriately reviewed part of the emergency preparedness program. 10 CFR 50.54(t) requires that emergency preparedness program elements be evaluated by individuals not responsible for program implementation. This was a violation of 10 CFR 50.54(t) for failure to conduct an appropriate review of the emergency preparedness program which is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered the item into its corrective action system as Action Request A0503012.
Inspection Report# : 2000007*(pdf)*

# Occupational Radiation Safety

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

**Significance:** G Apr 30, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to survey a high radiation area**
10 CFR 20.1501(a) requires that each licensee shall make or cause to be made, surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and are reasonable under the circumstances to evaluate the radiation levels and the potential radiological hazards. On April 30, 2001, the licensee identified a high radiation area above the 2-1 Deborating Demineralize resin fill connection access port which had dose rates as high as 170 millirems/hour at 30 centimeters. The licensee's investigation determined that the conditions existed for as long as 24 hours. See Action Request A0530296. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Mar 08, 2001
Identified By: Licensee
Item Type: NCV NonCited Violation
**Failure to lock a high radiation area with dose rates greater then 1 rem/hour**
Technical Specification 5.7.2 states that for high radiation areas with dose rates greater than 1.0 rem/hour at 30 centimeters from the radiation source, each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry. On March 8, 2001, the keycard reader door to containment was not locked, allowing potential unauthorized entrance to high-high radiation areas within the containment building. See Action Request A0527032. This is being treated as a noncited violation. Through the use of the Occupational Radiation Safety Significance Determination Process, the safety significance of this finding was determined to be very low because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001005*(pdf)*

**Significance:** G Feb 16, 2001
Identified By: NRC

Item Type: NCV NonCited Violation

**Failure to survey**

On February 13, 2001, during a walkdown of the radiological effluent release monitors and tanks located on Elevation 64 foot of the auxiliary building, the inspectors identified a radiation area and a high radiation area near the Spent Resin Tank Filters that were not surveyed and controlled. Surveys revealed that general area radiation levels ranged from 7 millirems per hour to as high as 500 millirems per hour. 10 CFR 20.1501(a) states, in part, that each licensee shall make or cause to be made surveys that are reasonable under the circumstances to evaluate the extent of the radiation levels and the potential radiological hazards. The failure to survey the areas surrounding the Spent Resin Tank Filters to evaluate the extent of the radiation levels and potential radiological hazards is a violation of 10 CFR 20.1501. This violation is in the licensee's corrective action program as Action Request AO 525568. This issue was determined to have very low safety significance, because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised.

Inspection Report# : 2000016*(pdf)*

**Significance:** G Nov 10, 2000

Identified By: NRC

Item Type: NCV NonCited Violation

**Violation of TS 5.7.1.e for entering High Radiation Areas without Knowledge of Dose Rates**

Technical Specification 5.7.1.e requires that entry into a high radiation area be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. On October 10, 2000, four workers in two work groups entered a high radiation area without obtaining the dose rate information, as described in the corrective action program, reference ARs A0516173 and A0516174.

Inspection Report# : 2000014*(pdf)*

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# Public Radiation Safety

**Significance:** G Jan 12, 2001

Identified By: Licensee

Item Type: NCV NonCited Violation

**Failure to control radioactive materials**

Technical Specification 5.4.1 requires procedures for the control of radioactivity. Section 7.1.1 of Procedure RCP D-614, "Release of Materials From the Radiologically Controlled Area," Revision 5A, states in part, that all material released from the radiologically controlled area shall have no detectable licensed radioactivity. On October 12, 1999, and August 29, 2000, detectable licensed radioactivity was released from the radiologically controlled area, as described in the licensee's corrective action program, reference Action Requests A0494102 and A0513515.

Inspection Report# : 2000016*(pdf)*

**Significance:** G Sep 20, 2000

Identified By: NRC

Item Type: FIN Finding

**Licensee failed to follow waste disposal facility site criteria requirement.**

On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10 CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a

"green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

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# Physical Protection

**Significance:** G Dec 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Control Personnel Access at the Plant Wharehouse**
The licensee's secondary alarm station operator failed to use closed-circuit television cameras to determine that the warehouse access control security officer was present prior to opening the protected area personnel access door for an NRC inspector in the plant warehouse. In addition, the operator failed to determine that the security officer was not under duress prior to opening the personnel access door. The failure to adequately control personnel access was a violation of Paragraph 3.2.1.1 of the Physical Security Plan (Revision 18, Change 18). This violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy (275; 323/0015-01). The licensee entered the violation into the corrective action program as Action Request A0522821. This issue was determined to be of very low safety significance (green) by the significance determination process because there were not greater than two similar findings in the last four quarters
Inspection Report# : 2000015*(pdf)*

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# Miscellaneous

**Significance: N/A** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The licensee was effective at identifying problems and placing them into the corrective action program with one exception in the area of operability determinations. Occasionally an operability determination being reviewed by engineering was not timely. For example, the licensee failed to identify and evaluate how differential pressure affected steam generator instrumentation and its affect on operability prior to starting the plant following a trip with unusual steam generator level indications. The licensee appropriately determined the extent of evaluation of individual problems and prioritized the schedule for implementation of corrective actions to address the safety significant issues. In general, corrective actions, when specified, were effective and were implemented in a timely manner. The licensee performed effective audits and assessments. Based on the interviews conducted during this inspection, workers at the site felt free to input safety issues into the problem identification and resolution program.
Inspection Report# : 2002002*(pdf)*

**Significance: N/A** Apr 05, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The team determined that a critical opportunity was missed to promptly identify and correct a risk significant condition adverse to quality involving a nonconservative safety features set point. The licensee's post trip event review process did not ensure that the Unit 2 plant response to a loss of feedwater flow to Steam Generator 2-4 was appropriate in that the steam generator level lo-lo engineered safety features and automatic reactor trip actuations did not occur when

required.
Inspection Report# : 2002007*(pdf)*

**Significance: N/A** Aug 25, 2001
Identified By: NRC
Item Type: FIN Finding
**Technical Specification limit for dose equivalent iodine was nonconservative**
The inspectors identified that the licensee had not taken action to docket a justification and schedule to correct a nonconservative Technical Specification. On March 4, 2000, the licensee identified that the reactor coolant system activity Technical Specification limit for dose equivalent iodine was nonconservative. Engineers determined that instead of the Technical Specification limit of 1 µci/g, the licensee must control reactor coolant system activity to .71 µci/g when normal letdown was in service and .47 µci/g while excess letdown was in service. The licensee implemented administrative controls to prevent exceeding the new limits, but took no action to docket a justification and schedule to correct Technical Specification 3.4.12 until prompted by the inspectors in August of 2001. This item was entered into the corrective action program as Action Request A0540317. The safety significance of the finding was evaluated initially using Manual Chapter 0610 Group 2 Questions for Reactor Safety-Initiating Events, Mitigating Systems, and Barrier Integrity. A no color determination was made based on the finding was determined not to: cause or increase the frequency of an initiating event; affect the operability, availability, reliability, or function of a system or train in a mitigating system; affect the integrity of fuel cladding, the reactor coolant system, reactor containment or control room envelope; or, involve degraded conditions that could concurrently influence any mitigation equipment and an initiating event (Section 4OA1).
Inspection Report# : 2001006*(pdf)*

**Significance: N/A** Mar 29, 2001
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The inspectors concluded that the implementation of the corrective action program at Diablo Canyon was acceptable. The Diablo Canyon staff adequately identified problems and entered them into the corrective action system. The overall corrective action backlog and the specific engineering and maintenance backlogs appeared to be appropriately prioritized and adequately managed. There was a low threshold for initiation of deficiency documents, and they were properly classified at the correct significance level. The depth of the root cause analysis for problems were appropriate. Corrective actions were generally adequate and completed in a timely manner, and as necessary prevented recurrence.
Inspection Report# : 2001004*(pdf)*

Last modified : December 02, 2002

# Diablo Canyon 2

## Initiating Events

## Mitigating Systems

**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)*

**Significance:** G Jul 11, 2002
Identified By: NRC
Item Type: FIN Finding
**Grounding resistor vulnerability**
The plant electrical distribution consisted of a design where the three redundant 4160 V safety buses and a non-safety bus were supplied from a common transformer winding during both normal and emergency operation. The 4160 V buses were interconnected by conductors so that a voltage disturbance on any part of the system would affect the entire system. The system had a high resistance grounding design to limit the magnitude of ground faults and to enable continued operation of a faulted load. The grounding resistor admits sufficient fault current to prevent severe over-voltages that could occur. However, if the grounding resistor developed an open circuit, the entire system would be susceptible to over-voltage. The licensee was periodically checking the continuity, but not the actual resistance of the grounding resistors and, thus, assumptions in the design were not being verified. The licensee issued Action Request A0561002 to evaluate the preventive maintenance program of the high resistance grounding program. This issue did not involve a violation of NRC requirements, but was considered to be a finding because it revealed a vulnerability in the licensee's design and maintenance that could result in a safety problem. However, the finding was determined to be of very low safety significance because there was no evidence that the grounding resistor had ever been degraded and that the probability of a grounding resistor failure in combination with a sparking ground fault was very small.
Inspection Report# : 2002006*(pdf)*

**Significance: TBD** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**The installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box.**
The inspectors identified a finding with respect to the placement of ventilation louvers on 12 kV grounding transformer fuse boxes. On August 4, 2001, Units 1 and 2 experienced a loss of startup power as a result of multiple electrical faults in Startup Transformer 1-1 Grounding Transformer Fuse Box. Nonconformance Report N0002130, "Loss of Unit 1 and 2 Startup Power," determined the primary cause of the electrical faults to be condensation inside the fuse box. The contributory cause of the event was the ventilation louver, which allowed outside (salty) air to be drawn into the fuse box. The inspectors' Phase 2 evaluation of this issue using the Significance Determination Process indicated

a condition that was potentially greater than green. The inspectors determined that the installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box represented an actual impact on safety since the preferred offsite power was momentarily lost from both units. Subsequently, auxiliary power continued to supply power to plant loads during the loss of startup power, and diesel generators were also available to supply power to safety-related equipment. This issue will remain as an unresolved issue (URI 50-275; 323/2002-02-01) pending completion of the significance determination process (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance:** G May 31, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**The failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786.**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786. The licensee, contrary to the procedural requirements, placed the issue in a process to validate the initial perception that diesel fuel oil tank capacity would meet design requirements. The licensee documented on July 19, 2001, that Calculation M-786 had not been updated with regard to changes that would affect diesel fuel usage in the Technical Specifications, Design Criteria Memorandum, the Final Safety Analysis Report Update, and the Emergency Operating Procedures. The licensee determined that such changes could have an adverse impact on the design and licensing basis related to adequate diesel fuel oil storage. The issue was determined to be of very low risk significance during Phase 1 of the NRC Significance Determination Process, because the Calculation M-786 was found to be conservative with respect to diesel generator loads and, therefore, the diesels remained operable. The failure to adequately address operability of potentially nonconforming conditions, if left uncorrected, could become a more significant safety concern, therefore, the issue was determined to be more than minor. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0553285. (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to limit the proximity of transient equipment near safety-related systems due to seismic interaction concerns**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to adequately limit the proximity of transient equipment from safety-related systems that may be required during a seismic event. Technical Specification 5.4.1.a requires that written procedures be implemented for equipment control. Procedure AD4.ID3, "SISIP Housekeeping Activities," Revision 4A, Section 5.1.1, required that transient equipment not create a potential seismically induced system interaction. Contrary to the above, on January 14, 2002, the inspectors discovered an unsecured portable welding machine staged approximately 8 inches from the normal and Class 1 air supply lines for Unit 2 atmospheric dump Valve MS-2-PCV-21. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0547478. This violation was more than minor because there was a credible impact on safety because the atmospheric dump valve could not be remotely operated due to loss of air supply in a seismic event. This issue was determined to be of very low safety significance because the other three atmospheric dump valves on the steam generators could be used to adequately cool the reactor coolant system.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Exceeding the licensed power limit due to a failure to follow procedures**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Procedures OP L-4, "Normal Operation at Power," Revision 39, Section 5.4 and OP B-9:I, "Primary Sampling System - Make Available and Place in Service," Revision 7, stated, in part, that when pressurizer steam space sampling to the volume control tank was initiated, two backup pressurizer heaters were to be placed in service. On December 28, 2001, operators initiated pressurizer steam space sampling to the volume control tank without placing two backup pressurizer heaters into service. This resulted in a dilution of the volume control tank that increased reactor power above 100 percent for approximately 2½ hours. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0546623. This violation was more than minor because it had credible impact on safety due to the unplanned change in reactivity. This issue was determined to be of very low safety significance (Green) because the reactivity addition was not of an appreciable amount to challenge the safety systems or operating limits, and operators were able to return reactor power to desired levels in a controlled manner.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform adequate postmaintenance test of a reactor protection system analog input card**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33 lists procedures for surveillance tests. Procedure STP I-33, "Reactor Trip and Engineered Safety Feature Response Time Test," Revision 6, partially implemented this requirement and stated in Section 3.3.3.b that replacement of an Eagle-21 card required time response testing of the appropriate channels. Contrary to the above, the licensee replaced Card 2 of Rack 13 of the Unit 2 Eagle 21 system on September 18, 2001, but did not perform time response testing as a postmaintenance test and returned the component to service. This card affected reactor trip and safety injection setpoints for Loop 3 reactor coolant system temperature, pressurizer pressure, and pressurizer level. Upon discovery, the time response test was successfully performed on March 7, 2002. This event is described in the licensee's corrective action program, reference Action Request A0550656. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation was more than minor because it had credible impact on safety due to the card affecting several mitigating systems and actuations. This issue was determined to be of very low safety significance (Green) because when the post maintenance testing was conducted, the applicable channels passed.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Licensee Restarted Unit 2 Before Recognizing Reactor Trip and Engineered Safety Features Actuation Associated with Lo-lo Steam Generator Water Level was Inoperable**
The failure to promptly identify and correct the steam generator narrow range water level-low low reactor trip system and engineered safety system instrumentation nonconservative setpoint bias following the Unit 2 manual reactor trip on February 9, 2002, is a violation of 10 CFR Part 50, Criterion XVI. The licensee's event review failed to recognize that an engineered safety feature, including a reactor trip, failed to actuate when required during a loss of feedwater event to Steam Generator 2-4. This failure resulted in the licensee restarting Unit 2 with the reactor trip and engineered safety system instrumentation inoperable, and in the operation of both units with the same instrumentation inoperable, in violation of Technical Specification 3.3.1. This issue is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (50-275; 323/2002-07-01). The licensee documented this deficiency in Action Request A0549031. The failure to promptly recognize inoperable trip and actuation functions and comply with Technical Specification requirements had a credible impact on safety. The resulting delays in an automatic reactor trip and engineered safety features actuations would have delayed the plant's response to a loss of feedwater event and reduced the water mass available for the heat sink function in the affected steam generator(s). Further, this deficiency had the potential to affect the integrity of the reactor coolant system boundary. A Phase 3 Significance Determination Process evaluation concluded that the issue had very low safety significance (Green). The finding represents a condition that existed for 5-days. The significance of the steam generator narrow range water level-low low setpoint offset (bias) is reduced if feedwater flow is lost to two or more steam generators. Based on the short duration from the time a single steam generator would dryout (the limiting initiator is a loss of feedwater to a single generator) and actuation of auxiliary feedwater, the condition does not result in an appreciable increase in the probability of a steam generator tube rupture occurring. The licensee's analysis using the plant specific simulator showed that the engineered safety feature actuation and reactor trip on steam generator water level-low low would have initiated at or before steam generator dryout would occur. The reactor coolant system physical over pressure protective features (safety relief and power operated relief valves) should not be challenged and there were other protective trips in place (over temperature-delta temperature and over pressure delta-temperature) in place that would have protected the reactor coolant system and fuel integrity in the event a manual reactor trip is not initiated on a loss of feedwater flow to a steam generator [Sections 4OA2.a.(2) and 5].

Inspection Report# : 2002007*(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

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## Occupational Radiation Safety

**Significance:** G Jan 08, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Airborne radiation monitor inoperable when required during work in spent fuel pool**
Technical Specification 5.4.1.a. requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Attachment 10.7 of Procedure RCP D-200, "Writing Radiation Work Permits," Revision 22A, states, in part, that radiation protection shall ensure that a constant air monitor is in operation in the fuel handling building while underwater work is being performed. On August 29, 2001, the licensee identified that underwater work was being performed in Unit 1 spent fuel pool without the required constant airborne monitor in operation. This event is described in the licensee's corrective action program, reference Action Request A0539922. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised.
Inspection Report# : 2001009*(pdf)*

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## Public Radiation Safety

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## Physical Protection

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## Miscellaneous

**Significance: N/A** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The licensee was effective at identifying problems and placing them into the corrective action program with one exception in the area of operability determinations. Occasionally an operability determination being reviewed by engineering was not timely. For example, the licensee failed to identify and evaluate how differential pressure affected steam generator instrumentation and its affect on operability prior to starting the plant following a trip with unusual steam generator level indications. The licensee appropriately determined the extent of evaluation of individual problems and prioritized the schedule for implementation of corrective actions to address the safety significant issues. In general, corrective actions, when specified, were effective and were implemented in a timely manner. The licensee performed effective audits and assessments. Based on the interviews conducted during this inspection, workers at the site felt free to input safety issues into the problem identification and resolution program.
Inspection Report# : 2002002*(pdf)*

**Significance: N/A** Apr 05, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The team determined that a critical opportunity was missed to promptly identify and correct a risk significant condition adverse to quality involving a nonconservative safety features set point. The licensee's post trip event review process did not ensure that the Unit 2 plant response to a loss of feedwater flow to Steam Generator 2-4 was appropriate in that the steam generator level lo-lo engineered safety features and automatic reactor trip actuations did not occur when required.
Inspection Report# : 2002007*(pdf)*

Last modified : March 25, 2003

# Diablo Canyon 2
# 1Q/2003 Plant Inspection Findings

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## Initiating Events

**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:** 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:** 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)*

**Significance:** G Jul 11, 2002
Identified By: NRC
Item Type: FIN Finding
**Grounding resistor vulnerability**
The plant electrical distribution consisted of a design where the three redundant 4160 V safety buses and a non-safety bus were supplied from a common transformer winding during both normal and emergency operation. The 4160 V buses were interconnected by conductors so that a voltage disturbance on any part of the system would affect the entire system. The system had a high resistance grounding design to limit the magnitude of ground faults and to enable continued operation of a faulted load. The grounding resistor admits sufficient fault current to prevent severe over-voltages that could occur. However, if the grounding resistor developed an open circuit, the entire system would be susceptible to over-voltage. The licensee was periodically checking the continuity, but not the actual resistance of the grounding resistors and, thus, assumptions in the design were not being verified. The licensee issued Action Request A0561002 to evaluate the preventive maintenance program of the high resistance grounding program. This issue did not involve a violation of NRC requirements, but was considered to be a finding because it revealed a vulnerability in the licensee's design and maintenance that could result in a safety problem. However, the finding was determined to be of very low safety significance because there was no evidence that the grounding resistor had ever been degraded and that the probability of a grounding resistor failure in combination with a sparking ground fault was very small.
Inspection Report# : 2002006*(pdf)*

**Significance: TBD** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**The installation of the ventilation louver, and the subsequent electrical fault associated with Startup**

**Transformer 1-1 Grounding Transformer Fuse Box.**
The inspectors identified a finding with respect to the placement of ventilation louvers on 12 kV grounding transformer fuse boxes. On August 4, 2001, Units 1 and 2 experienced a loss of startup power as a result of multiple electrical faults in Startup Transformer 1-1 Grounding Transformer Fuse Box. Nonconformance Report N0002130, "Loss of Unit 1 and 2 Startup Power," determined the primary cause of the electrical faults to be condensation inside the fuse box. The contributory cause of the event was the ventilation louver, which allowed outside (salty) air to be drawn into the fuse box. The inspectors' Phase 2 evaluation of this issue using the Significance Determination Process indicated a condition that was potentially greater than green. The inspectors determined that the installation of the ventilation louver, and the subsequent electrical fault associated with Startup Transformer 1-1 Grounding Transformer Fuse Box represented an actual impact on safety since the preferred offsite power was momentarily lost from both units. Subsequently, auxiliary power continued to supply power to plant loads during the loss of startup power, and diesel generators were also available to supply power to safety-related equipment. This issue will remain as an unresolved issue (URI 50-275; 323/2002-02-01) pending completion of the significance determination process (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance:** G May 31, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**The failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786.**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to initiate an operability assessment for a nonconforming condition associated with adequate fuel storage capacity to address increases of diesel generator loads in Calculation M-786. The licensee, contrary to the procedural requirements, placed the issue in a process to validate the initial perception that diesel fuel oil tank capacity would meet design requirements. The licensee documented on July 19, 2001, that Calculation M-786 had not been updated with regard to changes that would affect diesel fuel usage in the Technical Specifications, Design Criteria Memorandum, the Final Safety Analysis Report Update, and the Emergency Operating Procedures. The licensee determined that such changes could have an adverse impact on the design and licensing basis related to adequate diesel fuel oil storage. The issue was determined to be of very low risk significance during Phase 1 of the NRC Significance Determination Process, because the Calculation M-786 was found to be conservative with respect to diesel generator loads and, therefore, the diesels remained operable. The failure to adequately address operability of potentially nonconforming conditions, if left uncorrected, could become a more significant safety concern, therefore, the issue was determined to be more than minor. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the corrective action program as Action Request A0553285. (Section 4OA2).
Inspection Report# : 2002002*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to limit the proximity of transient equipment near safety-related systems due to seismic interaction concerns**
The inspectors identified a violation of Technical Specification 5.4.1.a for the failure to adequately limit the proximity of transient equipment from safety-related systems that may be required during a seismic event. Technical Specification 5.4.1.a requires that written procedures be implemented for equipment control. Procedure AD4.ID3, "SISIP Housekeeping Activities," Revision 4A, Section 5.1.1, required that transient equipment not create a potential seismically induced system interaction. Contrary to the above, on January 14, 2002, the inspectors discovered an unsecured portable welding machine staged approximately 8 inches from the normal and Class 1 air supply lines for Unit 2 atmospheric dump Valve MS-2-PCV-21. This violation is being treated as a noncited violation consistent with

Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0547478. This violation was more than minor because there was a credible impact on safety because the atmospheric dump valve could not be remotely operated due to loss of air supply in a seismic event. This issue was determined to be of very low safety significance because the other three atmospheric dump valves on the steam generators could be used to adequately cool the reactor coolant system.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Exceeding the licensed power limit due to a failure to follow procedures**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Procedures OP L-4, "Normal Operation at Power," Revision 39, Section 5.4 and OP B-9:I, "Primary Sampling System - Make Available and Place in Service," Revision 7, stated, in part, that when pressurizer steam space sampling to the volume control tank was initiated, two backup pressurizer heaters were to be placed in service. On December 28, 2001, operators initiated pressurizer steam space sampling to the volume control tank without placing two backup pressurizer heaters into service. This resulted in a dilution of the volume control tank that increased reactor power above 100 percent for approximately 2½ hours. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0546623. This violation was more than minor because it had credible impact on safety due to the unplanned change in reactivity. This issue was determined to be of very low safety significance (Green) because the reactivity addition was not of an appreciable amount to challenge the safety systems or operating limits, and operators were able to return reactor power to desired levels in a controlled manner.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 11, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to perform adequate postmaintenance test of a reactor protection system analog input card**
Technical Specification 5.4.1.a requires the implementation of procedures listed in Regulatory Guide 1.33, Appendix A. Regulatory Guide 1.33 lists procedures for surveillance tests. Procedure STP I-33, "Reactor Trip and Engineered Safety Feature Response Time Test," Revision 6, partially implemented this requirement and stated in Section 3.3.3.b that replacement of an Eagle-21 card required time response testing of the appropriate channels. Contrary to the above, the licensee replaced Card 2 of Rack 13 of the Unit 2 Eagle 21 system on September 18, 2001, but did not perform time response testing as a postmaintenance test and returned the component to service. This card affected reactor trip and safety injection setpoints for Loop 3 reactor coolant system temperature, pressurizer pressure, and pressurizer level. Upon discovery, the time response test was successfully performed on March 7, 2002. This event is described in the licensee's corrective action program, reference Action Request A0550656. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation was more than minor because it had credible impact on safety due to the card affecting several mitigating systems and actuations. This issue was determined to be of very low safety significance (Green) because when the post maintenance testing was conducted, the applicable channels passed.
Inspection Report# : 2001011*(pdf)*

**Significance:** G Apr 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation

**Licensee Restarted Unit 2 Before Recognizing Reactor Trip and Engineered Safety Features Actuation Associated with Lo-lo Steam Generator Water Level was Inoperable**
The failure to promptly identify and correct the steam generator narrow range water level-low low reactor trip system and engineered safety system instrumentation nonconservative setpoint bias following the Unit 2 manual reactor trip on February 9, 2002, is a violation of 10 CFR Part 50, Criterion XVI. The licensee's event review failed to recognize that an engineered safety feature, including a reactor trip, failed to actuate when required during a loss of feedwater event to Steam Generator 2-4. This failure resulted in the licensee restarting Unit 2 with the reactor trip and engineered safety system instrumentation inoperable, and in the operation of both units with the same instrumentation inoperable, in violation of Technical Specification 3.3.1. This issue is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (50-275; 323/2002-07-01). The licensee documented this deficiency in Action Request A0549031. The failure to promptly recognize inoperable trip and actuation functions and comply with Technical Specification requirements had a credible impact on safety. The resulting delays in an automatic reactor trip and engineered safety features actuations would have delayed the plant's response to a loss of feedwater event and reduced the water mass available for the heat sink function in the affected steam generator(s). Further, this deficiency had the potential to affect the integrity of the reactor coolant system boundary. A Phase 3 Significance Determination Process evaluation concluded that the issue had very low safety significance (Green). The finding represents a condition that existed for 5-days. The significance of the steam generator narrow range water level-low low setpoint offset (bias) is reduced if feedwater flow is lost to two or more steam generators. Based on the short duration from the time a single steam generator would dryout (the limiting initiator is a loss of feedwater to a single generator) and actuation of auxiliary feedwater, the condition does not result in an appreciable increase in the probability of a steam generator tube rupture occurring. The licensee's analysis using the plant specific simulator showed that the engineered safety feature actuation and reactor trip on steam generator water level-low low would have initiated at or before steam generator dryout would occur. The reactor coolant system physical over pressure protective features (safety relief and power operated relief valves) should not be challenged and there were other protective trips in place (over temperature-delta temperature and over pressure delta-temperature) in place that would have protected the reactor coolant system and fuel integrity in the event a manual reactor trip is not initiated on a loss of feedwater flow to a steam generator [Sections 4OA2.a.(2) and 5].
Inspection Report# : 2002007*(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** 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)*

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# Public Radiation Safety

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# 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)*

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# Miscellaneous

**Significance: N/A** May 31, 2002
Identified By: NRC
Item Type: FIN Finding
**Identification and Resolution of Problems**
The licensee was effective at identifying problems and placing them into the corrective action program with one exception in the area of operability determinations. Occasionally an operability determination being reviewed by engineering was not timely. For example, the licensee failed to identify and evaluate how differential pressure affected steam generator instrumentation and its affect on operability prior to starting the plant following a trip with unusual steam generator level indications. The licensee appropriately determined the extent of evaluation of individual problems and prioritized the schedule for implementation of corrective actions to address the safety significant issues. In general, corrective actions, when specified, were effective and were implemented in a timely manner. The licensee performed effective audits and assessments. Based on the interviews conducted during this inspection, workers at the site felt free to input safety issues into the problem identification and resolution program.
Inspection Report# : 2002002*(pdf)*

**Significance: N/A** Apr 05, 2002
Identified By: NRC

Item Type: FIN Finding
**Identification and Resolution of Problems**
The team determined that a critical opportunity was missed to promptly identify and correct a risk significant condition adverse to quality involving a nonconservative safety features set point. The licensee's post trip event review process did not ensure that the Unit 2 plant response to a loss of feedwater flow to Steam Generator 2-4 was appropriate in that the steam generator level lo-lo engineered safety features and automatic reactor trip actuations did not occur when required.
Inspection Report# : 2002007*(pdf)*

Last modified : May 30, 2003

# Diablo Canyon 2
# 2Q/2003 Plant Inspection Findings

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## 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:** 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)*

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# Mitigating Systems

**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)*

**Significance:** G Jul 11, 2002
Identified By: NRC
Item Type: FIN Finding
**Grounding resistor vulnerability**
The plant electrical distribution consisted of a design where the three redundant 4160 V safety buses and a non-safety bus were supplied from a common transformer winding during both normal and emergency operation. The 4160 V buses were interconnected by conductors so that a voltage disturbance on any part of the system would affect the entire system. The system had a high resistance grounding design to limit the magnitude of ground faults and to enable continued operation of a faulted load. The grounding resistor admits sufficient fault current to prevent severe over-voltages that could occur. However, if the grounding resistor developed an open circuit, the entire system would be susceptible to over-voltage. The licensee was periodically checking the continuity, but not the actual resistance of the grounding resistors and, thus, assumptions in the design were not being verified. The licensee issued Action Request A0561002 to evaluate the preventive maintenance program of the high resistance grounding program. This issue did not involve a violation of NRC requirements, but was considered to be a finding because it revealed a vulnerability in the licensee's design and maintenance that could result in a safety problem. However, the finding was determined to be of very low safety significance because there was no evidence that the grounding resistor had ever been degraded and that the probability of a grounding resistor failure in combination with a sparking ground fault was very small.
Inspection Report# : 2002006*(pdf)*

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# 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)*

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## Emergency Preparedness

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## 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)*

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## Public Radiation Safety

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## 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)*

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## Miscellaneous

Last modified : September 04, 2003

# Diablo Canyon 2
# 3Q/2003 Plant Inspection Findings

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## 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:** 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)*

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# Mitigating Systems

**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:** 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:** 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:** 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)*

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## 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)*

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## Emergency Preparedness

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## Occupational Radiation Safety

**Significance:** 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)*

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## Public Radiation Safety

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## 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)*

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## Miscellaneous

Last modified : December 01, 2003

# Diablo Canyon 2
# 4Q/2003 Plant Inspection Findings

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## 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:** 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)*

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# 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:** 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:** 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 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)*

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# 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)*

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# Emergency Preparedness

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# 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)*

**Significance:** 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)*

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## Public Radiation Safety

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## 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)*

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## Miscellaneous

Last modified : March 02, 2004

# Diablo Canyon 2
# 1Q/2004 Plant Inspection Findings

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## 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)*

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## 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:** 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:** 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:** 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:** 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:** 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:** 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)*

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## 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)*

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## Emergency Preparedness

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## 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)*

## Public Radiation Safety

## Physical Protection

## Miscellaneous

Last modified : May 05, 2004

# Diablo Canyon 2
# 2Q/2004 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Translate Design Basis of Diesel Fuel Oil Storage Tank into Implementing Procedures**
A oncited violation of 10 CFR 50, Appendix B, Criterion III, was identified for failure to translate the design basis of the fuel oil storage system into procedures. Calculation M-786 provided the basis that the technical specification capacity of the fuel oil storage tanks contained 7 days of fuel for a loss of offsite power for both units, based on each unit operating only the minimum safety related loads to achieve and maintain safe shutdown. However, this loading strategy was not translated into procedures, nor was any instructions to alert operators to take actions to conserve fuel oil. Without taking actions to minimize loads, if all six diesel generators ran fully loaded, Diablo Canyon would have enough fuel to last two days upon a loss of offsite power.

This issue affects the mitigating systems cornerstone and is more than minor because it could have actual impact on the ability to mitigate a long-term loss of offsite power event up to the 7 day design basis capacity of the fuel oil storage tank. Using the phase 1 significance determination process the inspectors determined that the issue screens to green because it did not involve unavailability of any technical specification system. The licensee has alternate means to obtain additional fuel oil from offsite sources in an expeditious manner. Therefore, this issue has very low safety signficance.
Inspection Report# : 2004002*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Adequate Procedures for Preventive Maintenance and Operation of Limitorque Motor-operated Valves in a Moist Environment**
A non-cited violation was identified by the inspectors for the failure to assure activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, as required by 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Pacific Gas and Electric Company failed to provide adequate procedures for preventive maintenance and operation of Limitorque motor-operated valves in a moist environment. The inadequate procedures resulted in the degraded operation of three Limitorque motor-operated valves in the ASW system during quarterly valve surveillance activities.

This finding impacted the mitigating systems cornerstone and is greater than minor because the finding would become a more significant safety concern if the problem was left uncorrected. Specifically, the problems of undiscovered rust formation on the valve declutch lever and the out-of-adjustment tripper fingers would continue to repeatedly affect the closing operation of the three ASW motor-operated valves, and thus affect the overall reliability of the ASW systems. Using the SDP Phase 1 Worksheet in Inspection Manual Chapter 0609, the inspectors determined that this finding is of very low safety significance. Although operation of the three ASW valves were degraded, the three motor-operated valves were available to perform their intended safety functions. The finding did not result in a loss of safety function or screen as potentially risk significant from the consideration of external event impacts.
Inspection Report# : 2004002*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Control Placement of Temporary Equipment With Regards to Potential Seismic Impact on Safety-Related Systems**
A non-cited violation was identified by the inspectors for the failure to adequately control the potential for seismically-induced impact of stored, temporary equipment on safety systems, as required by Technical Specification 5.4.1.a. Specifically, on March 18 and March 31, the inspectors and Pacific Gas and Electric Company staff identified four instances were transient equipment was stored in close proximity to safety-related systems. The transient equipment was determined not to impact the functionality of the safety-related systems in the event of an earthquake.

The finding impacted the mitigating systems cornerstone and was more than minor when compared to Example 4.a of Inspection Manual Chapter 0612, Appendix E. Similar to the example, the inspectors and Pacific Gas and Electric Company staff found several examples on the auxiliary building 140 ft. elevation where temporary equipment was stored contrary to procedures to protect safety-related systems from seismic impact. Using the Significance Determination Process Phase I worksheet in Inspection Manual Chapter 0609, Appendix A, the finding is of very low safety significance since it did not

screen as potentially risk significant due to a seismic event. Specifically, the inspectors determined that the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic event and it does not involve the total loss of any safety function with respect to a seismic event.
Inspection Report# : 2004002*(pdf)*

**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:** 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)*

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## Barrier Integrity

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## Emergency Preparedness

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## 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)*

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## Public Radiation Safety

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

Last modified : September 08, 2004

# Diablo Canyon 2
# 3Q/2004 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Identify Multiple Grounds in Containment Spray Pump 2-2 Control Circuitry**
A self-revealing, noncited 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to promptly identify multiple grounds in the breaker control circuitry for Containment Spray Pump 2-2. Specifically, Pacific Gas and Electric Company missed several opportunities, in part because of a failure to utilize the troubleshooting procedure, to pursue the cause of the ground and to address anomalous indications, the proximity of a known ground to other conductors, and operating experience. The grounds degraded control wires affecting the pump's manual/automatic breaker closure circuits, indication circuits, and overcurrent circuits for up to 70 days following the initial ground indication. A problem identification and resolution crosscutting aspect was identified for the troubleshooting and corrective actions associated with the grounds. The grounded cable was subsequently replaced. Similar to Example 4.f in Appendix E of Inspection Manual Chapter 0612, the finding is greater than minor because the multiple grounds affected the operability of containment spray pump 2-2. Using the Inspection Manual Chapter 0609 Phase I Screening Worksheet, the finding was of very low safety significance since there was not an actual reduction of the atmospheric pressure control function for containment.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Simulator with respect to Backup Seismic Alarm**
A noncited violation of 10 CFR 55.46 was identified by the inspectors for the failure to maintain the plant referenced simulator to respond to normal, transient and accident conditions. Pacific Gas and Electric Company removed from service, and abandoned the Backup Seismic System (Terra Tech Instrument) in place in June 2000. However, as of August 31, 2004, the plant referenced simulator still provided an annunciator fed from the backup seismic system when an earthquake of sufficient magnitude was felt. This provided operators with negative training in that operators were trained that the backup seismic system would provide annunciation and indication.

This finding affects the mitigating systems cornerstone and is greater than minor because it results in negative training of the operators to expect an annunciator from a backup seismic system in the event of an earthquake, if the earthquake force monitor was unavailable. Using the flow chart of Appendix I, of Inspection Manual Chapter 0609 of the Significance Determination Process, this issue affects operator actions in that operators may attempt to obtain ground motion from backup seismic monitors that did not exist. Per Inspection Manual Chapter 0609, Appendix I, Item 12, the inspectors determined that the finding was Green because the differences between the plant control room and the plant reference simulator negatively impacted operator actions and resulted in negative training.

Inspection Report# : 2004004*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Translate Design Basis of Diesel Fuel Oil Storage Tank into Implementing Procedures**
A noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified for the failure to translate the diesel emergency generator fuel oil usage design basis assumptions into procedures. Specifically, Calculation M-786 provided the basis for the Technical Specification minimum required volume of fuel oil in the fuel oil storage tanks to meet a 7 day fuel oil supply following a loss of offsite power for both units. The minimum volume was based on each unit operating only the minimum safety-related loads to achieve and maintain safe shutdown. However, the diesel engine generator minimum safety-related loads were not translated into procedures, nor were any instructions provided to alert operators to take actions to conserve fuel oil. With all six diesel engine generators running fully loaded there is insufficient fuel oil in the fuel oil storage tanks for 7 days of operation.

This issue affects the mitigating systems cornerstone objective to ensure the availability of onsite emergency AC power during the entire period described in the design basis. This issue is more than minor because it could have an actual impact on the ability of the diesel engine generators to mitigate a long-term loss of offsite power event. Using the Phase 1 significance determination process the inspectors determined that the issue was of very low safety-significance because the finding does not represent an actual loss of a safety system or a single train and did not meet the criteria for being risk significant because of an external event.

Inspection Report# : 2004002*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Adequate Procedures for Preventive Maintenance and Operation of Limitorque Motor-operated Valves in a Moist Environment**
A noncited violation with two examples was identified by the inspectors for the failure to assure activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, as required by 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Pacific Gas and Electric failed to provide adequate procedures for preventive maintenance and operation of Limitorque motor-operated valves. The inadequate procedures resulted in the degraded operation of three Limitorque motor-operated valves in the auxiliary saltwater system during quarterly valve surveillance activities.

The performance deficiency associated with the finding is the failure to provide adequate instructions for preventive maintenance and operation of Limitorque motor-operated valves. The preventive maintenance aspect was evident with the Limitorque valves located in a moist environment. This finding impacted the mitigating systems cornerstone for the reliability of the auxiliary saltwater system that affects both shutdown and operating equipment. The finding is greater than minor because the finding would become a more significant safety concern if the problem was left uncorrected. Specifically, the problems of undiscovered rust formation on the valve declutch lever and the out-of-adjustment tripper fingers would continue to affect manual operation of the Limitorque valves and the ability to re-engage the motor operator. Using the SDP Phase 1 Worksheet in Inspection Manual Chapter 0609, the inspectors determined that this finding is of very low safety significance. Although operation of the three auxiliary salt water valves were degraded, the three motor-operated valves were available to perform their intended safety functions. The finding did not result in a loss of safety function or screen as potentially risk significant from the consideration of external event impacts.
Inspection Report# : 2004002*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Control Placement of Temporary Equipment With Regards to Potential Seismic Impact on Safety-Related Systems**
A noncited violation of Technical Specification 5.4.1.a. was identified by the inspectors for the failure to adequately control the storage of temporary equipment that has a potential for seismically-induced system interaction with safety systems. Specifically, on March 18 and then on March 31, the inspectors identified an instance were transient equipment was stored in close proximity to safety systems and considered to be potential seismically-induced system interactions. On March 18, Pacific Gas and Electric identified two other instances where temporary equipment could cause a seismically-induce system interaction with safety systems. In each case the equipment was determined not to impact the functionality of the safety systems in the event of an earthquake.

The finding impacted the mitigating systems cornerstone for protection against external hazards. The issue was determined to be more than minor when compared to Example 4.a of Inspection Manual Chapter 0612, Appendix E. Similar to the example, the inspectors and Pacific Gas and Electric found four examples on the auxiliary building 140 ft. elevation where temporary equipment was stored contrary to procedures to protect safety-related systems from seismic impact. Using the Significance Determination Process Phase I worksheet in Inspection Manual Chapter 0609, Appendix A, the finding is of very low safety significance since it did not screen as potentially risk significant due to a seismic event. Specifically, the inspectors determined that the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic event and it does not involve the total loss of any safety function with respect to a seismic event.

Inspection Report# : 2004002*(pdf)*

**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)*

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## Barrier Integrity

**Significance:** G Sep 30, 2004
Identified By: NRC

Item Type: FIN Finding
**Failure to properly implement an operating instruction for an inoperable containment isolation valve.**
The inspectors identified a finding for the failure to properly isolate containment isolation Valve VAC-2-FCV-681(an air-operated containment isolation valve) after it failed to fully stroke open and was declared inoperable. Operators hung administrative tags on the control room switch for the valve but failed to remove the motive force from the valve by isolating air to the actuator. The associated operating instruction required that the valve be closed and deactivated. A human performance crosscutting aspect was identified for the failure to properly implement the operating instruction for an inoperable containment isolation valve.

This issue affects the barrier integrity cornerstone objective to ensure that systems penetrating the containment and are connected directly to the containment atmosphere have adequate isolation to protect the containment barrier. This issue is greater than minor because failure to properly close and deactivate containment isolation valves could have an actual impact on the ability to isolate a fault outside of containment. Using the Phase 1 significance determination process, the inspectors determined that the issue was of very low safety significance because the finding did not represent an actual open significant pathway to the environment and the penetration was isolated by an active valve having secured flow.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Change to the Emergency Plan with respect to Backup Seismic System**
A violation of 10CFR 50.54(q) was identified by the inspectors for failure to update and submit changes to the emergency plan within 30 days. Specifically, Section 7.5.1 of the Diablo Canyon Emergency Plan stated that a supplemental seismic system, supplied by Terra Tech Corporation, provided backup local indication and control room annunciation on strong ground motion. The Terra Tech system was removed from service, along with its annunciation in the control room, and abandoned in place in July of 2000, but as of September 30, 2004, Pacific Gas and Electric had not revised its emergency plan to reflect this change.

The finding was evaluated using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," Section IV, because licensee reductions in the effectiveness of its emergency plan impact the regulatory process. The finding had greater than minor significance because deletion of conditions indicative of a site area emergency has the potential to impact safety. The finding was determined to be a noncited Severity Level IV violation because the finding involved a violation of a regulatory requirement and did not constitute a failure to meet an emergency planning standard as defined by 10 CFR 50.47(b). This finding has been entered into the licensee's corrective action program as Action Request A0618799.
Inspection Report# : 2004004*(pdf)*

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## Emergency Preparedness

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## 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)*

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## Public Radiation Safety

## Physical Protection

Physical Protection information not publicly available.

## Miscellaneous

Last modified : December 29, 2004

# Diablo Canyon 2
# 4Q/2004 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Properly Implement Procedure for Spent Fuel Pool Skimmer Filter Replacement**
A self-revealing NCV was identified for the failure to appropriately implement the procedure for spent fuel pool skimmer filter replacement, as required by Technical Specification 5.4.1.a. On December 23, 2004, operators cleared the spent fuel pool skimmer system using Section 6.3.1 of Procedure OP B-7:III, "Spent Fuel Pool System - Shutdown and Clearing and Filter Replacement," Revision 15, instead of the appropriate section, which was Section 6.3.2. A human performance cross cutting aspect was identified for the failure on two occasions to address configuration control concerns with the system.

This finding impacted the Initiating Events Cornerstone and was considered more than minor using Example 5.a of IMC 0612. Specifically, Valve SFS-2-3 was mis-positioned due to the use of the wrong section of Procedure OP B-7:III and then returned to service. Additionally, operators had two opportunities to identify the mis-positioning of Valve SFS-2-3 but failed to identify the condition. The mis-positioned valve resulted in a loss of approximately 36,000 gallons of water from the spent fuel pool. Using the SDP Phase 1 screening worksheet of IMC 0609, Appendix A, the finding was evaluated as a transient initiator, and it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Therefore, the finding was screened as having very low safety significance
Inspection Report# : 2004005*(pdf)*

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## Mitigating Systems

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Wire and Connect Test Equipment Resulted in Vital Bus De-Energization**
A self-revealing, noncited violation was identified for the failure to set up phase sequence test equipment according to procedure, as required by 10 CFR Part 50, Appendix B, Criterion V. This failure resulted in the momentary de-energization of Vital 4kV Bus G and the auto-start of Diesel Engine Generator 2-1. Subsequent investigation by Pacific Gas & Electric Company (PG&E) revealed that the primary side of the test transformer was wired in a wye configuration instead of a delta configuration. This wiring configuration introduced a direct short to ground, which caused the second level undervoltage relay to sense a degraded bus voltage for Vital 4kV Bus G. Subsequently, the relay removed the auxiliary power supply from Bus G and caused DEG 2-1 to start and load onto the bus. This finding involved a human performance cross-cutting aspect for the failure to wire the phase sequence test equipment properly for Vital 4kV Bus G and DEG 2-1.

The finding impacted the Mitigating Systems Cornerstone for ensuring the availability and capability of systems that respond to initiating events to prevent undesirable consequences that was associated with a pre-event human error performance. Considering Example 4.b of IMC 0612, Appendix E, the finding is greater than minor since the incorrect wiring and connection of the test equipment resulted in a vital bus de-energization and the actuation of DEG 2-1. Using Checklist 4 of Inspection Manual Chapter (IMC) 0609, Appendix G, Attachment 1, the finding did not result in the Technical Specifications for AC and DC power sources not being met and the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade PG&E's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade PG&E's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Inadequate ASCO Valve Qualification Causes Plant Trip**
A self revealing violation of 10 CFR 50.49(f) was identified for the failure to maintain approximately 70 safety related solenoid operated valves in an environmentally qualified condition. On February 9, 2002, an age related ASCO solenoid operated valve failure caused a loss of steam generator feedwater event and a Unit 2 manual plant trip. Further, the licensee did not promptly evaluate the extent of condition of the ASCO failure (coil insulation failure), which delayed the identification of elastomer qualification issues for approximately 1 year. In a related finding,

the team identified that the licensee had missed earlier opportunities to identify ASCO elastomer qualification issues, in that they failed to thoroughly evaluate several pertinent NRC information notices and previous valve failures. The failure to: 1) properly establish equipment qualification limits; 2) thoroughly evaluate plant events and failures; and 3) properly evaluate industry operating experience constituted performance concerns. PG&E entered this issue into their corrective action program as Action Request 0613008. These issues have cross-cutting aspects in the area of problem identification and resolution because the original problem investigation did not identify the full scope of the cause and extent of condition, delaying some important corrective actions for approximately 1 year.

This finding was greater than minor because, if left uncorrected, these deficiencies would become a more significant safety concern by increasing the failure rate as the components age. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated delta-CDF for the finding is 2.2E-8/yr. This violation was of very low risk significance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Promptly Identify Multiple Grounds in Containment Spray Pump 2-2 Control Circuitry**
A self-revealing, noncited 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to promptly identify multiple grounds in the breaker control circuitry for Containment Spray Pump 2-2. Specifically, Pacific Gas and Electric Company missed several opportunities, in part because of a failure to utilize the troubleshooting procedure, to pursue the cause of the ground and to address anomalous indications, the proximity of a known ground to other conductors, and operating experience. The grounds degraded control wires affecting the pump's manual/automatic breaker closure circuits, indication circuits, and overcurrent circuits for up to 70 days following the initial ground indication. A problem identification and resolution crosscutting aspect was identified for the troubleshooting and corrective actions associated with the grounds. The grounded cable was subsequently replaced. Similar to Example 4.f in Appendix E of Inspection Manual Chapter 0612, the finding is greater than minor because the multiple grounds affected the operability of containment spray pump 2-2. Using the Inspection Manual Chapter 0609 Phase I Screening Worksheet, the finding was of very low safety significance since there was not an actual reduction of the atmospheric pressure control function for containment.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Simulator with respect to Backup Seismic Alarm**
A noncited violation of 10 CFR 55.46 was identified by the inspectors for the failure to maintain the plant referenced simulator to respond to normal, transient and accident conditions. Pacific Gas and Electric Company removed from service, and abandoned the Backup Seismic System (Terra Tech Instrument) in place in June 2000. However, as of August 31, 2004, the plant referenced simulator still provided an annunciator fed from the backup seismic system when an earthquake of sufficient magnitude was felt. This provided operators with negative training in that operators were trained that the backup seismic system would provide annunciation and indication.

This finding affects the mitigating systems cornerstone and is greater than minor because it results in negative training of the operators to expect an annunciator from a backup seismic system in the event of an earthquake, if the earthquake force monitor was unavailable. Using the flow chart of Appendix I, of Inspection Manual Chapter 0609 of the Significance Determination Process, this issue affects operator actions in that operators may attempt to obtain ground motion from backup seismic monitors that did not exist. Per Inspection Manual Chapter 0609, Appendix I, Item 12, the inspectors determined that the finding was Green because the differences between the plant control room and the plant reference simulator negatively impacted operator actions and resulted in negative training.

Inspection Report# : 2004004*(pdf)*

**Significance:** G Jun 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design and Test Controls of the Diesel Emergency Generator Fuel Oil Level Control Valves**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, for the failure to maintain design control of the diesel emergency generator system fuel oil transfer system requirements. Specifically, the fuel supply to each diesel required that an adequate air supply to operate the air-operated day tank level control valve be maintained in the starting air receivers. The team identified that when the licensee recognized that this design basis was not documented, a calculation was performed to support creating the design basis which did not account for operational leakage from the system, nor did it verify that existing leakage would not prevent fulfilling the safety function. This failure potentially affected the ability of each diesel emergency generator to provide sufficient fuel oil to support 7 days of continuous diesel generator operations following a loss of offsite power. This issue was entered into the corrective action program under Action Request A0613008. This finding involved cross-cutting aspects in the area of problem identification and resolution because the original corrective actions did not correct the problem and properly establish the design basis.

This finding was greater than minor because it was similar to Example 3.i of Manual Chapter 0612, Appendix E. This finding affected the mitigating systems cornerstone. This finding was evaluated using NRC Manual Chapter 0609, Significance Determination Process, Phase 1

worksheet under the mitigating systems cornerstone. The finding was determined to be of very low safety significance because the deficiency was confirmed not to result in a loss of function of the diesel engine generator as a power source per Generic Letter 91-18, Revision 1. The licensee was able to demonstrate that compensatory measures were in place so that this function could be performed manually in a reliable manner because operators would receive a control room alarm which triggered implementation of proceduralized step to manually perform the function. The team confirmed that operators were trained to perform this action.
Inspection Report# : 2004006*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Translate Design Basis of Diesel Fuel Oil Storage Tank into Implementing Procedures**
A noncited violation of 10 CFR Part 50, Appendix B, Criterion III, was identified for the failure to translate the diesel emergency generator fuel oil usage design basis assumptions into procedures. Specifically, Calculation M-786 provided the basis for the Technical Specification minimum required volume of fuel oil in the fuel oil storage tanks to meet a 7 day fuel oil supply following a loss of offsite power for both units. The minimum volume was based on each unit operating only the minimum safety-related loads to achieve and maintain safe shutdown. However, the diesel engine generator minimum safety-related loads were not translated into procedures, nor were any instructions provided to alert operators to take actions to conserve fuel oil. With all six diesel engine generators running fully loaded there is insufficient fuel oil in the fuel oil storage tanks for 7 days of operation.

This issue affects the mitigating systems cornerstone objective to ensure the availability of onsite emergency AC power during the entire period described in the design basis. This issue is more than minor because it could have an actual impact on the ability of the diesel engine generators to mitigate a long-term loss of offsite power event. Using the Phase 1 significance determination process the inspectors determined that the issue was of very low safety-significance because the finding does not represent an actual loss of a safety system or a single train and did not meet the criteria for being risk significant because of an external event.

Inspection Report# : 2004002*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Adequate Procedures for Preventive Maintenance and Operation of Limitorque Motor-operated Valves in a Moist Environment**
A noncited violation with two examples was identified by the inspectors for the failure to assure activities affecting quality shall be accomplished in accordance with documented instructions, procedures, or drawings, as required by 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Pacific Gas and Electric failed to provide adequate procedures for preventive maintenance and operation of Limitorque motor-operated valves. The inadequate procedures resulted in the degraded operation of three Limitorque motor-operated valves in the auxiliary saltwater system during quarterly valve surveillance activities.

The performance deficiency associated with the finding is the failure to provide adequate instructions for preventive maintenance and operation of Limitorque motor-operated valves. The preventive maintenance aspect was evident with the Limitorque valves located in a moist environment. This finding impacted the mitigating systems cornerstone for the reliability of the auxiliary saltwater system that affects both shutdown and operating equipment. The finding is greater than minor because the finding would become a more significant safety concern if the problem was left uncorrected. Specifically, the problems of undiscovered rust formation on the valve declutch lever and the out-of-adjustment tripper fingers would continue to affect manual operation of the Limitorque valves and the ability to re-engage the motor operator. Using the SDP Phase 1 Worksheet in Inspection Manual Chapter 0609, the inspectors determined that this finding is of very low safety significance. Although operation of the three auxiliary salt water valves were degraded, the three motor-operated valves were available to perform their intended safety functions. The finding did not result in a loss of safety function or screen as potentially risk significant from the consideration of external event impacts.
Inspection Report# : 2004002*(pdf)*

**Significance:** G Mar 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Control Placement of Temporary Equipment With Regards to Potential Seismic Impact on Safety-Related Systems**
A noncited violation of Technical Specification 5.4.1.a. was identified by the inspectors for the failure to adequately control the storage of temporary equipment that has a potential for seismically-induced system interaction with safety systems. Specifically, on March 18 and then on March 31, the inspectors identified an instance were transient equipment was stored in close proximity to safety systems and considered to be potential seismically-induced system interactions. On March 18, Pacific Gas and Electric identified two other instances where temporary equipment could cause a seismically-induce system interaction with safety systems. In each case the equipment was determined not to impact the functionality of the safety systems in the event of an earthquake.

The finding impacted the mitigating systems cornerstone for protection against external hazards. The issue was determined to be more than minor when compared to Example 4.a of Inspection Manual Chapter 0612, Appendix E. Similar to the example, the inspectors and Pacific Gas and Electric found four examples on the auxiliary building 140 ft. elevation where temporary equipment was stored contrary to procedures to protect safety-related systems from seismic impact. Using the Significance Determination Process Phase I worksheet in Inspection Manual Chapter 0609, Appendix A, the finding is of very low safety significance since it did not screen as potentially risk significant due to a seismic event. Specifically, the inspectors determined that the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic event and it does not involve the total loss of any safety function with respect to a seismic event.

Inspection Report# : 2004002*(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Mislabel of Neutron Flux Detector Resulted in Neutronic Decoupling of Detector From the Core**
The inspectors identified a noncited violation for the failure to develop a core offload sequence that maintained the source range neutron flux monitors operable, as required by 10 CFR Part 50, Appendix B, Criterion V. Inaccurate labeling of two neutron detectors in the core offload planning tool resulted in the development of a core offload sequence that when implemented resulted in one of the detectors becoming neutronically uncoupled from the core during core alterations. A human performance crosscutting aspect was identified for the labeling error in the core offload planning. A second human performance crosscutting aspect was identified for the failure to ascertain the cause of the downward trend when first identified by the inspectors.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute for procedure quality which could impact cladding. The finding is more than minor when compared to Example 4.e of Inspection Manual Chapter 0612, Appendix E. Similar to the example, Procedure OP B-8DS1, Step 5.2.1, described a responding nuclear instrument as having at least one fuel assembly face-adjacent or diagonally adjacent to the detector. Due to a labeling error in the core offload planning tool, the core offload sequence was developed in a manner that caused a neutron detector (Detector N-52) not to have an adjacent fuel assembly. Using Checklist 4 of Inspection Manual Chapter 0609, Appendix G, Attachment 1, the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade Pacific Gas & Electric Company's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade Pacific Gas & Electric Company's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Containment Fan Cooler Unit Reverse Rotation**
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly correct reverse rotation of containment fan cooler units (CFCUs) for both Units 1 and 2. PG&E observed reverse rotation of CFCUs for approximately 13 years, as a result of the CFCU backdraft dampers sticking partially open. The purpose of the backdraft dampers is to prevent reverse rotation of the CFCUs, which could cause the fan motor to trip on overcurrent when the CFCUs are started following a loss of coolant accident. Prior to Refueling Outage 2R12, 2 CFCUs in Unit 1 and 3 CFCUs in Unit 2 exhibited reverse rotation. One of the CFCUs in Unit 2 was considered inoperable due to reverse rotation and another was only considered operable if it was running.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute. The finding is more than minor when considering Example 3.g of IMC 0612, Appendix E. Similar to the example, PG&E observed reverse rotation of CFCUs for 13 years, and the reverse rotation impacted the operability of the CFCUs. Using the SDP Phase 1 Screening Worksheet from IMC 0609, the finding was determined to be of very low safety significance since it was determined that there was not an actual loss of defense-in-depth in containment pressure control or hydrogen control .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: FIN Finding
**Failure to properly implement an operating instruction for an inoperable containment isolation valve.**
The inspectors identified a finding for the failure to properly isolate containment isolation Valve VAC-2-FCV-681(an air-operated containment isolation valve) after it failed to fully stroke open and was declared inoperable. Operators hung administrative tags on the control room switch for the valve but failed to remove the motive force from the valve by isolating air to the actuator. The associated operating instruction required that the valve be closed and deactivated. A human performance crosscutting aspect was identified for the failure to properly implement the

operating instruction for an inoperable containment isolation valve.

This issue affects the barrier integrity cornerstone objective to ensure that systems penetrating the containment and are connected directly to the containment atmosphere have adequate isolation to protect the containment barrier. This issue is greater than minor because failure to properly close and deactivate containment isolation valves could have an actual impact on the ability to isolate a fault outside of containment. Using the Phase 1 significance determination process, the inspectors determined that the issue was of very low safety significance because the finding did not represent an actual open significant pathway to the environment and the penetration was isolated by an active valve having secured flow.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Change to the Emergency Plan with respect to Backup Seismic System**
A violation of 10CFR 50.54(q) was identified by the inspectors for failure to update and submit changes to the emergency plan within 30 days. Specifically, Section 7.5.1 of the Diablo Canyon Emergency Plan stated that a supplemental seismic system, supplied by Terra Tech Corporation, provided backup local indication and control room annunciation on strong ground motion. The Terra Tech system was removed from service, along with its annunciation in the control room, and abandoned in place in July of 2000, but as of September 30, 2004, Pacific Gas and Electric had not revised its emergency plan to reflect this change.

The finding was evaluated using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," Section IV, because licensee reductions in the effectiveness of its emergency plan impact the regulatory process. The finding had greater than minor significance because deletion of conditions indicative of a site area emergency has the potential to impact safety. The finding was determined to be a noncited Severity Level IV violation because the finding involved a violation of a regulatory requirement and did not constitute a failure to meet an emergency planning standard as defined by 10 CFR 50.47(b). This finding has been entered into the licensee's corrective action program as Action Request A0618799.
Inspection Report# : 2004004*(pdf)*

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# Emergency Preparedness

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Establish Compensatory Measures to Ensure the Implementation of the Diablo Canyon Emergency Plan**
The inspectors identified a violation of 10 CFR 50.54(q) and 50.47.b(4) for the failure to maintain the seismic force monitors during the periods, June 16-19,1999, December 1-4, 2000, April 25-27, 2002, May 25-29, 2002, November 6-8, 2003, December 30-31, 2003, and August 9-10, 2004, such that the emergency plan designed to meet planning standard (4) in 10 CFR 50.47(b) could be promptly implemented. Specifically, PG&E failed to provide a means for the emergency director to promptly classify seismic events at the notification of unusual event, alert or site area emergency levels, while the seismic force monitor utilized by the operators (emergency director) was out of service or being replaced. This finding had a human performance cross-cutting aspect associated with identifying compensatory measures to address the removal of the earthquake force monitors.

This performance deficiency impacted the emergency preparedness cornerstone because PG&E did not meet an emergency planning requirement and the cause was reasonably within PG&E 's control and should have been prevented. It is greater than minor because it has a potential to impact safety and because it was not a record keeping or administrative issue or an insignificant procedural error. This deficiency could have affected the EP Cornerstone objective of ensuring the capability to implement measures to protect the health and safety of the public during an emergency, and is associated with attributes of facilities and equipment, and offsite emergency preparedness. The finding is evaluated using the Emergency Preparedness "Failure to Comply" flowchart of the SDP and is a violation of 10 CFR 50.54(q) and planning standard 50.47(b)(4), which states, in part, that a standard emergency action level and classification system... is in use Utilizing the Failure to Comply Flow Chart in Manual Chapter 0609, the performance deficiency does not result in a failure of the risk significant planning standard (RSPS) or a degraded RSPS in that the unavailability of the seismic monitors would not prevent the declaration of a Site Area Emergency, Alert or Notification of Unusual Event .
Inspection Report# : 2004005*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2004

Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Lock a High Radiation Area with Dose Rates Greater than 1 Rem per Hour**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent unauthorized entry of a portion of the whole body into a high radiation area with dose rates greater than 1 rem per hour. Specifically, on November 14, 2004, PG&E failed to use an effective locking mechanism on the lower access flaps of the primary steam generator shield doors. The ineffective locking mechanism was discovered two days later when workers went to remove suction hoses. This could have allowed an individual to expose the arm above the elbow to dose rates greater than 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Access a High Radiation Area with Dose Rates Greater than 1 Rem per Hour with the Correct Radiation Work Permit**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent two individuals from entering a high radiation area with dose rates greater than 1 rem per hour on the incorrect radiation work permit. Two individuals entered an area with dose rates of 6 rem per hour in Reactor Coolant Pump Cubicle 2-4 using a radiation work permit which only allowed entry into areas with dose rates up to 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Aug 12, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Perform Radiological Survey of a High Radiation Area**
Green. A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified for the failure to perform required radiation surveys in Unit 2 to ensure compliance with 10 CFR 20.1902(b). Specifically, on January 28, 2003, during the performance of venting the volume control tank radiation protection personnel failed to perform adequate surveys of the Unit 2 Gas Decay Tank Room to post an expected high radiation area that would occur during this evolution. This finding involved cross-cutting aspects in the area of problem identification and resolution because the team noted that corrective actions for a similar event under the same circumstances had been ineffective in preventing recurrence. This issue was entered into the corrective action program under Action Request A0572997.

The finding is greater than minor because it was associated with one of the occupational radiation safety cornerstone attributes (exposure), and the finding affected the associated cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material. The inspector processed the issues through the Occupational Radiation Protection Significance Determination Process. This issues were determined to be a Green finding because it was not an ALARA planning and control issue, there was no personnel overexposure or substantial potential for personnel overexposure, and the licensee's ability to assess dose was not compromised.
Inspection Report# : 2004006*(pdf)*

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## Public Radiation Safety

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## Physical Protection

Physical Protection information not publicly available.

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# Miscellaneous

**Significance: N/A** Jun 25, 2004
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**

The team concluded that the licensee was effective in identifying, evaluating, and correcting problems, although the team identified some examples were identified where conditions adverse to quality were not properly entered into the Action Request system, allowing problem recurrence. The team found that the evaluation and prioritization of problems were mostly conducted properly, although some significant issues were identified as routine because the licensee's process assigned significance based on the actual consequences of problems, rather than considering the potential consequences under design basis conditions. Corrective actions were generally implemented in a timely manner. However, the team found weaknesses with the alignment of corrective actions with the cause, and with the quality of operability evaluations for issues assigned routine significance, because the licensee did not assign a probable cause statement to routine issues. Licensee audits and assessments were found to be responsive to plant performance issues and effective in identifying areas for improvement. During interviews, station personnel communicated a willingness to enter issues into the corrective action program. The team reviewed the licensee's improvement plans for significant cross-cutting issues in human performance and problem identification and resolution. Although it was too early to determine if these will be effective, the team noted that the Human Performance Improvement Plan did not address problems observed in coordinating and supervising operations during outages.

Inspection Report# : 2004006*(pdf)*

Last modified : March 09, 2005

# Diablo Canyon 2
# 1Q/2005 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Properly Implement Procedure for Spent Fuel Pool Skimmer Filter Replacement**
A self-revealing NCV was identified for the failure to appropriately implement the procedure for spent fuel pool skimmer filter replacement, as required by Technical Specification 5.4.1.a. On December 23, 2004, operators cleared the spent fuel pool skimmer system using Section 6.3.1 of Procedure OP B-7:III, "Spent Fuel Pool System - Shutdown and Clearing and Filter Replacement," Revision 15, instead of the appropriate section, which was Section 6.3.2. A human performance cross cutting aspect was identified for the failure on two occasions to address configuration control concerns with the system.

This finding impacted the Initiating Events Cornerstone and was considered more than minor using Example 5.a of IMC 0612. Specifically, Valve SFS-2-3 was mis-positioned due to the use of the wrong section of Procedure OP B-7:III and then returned to service. Additionally, operators had two opportunities to identify the mis-positioning of Valve SFS-2-3 but failed to identify the condition. The mis-positioned valve resulted in a loss of approximately 36,000 gallons of water from the spent fuel pool. Using the SDP Phase 1 screening worksheet of IMC 0609, Appendix A, the finding was evaluated as a transient initiator, and it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Therefore, the finding was screened as having very low safety significance
Inspection Report# : 2004005*(pdf)*

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## Mitigating Systems

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to correct fire program violation concerning qualifications of Operations Responders in support of the fire brigade**
The inspectors identified a noncited violation of Technical Specification 5.4.1.d for failure to implement procedures for Fire Protection Implementation, because of failure to provide adequate training for operations fire responders. Procedure OM8, "Fire Protection Program," Revision 2B, Section 7.8 states, in part, that quality problems associated with the Fire Protection Program shall be documented and resolved in accordance with Procedure OM7 "Corrective Action," Revision 2B. Section 9.5.1 of the Final Safety Analysis Report states that measures are established to ensure conditions adverse to fire protection are identified, reported and corrected, and that administrative procedures are established to implement this requirement. Contrary to the above, Pacific Gas & Electric Company did not adequately implement and maintain a procedure for fire protection. Specifically, Pacific Gas & Electric Company failed to adequately resolve a condition adverse to fire protection in accordance with Procedure OM7. As of March 1, 2005, operations responders were not required to participate in fire drills for initial qualification or maintenance of qualification, as was noted as a qualification deficiency in Non-cited Violation 50-275;323/2003-08-01, and Action Request (AR) A0600934. This finding has problem identification and resolution cross cutting aspects for failure to correct operations responder training deficiencies.

The performance deficiency associated with this finding is a failure to adequately implement the fire protection program with respect to the qualifications of the fire brigade operations responder. The finding impacted the mitigating systems cornerstone and was more than minor since there was an adverse impact to a fire protection defense-in-depth element. Using the Significance Determination Process (SDP) Phase I Screening Worksheet and the SDP Phase II Notebook in Appendix F of Inspection Manual Chapter (IMC) 0609, the inspectors determined that the finding was of very low safety significance. Specifically, the significance of the finding was evaluated by considering fire scenarios in the vital 4 kV Bus F switchgear room and auxiliary saltwater Pump 1-1 vault. These two areas have the highest dependence on fire brigade response since they have the highest fire ignition frequency for areas that do not have automatic fire suppression. The inspectors evaluated the risk-significance using half the nominal credit for manual fire suppression as a result of the finding. Using Tables 5.4, 5.5, and 5.6 of IMC 0609, both fire scenarios screened as very low safety significance. Since the two fire scenarios were considered worst-case for the finding, the inspectors determined that the finding was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Mar 31, 2005

Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to promptly correct diesel engine generator lube oil instrument line crack**
The inspectors identified a noncited violation for the failure to promptly correct a cracked lube oil instrument sensing line, as required by 10 CFR Part 50, Appendix B, Criterion XVI. On August 29, 2004, operators observed a lube oil leak from the weld connecting the outlet of Valve DEG-2-1084 to instrument tubing. Approximately one month later, the leak had increased and it was discovered that the circumferential crack was 180 degrees through-wall on the weld. As a result, there was an increased potential for diesel engine generator (DEG) 2-3 to trip on low lube oil level. The finding had problem identification and resolution crosscutting aspects associated with operations and engineering personnel not recognizing the significance of the degraded condition and not implementing timely corrective actions.

This finding impacted the Mitigating Systems Cornerstone for reliability of systems that respond to initiating events to prevent undesirable consequences, and it affects the equipment performance attribute. The finding was more than minor using Example 4.f of Inspection Manual Chapter 0612, Appendix E. Similar to Example 4.f, the inspectors determined that there was impact to DEG 2-3 operability. Using the SDP Phase 1 screening worksheets in Appendix A of Inspection Manual Chapter 0609, the finding was determined to have potentially greater than very low safety significance because the failure could have resulted in an actual loss of diesel engine Generator 2-3 during a loss of offsite power event. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated conditional core damage frequency was 1.2E-7/yr. This violation was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding
**Diesel fuel oil transfer modification did not adequately assess reliability impact**
A finding was identified for modifying the diesel fuel oil transfer system without properly assessing the resulting net affect on reliability from introducing a new failure potential associated with new active components. As a result, the licensee rejected a small design change, which would have eliminated the failure mode when it was recognized that failure of the new pressure control valves could fail the train. Because the failure potential was not fully assessed, the licensee decided not to implement a change that would have eliminated the impact of the failure, nor were the pressure control valves subject to any preventive maintenance to ensure their reliability. This issue was entered into the licensee's corrective action program under Action Request A0630383.

The failure to properly assess the net effect on system reliability and risk due to the positive and negative effects of this modification, or to mitigate or eliminate a new failure mode created by the modification was a performance deficiency. This issue is more than minor because it affected the design control attribute of the Mitigating Systems cornerstone objective to assure the reliability and capability of equipment needed for accident mitigation. This finding was determined to be of very low safety significance (Green) during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18 based on test results.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding
**Incomplete action for setting auxiliary feedwater pump minimum flow values**
The team identified a Green finding for inadequate response to industry operating experience regarding establishing minimum flow for the auxiliary feedwater pumps. The team concluded that the licensee recognized that the conditions reported in NRC Bulletin 88-04 were present in auxiliary feedwater pumps because of low settings in the minimum flow lines, but failed to take appropriate actions to minimize and manage, or to eliminate, the potential for pump damage.

This finding represented a performance deficiency because the licensee did not adequately address a degradation mechanism identified in NRC Bulletin 88-04, as required by the station's operating experience program. As a result, the auxiliary feedwater pumps continued to be operated with insufficient minimum flow to avoid unusual wear and aging without establishing increased monitoring and maintenance, or other compensating actions.

This issue was more than minor because it affected the equipment reliability objective of the Mitigating Systems cornerstone. This issue screened as Green during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18. This issue will be treated as a finding in accordance with Manual Chapter 0612: FIN 05000275, 323/2005006-08, Inadequate Response to Operating Experience for Auxiliary Feedwater Minimum Flow.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC

Item Type: NCV NonCited Violation

**No procedure for cross-tying trains of the diesel fuel oil transfer system**

A noncited violation was identified for not having a procedure to cross-tie fuel oil transfer trains in response to certain failures, contrary to the design and licensing basis of the system. The design and license basis of the diesel fuel oil transfer system credited the capability to cross-tie trains in order to meet requirements to maintain the system function and be able to withstand a worst-case single failure. The team identified that the licensee did not have a procedure or training to accomplish this task. Failure to incorporate design and licensing requirements into plant procedures was a violation of 10 CFR Part 50, Appendix B, Criterion III. This issue was entered into the licensee's corrective action program under Action Requests A0630010 and A0630015.

The failure to have a procedure needed to meet the design and license basis of the fuel oil transfer system was a performance deficiency. This finding was more than minor because it impacted the Mitigating Systems cornerstone objective of procedure quality to ensure the capability of the system, in that, the system would not be capable of supplying the emergency diesel generators for the required 7-day mission time in the event of a single failure. The team concluded that this would not result in a loss of function in accordance with Generic Letter 91-18; since procedures direct monitoring of fuel capacity, operators would be aware of the need for action for the following reasons: 1) there should be a relatively long time available to detect and correct the problem (in excess of 24 hours), 2) the expected actions are not complex, and 3) existing procedures require monitoring of the remaining fuel oil capacity during extended diesel runs. Therefore, this finding was determined to be of very low safety significance (Green) in Phase 1 of the significance determination process. The licensee took prompt compensatory measures to ensure the full mission time could be met.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005

Identified By: NRC

Item Type: NCV NonCited Violation

**Diesel fuel oil storage tank calculation did not adequately account for vortexing**

A noncited violation was identified for inadequate design control because the licensee did not properly account for vortex prevention in the calculation used to determine the usable volume in the diesel fuel oil storage tank, which could cause the pump to ingest air. The licensee was unable to locate a technical basis for this part of the calculation. The team independently calculated that 4.1 inches was necessary, compared to the 2.0 inches used in the calculation. The licensee performed a similar calculation and reached the same conclusion, which reduced the tanks' unusable volumes by a little less than 1,000 gallons in this 50,000 gallon tank. Failure to properly account for the unusable fuel oil storage tank volume necessary to prevent vortexing was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0629779.

The failure to properly evaluate and document the unusable volume of the diesel fuel oil storage tank needed to prevent vortexing and ingesting air into the transfer pump was a performance deficiency. Through calculations, the licensee was able to demonstrate that there was sufficient available margin in both the tank capacity and the existing technical specification requirement to account for this without affecting operability or necessitating a technical specification change. This finding affected the Mitigating Systems cornerstone. The issue is more than minor because it was similar to Example 3.i of Appendix E to Manual Chapter 0609, since it was necessary to re-perform a calculation to determine whether the existing condition was acceptable. The finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process, since there was available margin in the tank capacity and technical specification minimum required volume and it was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005

Identified By: NRC

Item Type: NCV NonCited Violation

**Failure to use the highest pressure in calculation to verify adequate auxiliary feedwater flow**

A noncited violation was identified for inadequate design control, because Calculation STA-135, "Auxiliary Feedwater System," Revision 2, which was intended to demonstrate that the auxiliary feedwater pumps have adequate capacity to meet their design basis, did not correctly identify the highest pressure under which the pumps needed to function. Specifically, the calculation did not account for the dynamic pressure loss between the feedwater inlet ring and the main steam safety valves. The licensee was able to perform an analysis that concluded the pumps had sufficient flow margin at the new pressure. Failure to properly translate the peak pressure against which the auxiliary feedwater pumps must deliver the required flow rate was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0630804.

The failure to document the capability of the auxiliary feedwater pumps to deliver the required flow at the maximum possible pressure was a performance deficiency. The issue is more than minor because a calculation was needed to determine whether the existing condition was acceptable, consistent with Example 3.i of Appendix E to Manual Chapter 0609. This issue affected the Mitigating Systems cornerstone. Because there was available margin in the pump capacity, this issue was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18. Therefore, the finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005

Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate power operated relief valve accumulator calculation**
A noncited violation was identified for inadequately translating design requirements into calculations used to demonstrate the capabilities of the pressurizer power operated relief valve backup accumulators. The calculation was found to contain a number of non-conservative errors and did not contain the most current acceptance criteria from accident analyses. As a result, this calculation failed to demonstrate that the backup nitrogen accumulators could operate the pressurizer power operated relief valves for the required number of cycles. Failure to properly demonstrate that design requirements for the number of power operated relief valve cycles needed to respond to an inadvertent safety injection actuation were satisfied through a design calculation was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Requests A0631420, A0630719 and A0630740.

The failure to adequately assess and document the capability of the backup accumulators to provide enough gas to operate the power operated relief valves through the required number of cycles was a performance deficiency. This issue was greater than minor because it was similar to Example 3.i in Manual Chapter 0612, Appendix E, in that, calculations had to be performed to demonstrate that the system could satisfy the accident analyses. This finding affected the Mitigating System cornerstone. This finding screened as having very low safety significance (Green) during a Phase 1 significance determination process, since the issue was confirmed to not have resulted in a loss of function in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Analyses did not demonstrate proper load sequencing with timer anomalies**
A noncited violation was identified for failure to demonstrate that load sequencing would satisfy regulatory requirements. The team identified that a single postulated fault occurring during load sequencing with offsite power available could restart load sequencing timers in all three engineered safety features buses and result in a more limiting scenario than previously analyzed by the licensee. This could result in overlaping starting transients for motors that were intended to start separately, which was not evaluated in existing calculations. The combined effects of this could cause later starting times for safety-related loads, potentially affecting system performance assumed in accident analyses. Failure to demonstrate that the system could perform as required considering a single fault was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request 0630036.

This failure to demonstrate through analyses that the electrical distribution system was capable of performing its required function following a single postulated fault was a performance deficiency. This issue was more than minor because it affected the Mitigating System cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a design basis accident. The licensee was subsequently able to demonstrate that there would be no loss of safety function even considering the effects of a fault as described above. Therefore, this finding was determined to be of very low safety significance (Green) in Phase I of the significance determination process.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Wire and Connect Test Equipment Resulted in Vital Bus De-Energization**
A self-revealing, noncited violation was identified for the failure to set up phase sequence test equipment according to procedure, as required by 10 CFR Part 50, Appendix B, Criterion V. This failure resulted in the momentary de-energization of Vital 4kV Bus G and the auto-start of Diesel Engine Generator 2-1. Subsequent investigation by Pacific Gas & Electric Company (PG&E) revealed that the primary side of the test transformer was wired in a wye configuration instead of a delta configuration. This wiring configuration introduced a direct short to ground, which caused the second level undervoltage relay to sense a degraded bus voltage for Vital 4kV Bus G. Subsequently, the relay removed the auxiliary power supply from Bus G and caused DEG 2-1 to start and load onto the bus. This finding involved a human performance cross-cutting aspect for the failure to wire the phase sequence test equipment properly for Vital 4kV Bus G and DEG 2-1.

The finding impacted the Mitigating Systems Cornerstone for ensuring the availability and capability of systems that respond to initiating events to prevent undesirable consequences that was associated with a pre-event human error performance. Considering Example 4.b of IMC 0612, Appendix E, the finding is greater than minor since the incorrect wiring and connection of the test equipment resulted in a vital bus de-energization and the actuation of DEG 2-1. Using Checklist 4 of Inspection Manual Chapter (IMC) 0609, Appendix G, Attachment 1, the finding did not result in the Technical Specifications for AC and DC power sources not being met and the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade PG&E's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade PG&E's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation

**Inadequate ASCO Valve Qualification Causes Plant Trip**
A self revealing violation of 10 CFR 50.49(f) was identified for the failure to maintain approximately 70 safety related solenoid operated valves in an environmentally qualified condition. On February 9, 2002, an age related ASCO solenoid operated valve failure caused a loss of steam generator feedwater event and a Unit 2 manual plant trip. Further, the licensee did not promptly evaluate the extent of condition of the ASCO failure (coil insulation failure), which delayed the identification of elastomer qualification issues for approximately 1 year. In a related finding, the team identified that the licensee had missed earlier opportunities to identify ASCO elastomer qualification issues, in that they failed to thoroughly evaluate several pertinent NRC information notices and previous valve failures. The failure to: 1) properly establish equipment qualification limits; 2) thoroughly evaluate plant events and failures; and 3) properly evaluate industry operating experience constituted performance concerns. PG&E entered this issue into their corrective action program as Action Request 0613008. These issues have cross-cutting aspects in the area of problem identification and resolution because the original problem investigation did not identify the full scope of the cause and extent of condition, delaying some important corrective actions for approximately 1 year.

This finding was greater than minor because, if left uncorrected, these deficiencies would become a more significant safety concern by increasing the failure rate as the components age. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated delta-CDF for the finding is 2.2E-8/yr. This violation was of very low risk significance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Promptly Identify Multiple Grounds in Containment Spray Pump 2-2 Control Circuitry**
A self-revealing, noncited 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to promptly identify multiple grounds in the breaker control circuitry for Containment Spray Pump 2-2. Specifically, Pacific Gas and Electric Company missed several opportunities, in part because of a failure to utilize the troubleshooting procedure, to pursue the cause of the ground and to address anomalous indications, the proximity of a known ground to other conductors, and operating experience. The grounds degraded control wires affecting the pump's manual/automatic breaker closure circuits, indication circuits, and overcurrent circuits for up to 70 days following the initial ground indication. A problem identification and resolution crosscutting aspect was identified for the troubleshooting and corrective actions associated with the grounds. The grounded cable was subsequently replaced. Similar to Example 4.f in Appendix E of Inspection Manual Chapter 0612, the finding is greater than minor because the multiple grounds affected the operability of containment spray pump 2-2. Using the Inspection Manual Chapter 0609 Phase I Screening Worksheet, the finding was of very low safety significance since there was not an actual reduction of the atmospheric pressure control function for containment.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Simulator with respect to Backup Seismic Alarm**
A noncited violation of 10 CFR 55.46 was identified by the inspectors for the failure to maintain the plant referenced simulator to respond to normal, transient and accident conditions. Pacific Gas and Electric Company removed from service, and abandoned the Backup Seismic System (Terra Tech Instrument) in place in June 2000. However, as of August 31, 2004, the plant referenced simulator still provided an annunciator fed from the backup seismic system when an earthquake of sufficient magnitude was felt. This provided operators with negative training in that operators were trained that the backup seismic system would provide annunciation and indication.

This finding affects the mitigating systems cornerstone and is greater than minor because it results in negative training of the operators to expect an annunciator from a backup seismic system in the event of an earthquake, if the earthquake force monitor was unavailable. Using the flow chart of Appendix I, of Inspection Manual Chapter 0609 of the Significance Determination Process, this issue affects operator actions in that operators may attempt to obtain ground motion from backup seismic monitors that did not exist. Per Inspection Manual Chapter 0609, Appendix I, Item 12, the inspectors determined that the finding was Green because the differences between the plant control room and the plant reference simulator negatively impacted operator actions and resulted in negative training.

Inspection Report# : 2004004*(pdf)*

**Significance:** G Jun 25, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design and Test Controls of the Diesel Emergency Generator Fuel Oil Level Control Valves**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, for the failure to maintain design control of the diesel emergency generator system fuel oil transfer system requirements. Specifically, the fuel supply to each diesel required that an adequate air supply to operate the air-operated day tank level control valve be maintained in the starting air receivers. The team identified that when the licensee recognized that this design basis was not documented, a calculation was performed to support creating the design basis which did not account for operational leakage from the system, nor did it verify that existing leakage would not prevent fulfilling the safety function. This failure potentially affected the ability of each diesel emergency generator to provide sufficient fuel oil to support 7 days of continuous diesel generator operations following a loss of offsite power. This issue was entered into the corrective action program under Action Request

A0613008. This finding involved cross-cutting aspects in the area of problem identification and resolution because the original corrective actions did not correct the problem and properly establish the design basis.

This finding was greater than minor because it was similar to Example 3.i of Manual Chapter 0612, Appendix E. This finding affected the mitigating systems cornerstone. This finding was evaluated using NRC Manual Chapter 0609, Significance Determination Process, Phase 1 worksheet under the mitigating systems cornerstone. The finding was determined to be of very low safety significance because the deficiency was confirmed not to result in a loss of function of the diesel engine generator as a power source per Generic Letter 91-18, Revision 1. The licensee was able to demonstrate that compensatory measures were in place so that this function could be performed manually in a reliable manner because operators would receive a control room alarm which triggered implementation of proceduralized step to manually perform the function. The team confirmed that operators were trained to perform this action.
Inspection Report# : 2004006*(pdf)*

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## Barrier Integrity

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to properly pre-plan CRVS maintenance when opening the control room boundary**
Two examples of a self-revealing violation of Technical Specification 5.4.1.a were identified for failure to pre-plan maintenance associated with the Control Room Ventilation System (CRVS). On January 4 and February 1, 2005, both trains of CRVS were rendered inoperable for short periods of time when the system was opened to the atmosphere for maintenance without pre-planning the administrative controls prescribed by Technical Specification 3.7.10, because of personnel error. Technical Specification 3.7.10 states that the control room boundary may be opened intermittently under administrative controls, and that if two trains of CRVS are inoperable because of the control room boundary being open, then the system must be restored to operability within 24 hours. The Bases for Technical Specification 3.7.10 states that proper administrative controls to invoke this Technical Specification exception consist of stationing a dedicated individual who is in continuous communication with the control room, who has a method of rapidly closing the control room boundary, and has been specifically trained on these duties. These administrative controls were not in place when the control room boundary was inadvertently opened on January 4 and February 1, 2005. A human performance crosscutting aspect was identified for failure to pre-plan maintenance associated with the CRVS that resulted in the control room boundary being opened without administrative controls.

This issue is more than minor and affects the Barrier Integrity Cornerstone, because it represents partial losses of function of the CRVS on two occasions. On January 4 (for 15 minutes) and February 1, 2005, (four hours) both trains of CRVS were rendered inoperable because of an opening in the CRVS boundary which would have prohibited pressurization of the control room. Although Technical Specification 3.7.10 allowed this condition for up to 24 hours, it only allows opening of the control room boundary under strict administrative controls, that were not in place. This issue screens to Green in accordance with Item 1 of the Containment Barriers Cornerstone Phase 1 review, because it constitutes a CRVS issue only, and is therefore of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Mislabel of Neutron Flux Detector Resulted in Neutronic Decoupling of Detector From the Core**
The inspectors identified a noncited violation for the failure to develop a core offload sequence that maintained the source range neutron flux monitors operable, as required by 10 CFR Part 50, Appendix B, Criterion V. Inaccurate labeling of two neutron detectors in the core offload planning tool resulted in the development of a core offload sequence that when implemented resulted in one of the detectors becoming neutronically uncoupled from the core during core alterations. A human performance crosscutting aspect was identified for the labeling error in the core offload planning. A second human performance crosscutting aspect was identified for the failure to ascertain the cause of the downward trend when first identified by the inspectors.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute for procedure quality which could impact cladding. The finding is more than minor when compared to Example 4.e of Inspection Manual Chapter 0612, Appendix E. Similar to the example, Procedure OP B-8DS1, Step 5.2.1, described a responding nuclear instrument as having at least one fuel assembly face-adjacent or diagonally adjacent to the detector. Due to a labeling error in the core offload planning tool, the core offload sequence was developed in a manner that caused a neutron detector (Detector N-52) not to have an adjacent fuel assembly. Using Checklist 4 of Inspection Manual Chapter 0609, Appendix G, Attachment 1, the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade Pacific Gas & Electric Company's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade Pacific Gas & Electric Company's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance.

Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Containment Fan Cooler Unit Reverse Rotation**
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly correct reverse rotation of containment fan cooler units (CFCUs) for both Units 1 and 2. PG&E observed reverse rotation of CFCUs for approximately 13 years, as a result of the CFCU backdraft dampers sticking partially open. The purpose of the backdraft dampers is to prevent reverse rotation of the CFCUs, which could cause the fan motor to trip on overcurrent when the CFCUs are started following a loss of coolant accident. Prior to Refueling Outage 2R12, 2 CFCUs in Unit 1 and 3 CFCUs in Unit 2 exhibited reverse rotation. One of the CFCUs in Unit 2 was considered inoperable due to reverse rotation and another was only considered operable if it was running.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute. The finding is more than minor when considering Example 3.g of IMC 0612, Appendix E. Similar to the example, PG&E observed reverse rotation of CFCUs for 13 years, and the reverse rotation impacted the operability of the CFCUs. Using the SDP Phase 1 Screening Worksheet from IMC 0609, the finding was determined to be of very low safety significance since it was determined that there was not an actual loss of defense-in-depth in containment pressure control or hydrogen control .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: FIN Finding
**Failure to properly implement an operating instruction for an inoperable containment isolation valve.**
The inspectors identified a finding for the failure to properly isolate containment isolation Valve VAC-2-FCV-681(an air-operated containment isolation valve) after it failed to fully stroke open and was declared inoperable. Operators hung administrative tags on the control room switch for the valve but failed to remove the motive force from the valve by isolating air to the actuator. The associated operating instruction required that the valve be closed and deactivated. A human performance crosscutting aspect was identified for the failure to properly implement the operating instruction for an inoperable containment isolation valve.

This issue affects the barrier integrity cornerstone objective to ensure that systems penetrating the containment and are connected directly to the containment atmosphere have adequate isolation to protect the containment barrier. This issue is greater than minor because failure to properly close and deactivate containment isolation valves could have an actual impact on the ability to isolate a fault outside of containment. Using the Phase 1 significance determination process, the inspectors determined that the issue was of very low safety significance because the finding did not represent an actual open significant pathway to the environment and the penetration was isolated by an active valve having secured flow.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Change to the Emergency Plan with respect to Backup Seismic System**
A violation of 10CFR 50.54(q) was identified by the inspectors for failure to update and submit changes to the emergency plan within 30 days. Specifically, Section 7.5.1 of the Diablo Canyon Emergency Plan stated that a supplemental seismic system, supplied by Terra Tech Corporation, provided backup local indication and control room annunciation on strong ground motion. The Terra Tech system was removed from service, along with its annunciation in the control room, and abandoned in place in July of 2000, but as of September 30, 2004, Pacific Gas and Electric had not revised its emergency plan to reflect this change.

The finding was evaluated using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," Section IV, because licensee reductions in the effectiveness of its emergency plan impact the regulatory process. The finding had greater than minor significance because deletion of conditions indicative of a site area emergency has the potential to impact safety. The finding was determined to be a noncited Severity Level IV violation because the finding involved a violation of a regulatory requirement and did not constitute a failure to meet an emergency planning standard as defined by 10 CFR 50.47(b). This finding has been entered into the licensee's corrective action program as Action Request A0618799.
Inspection Report# : 2004004*(pdf)*

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## Emergency Preparedness

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Establish Compensatory Measures to Ensure the Implementation of the Diablo Canyon Emergency Plan**
The inspectors identified a violation of 10 CFR 50.54(q) and 50.47.b(4) for the failure to maintain the seismic force monitors during the periods, June 16-19,1999, December 1-4, 2000, April 25-27, 2002, May 25-29, 2002, November 6-8, 2003, December 30-31, 2003, and August 9-10, 2004, such that the emergency plan designed to meet planning standard (4) in 10 CFR 50.47(b) could be promptly implemented. Specifically, PG&E failed to provide a means for the emergency director to promptly classify seismic events at the notification of unusual event, alert or site area emergency levels, while the seismic force monitor utilized by the operators (emergency director) was out of service or being replaced. This finding had a human performance cross-cutting aspect associated with identifying compensatory measures to address the removal of the earthquake force monitors.

This performance deficiency impacted the emergency preparedness cornerstone because PG&E did not meet an emergency planning requirement and the cause was reasonably within PG&E 's control and should have been prevented. It is greater than minor because it has a potential to impact safety and because it was not a record keeping or administrative issue or an insignificant procedural error. This deficiency could have affected the EP Cornerstone objective of ensuring the capability to implement measures to protect the health and safety of the public during an emergency, and is associated with attributes of facilities and equipment, and offsite emergency preparedness. The finding is evaluated using the Emergency Preparedness "Failure to Comply" flowchart of the SDP and is a violation of 10 CFR 50.54(q) and planning standard 50.47(b)(4), which states, in part, that a standard emergency action level and classification system... is in use Utilizing the Failure to Comply Flow Chart in Manual Chapter 0609, the performance deficiency does not result in a failure of the risk significant planning standard (RSPS) or a degraded RSPS in that the unavailability of the seismic monitors would not prevent the declaration of a Site Area Emergency, Alert or Notification of Unusual Event .
Inspection Report# : 2004005*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Perform an Adequate Survey to Evaluate Radiological Hazards**
A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified when the licensee failed to perform an adequate survey to evaluate the radiological hazards associated with venting the steam generator exhaust into containment during the Unit 2 refueling outage. On February 7, 2003, the licensee failed to take air samples to account for the decay of tellurium-132 into iodine-132 in the steam generator exhaust prior to venting into the containment building. Consequently, fifty-two workers in containment received unplanned and unintended low-level intakes (less than 10 millirem) of iodine-132. This issue has been entered into the licensee's corrective action program as Action Request No. A0628334.

The failure to perform a survey to evaluate radiological hazards is a performance deficiency. The finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of Exposure Control and involved unplanned and unintended dose to workers that resulted from actions contrary to NRC requirements. Therefore the Occupational Radiation Safety Significance Determination Process was used to analyze the significance of the finding which was determined to be of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.
Inspection Report# : 2004009*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Lock a High Radiation Area with Dose Rates Greater than 1 Rem per Hour**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent unauthorized entry of a portion of the whole body into a high radiation area with dose rates greater than 1 rem per hour. Specifically, on November 14, 2004, PG&E failed to use an effective locking mechanism on the lower access flaps of the primary steam generator shield doors. The ineffective locking mechanism was discovered two days later when workers went to remove suction hoses. This could have allowed an individual to expose the arm above the elbow to dose rates greater than 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Access a High Radiation Area with Dose Rates Greater than 1 Rem per Hour with the Correct Radiation Work Permit**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent two individuals from entering a high radiation area with dose rates greater than 1 rem per hour on the incorrect radiation work permit. Two individuals entered an area with dose rates of 6 rem per hour in Reactor Coolant Pump Cubicle 2-4 using a radiation work permit which only allowed entry into areas with dose rates up to 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Aug 12, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Perform Radiological Survey of a High Radiation Area**
Green. A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified for the failure to perform required radiation surveys in Unit 2 to ensure compliance with 10 CFR 20.1902(b). Specifically, on January 28, 2003, during the performance of venting the volume control tank radiation protection personnel failed to perform adequate surveys of the Unit 2 Gas Decay Tank Room to post an expected high radiation area that would occur during this evolution. This finding involved cross-cutting aspects in the area of problem identification and resolution because the team noted that corrective actions for a similar event under the same circumstances had been ineffective in preventing recurrence. This issue was entered into the corrective action program under Action Request A0572997.

The finding is greater than minor because it was associated with one of the occupational radiation safety cornerstone attributes (exposure), and the finding affected the associated cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material. The inspector processed the issues through the Occupational Radiation Protection Significance Determination Process. This issues were determined to be a Green finding because it was not an ALARA planning and control issue, there was no personnel overexposure or substantial potential for personnel overexposure, and the licensee's ability to assess dose was not compromised.
Inspection Report# : 2004006*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to control radioactive material contained in certain generally-licensed devices in accordance with 10 CFR 31.5**
The team identified a non-cited violation of 10 CFR 31.5(c) because the licensee failed to maintain a program for generally-licensed radioactive devices used for reactor operations in accordance with the regulatory requirements. The licensee failed to implement a program for the use of generally-licensed devices used for monitoring personnel, and consequently failed to maintain and test 14 radioactive sources housed within the generally-licensed devices. Specifically, the licensee had not (1) conducted contamination leak tests on the device and the 10-millicurie Nickel-63 source housed in each device at the required frequency and (2) assigned an individual with the regulatory knowledge or authority to ensure compliance with 10 CFR 31.5. This issue has been entered into the licensee's corrective action program as Action Request A0628345.

The licensee's failure to control generally-licensed devices containing radioactive material in accordance with 10 CFR 31.5 was a performance deficiency. The finding was more than minor because it affected the Public Radiation Safety cornerstone attribute and affected the associated cornerstone objective. In order to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain, the licensee is required to leak test each generally-licensed device. Using the Public Radiation Safety Significance Determination Process, the finding had very low safety significance (Green) because: (1) it was not a transportation issue, (2) public exposure was not more than 5 millirem, and (3) there were not more than five occurrences. This finding also had crosscutting aspects associated with the effectiveness of problem identification and resolution.
Inspection Report# : 2004009*(pdf)*

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## Physical Protection

Physical Protection information not publicly available.

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# Miscellaneous

**Significance: N/A** Jun 25, 2004
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**

The team concluded that the licensee was effective in identifying, evaluating, and correcting problems, although the team identified some examples were identified where conditions adverse to quality were not properly entered into the Action Request system, allowing problem recurrence. The team found that the evaluation and prioritization of problems were mostly conducted properly, although some significant issues were identified as routine because the licensee's process assigned significance based on the actual consequences of problems, rather than considering the potential consequences under design basis conditions. Corrective actions were generally implemented in a timely manner. However, the team found weaknesses with the alignment of corrective actions with the cause, and with the quality of operability evaluations for issues assigned routine significance, because the licensee did not assign a probable cause statement to routine issues. Licensee audits and assessments were found to be responsive to plant performance issues and effective in identifying areas for improvement. During interviews, station personnel communicated a willingness to enter issues into the corrective action program. The team reviewed the licensee's improvement plans for significant cross-cutting issues in human performance and problem identification and resolution. Although it was too early to determine if these will be effective, the team noted that the Human Performance Improvement Plan did not address problems observed in coordinating and supervising operations during outages.

Inspection Report# : 2004006*(pdf)*

Last modified : June 17, 2005

# Diablo Canyon 2
# 2Q/2005 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Properly Implement Procedure for Spent Fuel Pool Skimmer Filter Replacement**
A self-revealing NCV was identified for the failure to appropriately implement the procedure for spent fuel pool skimmer filter replacement, as required by Technical Specification 5.4.1.a. On December 23, 2004, operators cleared the spent fuel pool skimmer system using Section 6.3.1 of Procedure OP B-7:III, "Spent Fuel Pool System - Shutdown and Clearing and Filter Replacement," Revision 15, instead of the appropriate section, which was Section 6.3.2. A human performance cross cutting aspect was identified for the failure on two occasions to address configuration control concerns with the system.

This finding impacted the Initiating Events Cornerstone and was considered more than minor using Example 5.a of IMC 0612. Specifically, Valve SFS-2-3 was mis-positioned due to the use of the wrong section of Procedure OP B-7:III and then returned to service. Additionally, operators had two opportunities to identify the mis-positioning of Valve SFS-2-3 but failed to identify the condition. The mis-positioned valve resulted in a loss of approximately 36,000 gallons of water from the spent fuel pool. Using the SDP Phase 1 screening worksheet of IMC 0609, Appendix A, the finding was evaluated as a transient initiator, and it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Therefore, the finding was screened as having very low safety significance
Inspection Report# : 2004005*(pdf)*

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## Mitigating Systems

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to correct fire program violation concerning qualifications of Operations Responders in support of the fire brigade**
The inspectors identified a noncited violation of Technical Specification 5.4.1.d for failure to implement procedures for Fire Protection Implementation, because of failure to provide adequate training for operations fire responders. Procedure OM8, "Fire Protection Program," Revision 2B, Section 7.8 states, in part, that quality problems associated with the Fire Protection Program shall be documented and resolved in accordance with Procedure OM7 "Corrective Action," Revision 2B. Section 9.5.1 of the Final Safety Analysis Report states that measures are established to ensure conditions adverse to fire protection are identified, reported and corrected, and that administrative procedures are established to implement this requirement. Contrary to the above, Pacific Gas & Electric Company did not adequately implement and maintain a procedure for fire protection. Specifically, Pacific Gas & Electric Company failed to adequately resolve a condition adverse to fire protection in accordance with Procedure OM7. As of March 1, 2005, operations responders were not required to participate in fire drills for initial qualification or maintenance of qualification, as was noted as a qualification deficiency in Non-cited Violation 50-275;323/2003-08-01, and Action Request (AR) A0600934. This finding has problem identification and resolution cross cutting aspects for failure to correct operations responder training deficiencies.

The performance deficiency associated with this finding is a failure to adequately implement the fire protection program with respect to the qualifications of the fire brigade operations responder. The finding impacted the mitigating systems cornerstone and was more than minor since there was an adverse impact to a fire protection defense-in-depth element. Using the Significance Determination Process (SDP) Phase I Screening Worksheet and the SDP Phase II Notebook in Appendix F of Inspection Manual Chapter (IMC) 0609, the inspectors determined that the finding was of very low safety significance. Specifically, the significance of the finding was evaluated by considering fire scenarios in the vital 4 kV Bus F switchgear room and auxiliary saltwater Pump 1-1 vault. These two areas have the highest dependence on fire brigade response since they have the highest fire ignition frequency for areas that do not have automatic fire suppression. The inspectors evaluated the risk-significance using half the nominal credit for manual fire suppression as a result of the finding. Using Tables 5.4, 5.5, and 5.6 of IMC 0609, both fire scenarios screened as very low safety significance. Since the two fire scenarios were considered worst-case for the finding, the inspectors determined that the finding was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Mar 31, 2005

Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to promptly correct diesel engine generator lube oil instrument line crack**
The inspectors identified a noncited violation for the failure to promptly correct a cracked lube oil instrument sensing line, as required by 10 CFR Part 50, Appendix B, Criterion XVI. On August 29, 2004, operators observed a lube oil leak from the weld connecting the outlet of Valve DEG-2-1084 to instrument tubing. Approximately one month later, the leak had increased and it was discovered that the circumferential crack was 180 degrees through-wall on the weld. As a result, there was an increased potential for diesel engine generator (DEG) 2-3 to trip on low lube oil level. The finding had problem identification and resolution crosscutting aspects associated with operations and engineering personnel not recognizing the significance of the degraded condition and not implementing timely corrective actions.

This finding impacted the Mitigating Systems Cornerstone for reliability of systems that respond to initiating events to prevent undesirable consequences, and it affects the equipment performance attribute. The finding was more than minor using Example 4.f of Inspection Manual Chapter 0612, Appendix E. Similar to Example 4.f, the inspectors determined that there was impact to DEG 2-3 operability. Using the SDP Phase 1 screening worksheets in Appendix A of Inspection Manual Chapter 0609, the finding was determined to have potentially greater than very low safety significance because the failure could have resulted in an actual loss of diesel engine Generator 2-3 during a loss of offsite power event. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated conditional core damage frequency was 1.2E-7/yr. This violation was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding
**Incomplete action for setting auxiliary feedwater pump minimum flow values**
The team identified a Green finding for inadequate response to industry operating experience regarding establishing minimum flow for the auxiliary feedwater pumps. The team concluded that the licensee recognized that the conditions reported in NRC Bulletin 88-04 were present in auxiliary feedwater pumps because of low settings in the minimum flow lines, but failed to take appropriate actions to minimize and manage, or to eliminate, the potential for pump damage.

This finding represented a performance deficiency because the licensee did not adequately address a degradation mechanism identified in NRC Bulletin 88-04, as required by the station's operating experience program. As a result, the auxiliary feedwater pumps continued to be operated with insufficient minimum flow to avoid unusual wear and aging without establishing increased monitoring and maintenance, or other compensating actions.

This issue was more than minor because it affected the equipment reliability objective of the Mitigating Systems cornerstone. This issue screened as Green during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18. This issue will be treated as a finding in accordance with Manual Chapter 0612: FIN 05000275, 323/2005006-08, Inadequate Response to Operating Experience for Auxiliary Feedwater Minimum Flow.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding
**Diesel fuel oil transfer modification did not adequately assess reliability impact**
A finding was identified for modifying the diesel fuel oil transfer system without properly assessing the resulting net affect on reliability from introducing a new failure potential associated with new active components. As a result, the licensee rejected a small design change, which would have eliminated the failure mode when it was recognized that failure of the new pressure control valves could fail the train. Because the failure potential was not fully assessed, the licensee decided not to implement a change that would have eliminated the impact of the failure, nor were the pressure control valves subject to any preventive maintenance to ensure their reliability. This issue was entered into the licensee's corrective action program under Action Request A0630383.

The failure to properly assess the net effect on system reliability and risk due to the positive and negative effects of this modification, or to mitigate or eliminate a new failure mode created by the modification was a performance deficiency. This issue is more than minor because it affected the design control attribute of the Mitigating Systems cornerstone objective to assure the reliability and capability of equipment needed for accident mitigation. This finding was determined to be of very low safety significance (Green) during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18 based on test results.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC

Item Type: NCV NonCited Violation

**No procedure for cross-tying trains of the diesel fuel oil transfer system**

A noncited violation was identified for not having a procedure to cross-tie fuel oil transfer trains in response to certain failures, contrary to the design and licensing basis of the system. The design and license basis of the diesel fuel oil transfer system credited the capability to cross-tie trains in order to meet requirements to maintain the system function and be able to withstand a worst-case single failure. The team identified that the licensee did not have a procedure or training to accomplish this task. Failure to incorporate design and licensing requirements into plant procedures was a violation of 10 CFR Part 50, Appendix B, Criterion III. This issue was entered into the licensee's corrective action program under Action Requests A0630010 and A0630015.

The failure to have a procedure needed to meet the design and license basis of the fuel oil transfer system was a performance deficiency. This finding was more than minor because it impacted the Mitigating Systems cornerstone objective of procedure quality to ensure the capability of the system, in that, the system would not be capable of supplying the emergency diesel generators for the required 7-day mission time in the event of a single failure. The team concluded that this would not result in a loss of function in accordance with Generic Letter 91-18; since procedures direct monitoring of fuel capacity, operators would be aware of the need for action for the following reasons: 1) there should be a relatively long time available to detect and correct the problem (in excess of 24 hours), 2) the expected actions are not complex, and 3) existing procedures require monitoring of the remaining fuel oil capacity during extended diesel runs. Therefore, this finding was determined to be of very low safety significance (Green) in Phase 1 of the significance determination process. The licensee took prompt compensatory measures to ensure the full mission time could be met.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005

Identified By: NRC

Item Type: NCV NonCited Violation

**Diesel fuel oil storage tank calculation did not adequately account for vortexing**

A noncited violation was identified for inadequate design control because the licensee did not properly account for vortex prevention in the calculation used to determine the usable volume in the diesel fuel oil storage tank, which could cause the pump to ingest air. The licensee was unable to locate a technical basis for this part of the calculation. The team independently calculated that 4.1 inches was necessary, compared to the 2.0 inches used in the calculation. The licensee performed a similar calculation and reached the same conclusion, which reduced the tanks' unusable volumes by a little less than 1,000 gallons in this 50,000 gallon tank. Failure to properly account for the unusable fuel oil storage tank volume necessary to prevent vortexing was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0629779.

The failure to properly evaluate and document the unusable volume of the diesel fuel oil storage tank needed to prevent vortexing and ingesting air into the transfer pump was a performance deficiency. Through calculations, the licensee was able to demonstrate that there was sufficient available margin in both the tank capacity and the existing technical specification requirement to account for this without affecting operability or necessitating a technical specification change. This finding affected the Mitigating Systems cornerstone. The issue is more than minor because it was similar to Example 3.i of Appendix E to Manual Chapter 0609, since it was necessary to re-perform a calculation to determine whether the existing condition was acceptable. The finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process, since there was available margin in the tank capacity and technical specification minimum required volume and it was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005

Identified By: NRC

Item Type: NCV NonCited Violation

**Failure to use the highest pressure in calculation to verify adequate auxiliary feedwater flow**

A noncited violation was identified for inadequate design control, because Calculation STA-135, "Auxiliary Feedwater System," Revision 2, which was intended to demonstrate that the auxiliary feedwater pumps have adequate capacity to meet their design basis, did not correctly identify the highest pressure under which the pumps needed to function. Specifically, the calculation did not account for the dynamic pressure loss between the feedwater inlet ring and the main steam safety valves. The licensee was able to perform an analysis that concluded the pumps had sufficient flow margin at the new pressure. Failure to properly translate the peak pressure against which the auxiliary feedwater pumps must deliver the required flow rate was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0630804.

The failure to document the capability of the auxiliary feedwater pumps to deliver the required flow at the maximum possible pressure was a performance deficiency. The issue is more than minor because a calculation was needed to determine whether the existing condition was acceptable, consistent with Example 3.i of Appendix E to Manual Chapter 0609. This issue affected the Mitigating Systems cornerstone. Because there was available margin in the pump capacity, this issue was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18. Therefore, the finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005

Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate power operated relief valve accumulator calculation**
A noncited violation was identified for inadequately translating design requirements into calculations used to demonstrate the capabilities of the pressurizer power operated relief valve backup accumulators. The calculation was found to contain a number of non-conservative errors and did not contain the most current acceptance criteria from accident analyses. As a result, this calculation failed to demonstrate that the backup nitrogen accumulators could operate the pressurizer power operated relief valves for the required number of cycles. Failure to properly demonstrate that design requirements for the number of power operated relief valve cycles needed to respond to an inadvertent safety injection actuation were satisfied through a design calculation was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Requests A0631420, A0630719 and A0630740.

The failure to adequately assess and document the capability of the backup accumulators to provide enough gas to operate the power operated relief valves through the required number of cycles was a performance deficiency. This issue was greater than minor because it was similar to Example 3.i in Manual Chapter 0612, Appendix E, in that, calculations had to be performed to demonstrate that the system could satisfy the accident analyses. This finding affected the Mitigating System cornerstone. This finding screened as having very low safety significance (Green) during a Phase 1 significance determination process, since the issue was confirmed to not have resulted in a loss of function in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Analyses did not demonstrate proper load sequencing with timer anomalies**
A noncited violation was identified for failure to demonstrate that load sequencing would satisfy regulatory requirements. The team identified that a single postulated fault occurring during load sequencing with offsite power available could restart load sequencing timers in all three engineered safety features buses and result in a more limiting scenario than previously analyzed by the licensee. This could result in overlaping starting transients for motors that were intended to start separately, which was not evaluated in existing calculations. The combined effects of this could cause later starting times for safety-related loads, potentially affecting system performance assumed in accident analyses. Failure to demonstrate that the system could perform as required considering a single fault was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request 0630036.

This failure to demonstrate through analyses that the electrical distribution system was capable of performing its required function following a single postulated fault was a performance deficiency. This issue was more than minor because it affected the Mitigating System cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a design basis accident. The licensee was subsequently able to demonstrate that there would be no loss of safety function even considering the effects of a fault as described above. Therefore, this finding was determined to be of very low safety significance (Green) in Phase I of the significance determination process.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Jan 27, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Corrective Actions to Prevent Recurrence for Pressurizer Safety Valve Out-of-Tolerance Lift Setpoints**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI for failure to take corrective actions to prevent recurrence for a significant condition adverse to quality. On January 27, 2005, Pacific Gas and Electric Co. identified that the Unit 2 pressurizer safety valve lift setpoints were determined to be significantly out-of-tolerance, as compared to historical and industry-wide experience. However, Pacific Gas and Electric Co. failed to identify the root cause and propose any corrective actions to prevent recurrence, despite a history of pressurizer safety valve lift setpoints being out-of-tolerance.

The finding impacted the Mitigating Systems Cornerstone and was determined to be more than minor because it impacted the cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance since it did not represent an actual loss of safety function, represent an actual loss of a safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. Specifically, analysis demonstrated that the two valves having lift setpoints 4.4 and 3.6 percent low would not adversely affect the proper lift of the power-operated relief valves, and would not result in a spurious lift of the pressurizer safety valves during a normal transient.

Inspection Report# : 2005003*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Wire and Connect Test Equipment Resulted in Vital Bus De-Energization**

A self-revealing, noncited violation was identified for the failure to set up phase sequence test equipment according to procedure, as required by 10 CFR Part 50, Appendix B, Criterion V. This failure resulted in the momentary de-energization of Vital 4kV Bus G and the auto-start of Diesel Engine Generator 2-1. Subsequent investigation by Pacific Gas & Electric Company (PG&E) revealed that the primary side of the test transformer was wired in a wye configuration instead of a delta configuration. This wiring configuration introduced a direct short to ground, which caused the second level undervoltage relay to sense a degraded bus voltage for Vital 4kV Bus G. Subsequently, the relay removed the auxiliary power supply from Bus G and caused DEG 2-1 to start and load onto the bus. This finding involved a human performance cross-cutting aspect for the failure to wire the phase sequence test equipment properly for Vital 4kV Bus G and DEG 2-1.

The finding impacted the Mitigating Systems Cornerstone for ensuring the availability and capability of systems that respond to initiating events to prevent undesirable consequences that was associated with a pre-event human error performance. Considering Example 4.b of IMC 0612, Appendix E, the finding is greater than minor since the incorrect wiring and connection of the test equipment resulted in a vital bus de-energization and the actuation of DEG 2-1. Using Checklist 4 of Inspection Manual Chapter (IMC) 0609, Appendix G, Attachment 1, the finding did not result in the Technical Specifications for AC and DC power sources not being met and the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade PG&E's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade PG&E's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Inadequate ASCO Valve Qualification Causes Plant Trip**
A self revealing violation of 10 CFR 50.49(f) was identified for the failure to maintain approximately 70 safety related solenoid operated valves in an environmentally qualified condition. On February 9, 2002, an age related ASCO solenoid operated valve failure caused a loss of steam generator feedwater event and a Unit 2 manual plant trip. Further, the licensee did not promptly evaluate the extent of condition of the ASCO failure (coil insulation failure), which delayed the identification of elastomer qualification issues for approximately 1 year. In a related finding, the team identified that the licensee had missed earlier opportunities to identify ASCO elastomer qualification issues, in that they failed to thoroughly evaluate several pertinent NRC information notices and previous valve failures. The failure to: 1) properly establish equipment qualification limits; 2) thoroughly evaluate plant events and failures; and 3) properly evaluate industry operating experience constituted performance concerns. PG&E entered this issue into their corrective action program as Action Request 0613008. These issues have cross-cutting aspects in the area of problem identification and resolution because the original problem investigation did not identify the full scope of the cause and extent of condition, delaying some important corrective actions for approximately 1 year.

This finding was greater than minor because, if left uncorrected, these deficiencies would become a more significant safety concern by increasing the failure rate as the components age. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated delta-CDF for the finding is 2.2E-8/yr. This violation was of very low risk significance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Promptly Identify Multiple Grounds in Containment Spray Pump 2-2 Control Circuitry**
A self-revealing, noncited 10 CFR Part 50, Appendix B, Criterion XVI, was identified for the failure to promptly identify multiple grounds in the breaker control circuitry for Containment Spray Pump 2-2. Specifically, Pacific Gas and Electric Company missed several opportunities, in part because of a failure to utilize the troubleshooting procedure, to pursue the cause of the ground and to address anomalous indications, the proximity of a known ground to other conductors, and operating experience. The grounds degraded control wires affecting the pump's manual/automatic breaker closure circuits, indication circuits, and overcurrent circuits for up to 70 days following the initial ground indication. A problem identification and resolution crosscutting aspect was identified for the troubleshooting and corrective actions associated with the grounds. The grounded cable was subsequently replaced. Similar to Example 4.f in Appendix E of Inspection Manual Chapter 0612, the finding is greater than minor because the multiple grounds affected the operability of containment spray pump 2-2. Using the Inspection Manual Chapter 0609 Phase I Screening Worksheet, the finding was of very low safety significance since there was not an actual reduction of the atmospheric pressure control function for containment.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Simulator with respect to Backup Seismic Alarm**
A noncited violation of 10 CFR 55.46 was identified by the inspectors for the failure to maintain the plant referenced simulator to respond to normal, transient and accident conditions. Pacific Gas and Electric Company removed from service, and abandoned the Backup Seismic System (Terra Tech Instrument) in place in June 2000. However, as of August 31, 2004, the plant referenced simulator still provided an annunciator fed from the backup seismic system when an earthquake of sufficient magnitude was felt. This provided operators with negative training in that operators were trained that the backup seismic system would provide annunciation and indication.

This finding affects the mitigating systems cornerstone and is greater than minor because it results in negative training of the operators to expect an annunciator from a backup seismic system in the event of an earthquake, if the earthquake force monitor was unavailable. Using the flow chart of Appendix I, of Inspection Manual Chapter 0609 of the Significance Determination Process, this issue affects operator actions in that operators may attempt to obtain ground motion from backup seismic monitors that did not exist. Per Inspection Manual Chapter 0609, Appendix I, Item 12, the inspectors determined that the finding was Green because the differences between the plant control room and the plant reference simulator negatively impacted operator actions and resulted in negative training.

Inspection Report# : 2004004*(pdf)*

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# Barrier Integrity

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to properly pre-plan CRVS maintenance when opening the control room boundary**
Two examples of a self-revealing violation of Technical Specification 5.4.1.a were identified for failure to pre-plan maintenance associated with the Control Room Ventilation System (CRVS). On January 4 and February 1, 2005, both trains of CRVS were rendered inoperable for short periods of time when the system was opened to the atmosphere for maintenance without pre-planning the administrative controls prescribed by Technical Specification 3.7.10, because of personnel error. Technical Specification 3.7.10 states that the control room boundary may be opened intermittently under administrative controls, and that if two trains of CRVS are inoperable because of the control room boundary being open, then the system must be restored to operability within 24 hours. The Bases for Technical Specification 3.7.10 states that proper administrative controls to invoke this Technical Specification exception consist of stationing a dedicated individual who is in continuous communication with the control room, who has a method of rapidly closing the control room boundary, and has been specifically trained on these duties. These administrative controls were not in place when the control room boundary was inadvertently opened on January 4 and February 1, 2005. A human performance crosscutting aspect was identified for failure to pre-plan maintenance associated with the CRVS that resulted in the control room boundary being opened without administrative controls.

This issue is more than minor and affects the Barrier Integrity Cornerstone, because it represents partial losses of function of the CRVS on two occasions. On January 4 (for 15 minutes) and February 1, 2005, (four hours) both trains of CRVS were rendered inoperable because of an opening in the CRVS boundary which would have prohibited pressurization of the control room. Although Technical Specification 3.7.10 allowed this condition for up to 24 hours, it only allows opening of the control room boundary under strict administrative controls, that were not in place. This issue screens to Green in accordance with Item 1 of the Containment Barriers Cornerstone Phase 1 review, because it constitutes a CRVS issue only, and is therefore of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Mislabel of Neutron Flux Detector Resulted in Neutronic Decoupling of Detector From the Core**
The inspectors identified a noncited violation for the failure to develop a core offload sequence that maintained the source range neutron flux monitors operable, as required by 10 CFR Part 50, Appendix B, Criterion V. Inaccurate labeling of two neutron detectors in the core offload planning tool resulted in the development of a core offload sequence that when implemented resulted in one of the detectors becoming neutronically uncoupled from the core during core alterations. A human performance crosscutting aspect was identified for the labeling error in the core offload planning. A second human performance crosscutting aspect was identified for the failure to ascertain the cause of the downward trend when first identified by the inspectors.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute for procedure quality which could impact cladding. The finding is more than minor when compared to Example 4.e of Inspection Manual Chapter 0612, Appendix E. Similar to the example, Procedure OP B-8DS1, Step 5.2.1, described a responding nuclear instrument as having at least one fuel assembly face-adjacent or diagonally adjacent to the detector. Due to a labeling error in the core offload planning tool, the core offload sequence was developed in a manner that caused a neutron detector (Detector N-52) not to have an adjacent fuel assembly. Using Checklist 4 of Inspection Manual Chapter 0609, Appendix G, Attachment 1, the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade Pacific Gas & Electric Company's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade Pacific Gas & Electric Company's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Containment Fan Cooler Unit Reverse Rotation**
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly correct reverse rotation of containment fan cooler units (CFCUs) for both Units 1 and 2. PG&E observed reverse rotation of CFCUs for approximately 13 years, as a result of the CFCU backdraft dampers sticking partially open. The purpose of the backdraft dampers is to prevent reverse rotation of the CFCUs, which could cause the fan motor to trip on overcurrent when the CFCUs are started following a loss of coolant accident. Prior to Refueling Outage 2R12, 2 CFCUs in Unit 1 and 3 CFCUs in Unit 2 exhibited reverse rotation. One of the CFCUs in Unit 2 was considered inoperable due to reverse rotation and another was only considered operable if it was running.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute. The finding is more than minor when considering Example 3.g of IMC 0612, Appendix E. Similar to the example, PG&E observed reverse rotation of CFCUs for 13 years, and the reverse rotation impacted the operability of the CFCUs. Using the SDP Phase 1 Screening Worksheet from IMC 0609, the finding was determined to be of very low safety significance since it was determined that there was not an actual loss of defense-in-depth in containment pressure control or hydrogen control .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: FIN Finding
**Failure to properly implement an operating instruction for an inoperable containment isolation valve.**
The inspectors identified a finding for the failure to properly isolate containment isolation Valve VAC-2-FCV-681(an air-operated containment isolation valve) after it failed to fully stroke open and was declared inoperable. Operators hung administrative tags on the control room switch for the valve but failed to remove the motive force from the valve by isolating air to the actuator. The associated operating instruction required that the valve be closed and deactivated. A human performance crosscutting aspect was identified for the failure to properly implement the operating instruction for an inoperable containment isolation valve.

This issue affects the barrier integrity cornerstone objective to ensure that systems penetrating the containment and are connected directly to the containment atmosphere have adequate isolation to protect the containment barrier. This issue is greater than minor because failure to properly close and deactivate containment isolation valves could have an actual impact on the ability to isolate a fault outside of containment. Using the Phase 1 significance determination process, the inspectors determined that the issue was of very low safety significance because the finding did not represent an actual open significant pathway to the environment and the penetration was isolated by an active valve having secured flow.
Inspection Report# : 2004004*(pdf)*

**Significance:** G Sep 30, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Change to the Emergency Plan with respect to Backup Seismic System**
A violation of 10CFR 50.54(q) was identified by the inspectors for failure to update and submit changes to the emergency plan within 30 days. Specifically, Section 7.5.1 of the Diablo Canyon Emergency Plan stated that a supplemental seismic system, supplied by Terra Tech Corporation, provided backup local indication and control room annunciation on strong ground motion. The Terra Tech system was removed from service, along with its annunciation in the control room, and abandoned in place in July of 2000, but as of September 30, 2004, Pacific Gas and Electric had not revised its emergency plan to reflect this change.

The finding was evaluated using NUREG-1600, "General Statement of Policy and Procedure for NRC Enforcement Actions," Section IV, because licensee reductions in the effectiveness of its emergency plan impact the regulatory process. The finding had greater than minor significance because deletion of conditions indicative of a site area emergency has the potential to impact safety. The finding was determined to be a noncited Severity Level IV violation because the finding involved a violation of a regulatory requirement and did not constitute a failure to meet an emergency planning standard as defined by 10 CFR 50.47(b). This finding has been entered into the licensee's corrective action program as Action Request A0618799.
Inspection Report# : 2004004*(pdf)*

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# Emergency Preparedness

**Significance:** G Dec 31, 2004
Identified By: NRC

Item Type: NCV NonCited Violation
**Failure to Establish Compensatory Measures to Ensure the Implementation of the Diablo Canyon Emergency Plan**
The inspectors identified a violation of 10 CFR 50.54(q) and 50.47.b(4) for the failure to maintain the seismic force monitors during the periods, June 16-19,1999, December 1-4, 2000, April 25-27, 2002, May 25-29, 2002, November 6-8, 2003, December 30-31, 2003, and August 9-10, 2004, such that the emergency plan designed to meet planning standard (4) in 10 CFR 50.47(b) could be promptly implemented. Specifically, PG&E failed to provide a means for the emergency director to promptly classify seismic events at the notification of unusual event, alert or site area emergency levels, while the seismic force monitor utilized by the operators (emergency director) was out of service or being replaced. This finding had a human performance cross-cutting aspect associated with identifying compensatory measures to address the removal of the earthquake force monitors.

This performance deficiency impacted the emergency preparedness cornerstone because PG&E did not meet an emergency planning requirement and the cause was reasonably within PG&E 's control and should have been prevented. It is greater than minor because it has a potential to impact safety and because it was not a record keeping or administrative issue or an insignificant procedural error. This deficiency could have affected the EP Cornerstone objective of ensuring the capability to implement measures to protect the health and safety of the public during an emergency, and is associated with attributes of facilities and equipment, and offsite emergency preparedness. The finding is evaluated using the Emergency Preparedness "Failure to Comply" flowchart of the SDP and is a violation of 10 CFR 50.54(q) and planning standard 50.47(b)(4), which states, in part, that a standard emergency action level and classification system... is in use Utilizing the Failure to Comply Flow Chart in Manual Chapter 0609, the performance deficiency does not result in a failure of the risk significant planning standard (RSPS) or a degraded RSPS in that the unavailability of the seismic monitors would not prevent the declaration of a Site Area Emergency, Alert or Notification of Unusual Event .
Inspection Report# : 2004005*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Perform an Adequate Survey to Evaluate Radiological Hazards**
A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified when the licensee failed to perform an adequate survey to evaluate the radiological hazards associated with venting the steam generator exhaust into containment during the Unit 2 refueling outage. On February 7, 2003, the licensee failed to take air samples to account for the decay of tellurium-132 into iodine-132 in the steam generator exhaust prior to venting into the containment building. Consequently, fifty-two workers in containment received unplanned and unintended low-level intakes (less than 10 millirem) of iodine-132. This issue has been entered into the licensee's corrective action program as Action Request No. A0628334.

The failure to perform a survey to evaluate radiological hazards is a performance deficiency. The finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of Exposure Control and involved unplanned and unintended dose to workers that resulted from actions contrary to NRC requirements. Therefore the Occupational Radiation Safety Significance Determination Process was used to analyze the significance of the finding which was determined to be of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.
Inspection Report# : 2004009*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Lock a High Radiation Area with Dose Rates Greater than 1 Rem per Hour**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent unauthorized entry of a portion of the whole body into a high radiation area with dose rates greater than 1 rem per hour. Specifically, on November 14, 2004, PG&E failed to use an effective locking mechanism on the lower access flaps of the primary steam generator shield doors. The ineffective locking mechanism was discovered two days later when workers went to remove suction hoses. This could have allowed an individual to expose the arm above the elbow to dose rates greater than 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004

Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Access a High Radiation Area with Dose Rates Greater than 1 Rem per Hour with the Correct Radiation Work Permit**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent two individuals from entering a high radiation area with dose rates greater than 1 rem per hour on the incorrect radiation work permit. Two individuals entered an area with dose rates of 6 rem per hour in Reactor Coolant Pump Cubicle 2-4 using a radiation work permit which only allowed entry into areas with dose rates up to 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Aug 12, 2004
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Failure to Perform Radiological Survey of a High Radiation Area**
Green. A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified for the failure to perform required radiation surveys in Unit 2 to ensure compliance with 10 CFR 20.1902(b). Specifically, on January 28, 2003, during the performance of venting the volume control tank radiation protection personnel failed to perform adequate surveys of the Unit 2 Gas Decay Tank Room to post an expected high radiation area that would occur during this evolution. This finding involved cross-cutting aspects in the area of problem identification and resolution because the team noted that corrective actions for a similar event under the same circumstances had been ineffective in preventing recurrence. This issue was entered into the corrective action program under Action Request A0572997.

The finding is greater than minor because it was associated with one of the occupational radiation safety cornerstone attributes (exposure), and the finding affected the associated cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material. The inspector processed the issues through the Occupational Radiation Protection Significance Determination Process. This issues were determined to be a Green finding because it was not an ALARA planning and control issue, there was no personnel overexposure or substantial potential for personnel overexposure, and the licensee's ability to assess dose was not compromised.
Inspection Report# : 2004006*(pdf)*

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# Public Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to control radioactive material contained in certain generally-licensed devices in accordance with 10 CFR 31.5**
The team identified a non-cited violation of 10 CFR 31.5(c) because the licensee failed to maintain a program for generally-licensed radioactive devices used for reactor operations in accordance with the regulatory requirements. The licensee failed to implement a program for the use of generally-licensed devices used for monitoring personnel, and consequently failed to maintain and test 14 radioactive sources housed within the generally-licensed devices. Specifically, the licensee had not (1) conducted contamination leak tests on the device and the 10-millicurie Nickel-63 source housed in each device at the required frequency and (2) assigned an individual with the regulatory knowledge or authority to ensure compliance with 10 CFR 31.5. This issue has been entered into the licensee's corrective action program as Action Request A0628345.

The licensee's failure to control generally-licensed devices containing radioactive material in accordance with 10 CFR 31.5 was a performance deficiency. The finding was more than minor because it affected the Public Radiation Safety cornerstone attribute and affected the associated cornerstone objective. In order to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain, the licensee is required to leak test each generally-licensed device. Using the Public Radiation Safety Significance Determination Process, the finding had very low safety significance (Green) because: (1) it was not a transportation issue, (2) public exposure was not more than 5 millirem, and (3) there were not more than five occurrences. This finding also had crosscutting aspects associated with the effectiveness of problem identification and resolution.
Inspection Report# : 2004009*(pdf)*

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# Physical Protection

Physical Protection information not publicly available.

# Miscellaneous

Last modified : August 24, 2005

# Diablo Canyon 2
# 3Q/2005 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Properly Implement Procedure for Spent Fuel Pool Skimmer Filter Replacement**
A self-revealing NCV was identified for the failure to appropriately implement the procedure for spent fuel pool skimmer filter replacement, as required by Technical Specification 5.4.1.a. On December 23, 2004, operators cleared the spent fuel pool skimmer system using Section 6.3.1 of Procedure OP B-7:III, "Spent Fuel Pool System - Shutdown and Clearing and Filter Replacement," Revision 15, instead of the appropriate section, which was Section 6.3.2. A human performance cross cutting aspect was identified for the failure on two occasions to address configuration control concerns with the system.

This finding impacted the Initiating Events Cornerstone and was considered more than minor using Example 5.a of IMC 0612. Specifically, Valve SFS-2-3 was mis-positioned due to the use of the wrong section of Procedure OP B-7:III and then returned to service. Additionally, operators had two opportunities to identify the mis-positioning of Valve SFS-2-3 but failed to identify the condition. The mis-positioned valve resulted in a loss of approximately 36,000 gallons of water from the spent fuel pool. Using the SDP Phase 1 screening worksheet of IMC 0609, Appendix A, the finding was evaluated as a transient initiator, and it did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. Therefore, the finding was screened as having very low safety significance
Inspection Report# : 2004005*(pdf)*

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## Mitigating Systems

**Significance:** G Jul 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Assure That Appropriate Quality Standards Are Specified and Included in Design Documents and That Deviations are Controlled**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that appropriate quality standards are specified and included in the design documents and that deviations from such standards are controlled. Specifically, Pacific Gas and Electric Company failed to control the quality of work performed by contractors to ensure adequate cable bend radius for the newly installed vital battery chargers. Pacific Gas and Electric Company subsequently reworked to restore the proper bend radius. The quality control documents for cable terminations and installation have been modified to ensure that cable bend radius is assessed.

This finding impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is more than minor since it is similar to Inspection Manual Chapter 0612, Appendix E, Example 3.a, in that all vital battery chargers must have their connections and cables reworked for long term reliability. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspectors determined that 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. Therefore, the finding was determined to be of very low safety significance.
Inspection Report# : 2005004*(pdf)*

**Significance:** G Jun 17, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify Non-conservative Containment Recirculation Sump Valve Differential Pressure**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company initially screened industry operating experience regarding the potential for containment recirculation sump valves failing to open following certain small-break loss-of-coolant accidents as not being applicable to Diablo Canyon Power Plant. Upon questioning from the inspectors, the industry operating experience was found to be applicable and the calculation concerning containment recirculation sump valves were determined to be nonconforming but the valves remained operable. Additionally, the inspectors questioned Pacific Gas and Electric Company regarding the need for a prompt operability assessment for the valves. For corrective

actions, Pacific Gas and Electric Company planned to revise the calculation associated with the differential pressure across the containment recirculation sump valves and base future testing of the valves from the new calculation. This finding had cross-cutting aspects in the area problem identification and resolution.

The finding impacted the Mitigating Systems Cornerstone and was determined to be more than minor since it impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding affected the cornerstone attribute of design control, and the failure to recognize the applicability of the industry operating experience would allow the non-conservative design and testing of the containment recirculation sump valves to continue to exist. Using the Significance Determination Process Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance since the finding is a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18, Revision 1.
Inspection Report# : 2005004*(pdf)*

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to correct fire program violation concerning qualifications of Operations Responders in support of the fire brigade**
The inspectors identified a noncited violation of Technical Specification 5.4.1.d for failure to implement procedures for Fire Protection Implementation, because of failure to provide adequate training for operations fire responders. Procedure OM8, "Fire Protection Program," Revision 2B, Section 7.8 states, in part, that quality problems associated with the Fire Protection Program shall be documented and resolved in accordance with Procedure OM7 "Corrective Action," Revision 2B. Section 9.5.1 of the Final Safety Analysis Report states that measures are established to ensure conditions adverse to fire protection are identified, reported and corrected, and that administrative procedures are established to implement this requirement. Contrary to the above, Pacific Gas & Electric Company did not adequately implement and maintain a procedure for fire protection. Specifically, Pacific Gas & Electric Company failed to adequately resolve a condition adverse to fire protection in accordance with Procedure OM7. As of March 1, 2005, operations responders were not required to participate in fire drills for initial qualification or maintenance of qualification, as was noted as a qualification deficiency in Non-cited Violation 50-275;323/2003-08-01, and Action Request (AR) A0600934. This finding has problem identification and resolution cross cutting aspects for failure to correct operations responder training deficiencies.

The performance deficiency associated with this finding is a failure to adequately implement the fire protection program with respect to the qualifications of the fire brigade operations responder. The finding impacted the mitigating systems cornerstone and was more than minor since there was an adverse impact to a fire protection defense-in-depth element. Using the Significance Determination Process (SDP) Phase I Screening Worksheet and the SDP Phase II Notebook in Appendix F of Inspection Manual Chapter (IMC) 0609, the inspectors determined that the finding was of very low safety significance. Specifically, the significance of the finding was evaluated by considering fire scenarios in the vital 4 kV Bus F switchgear room and auxiliary saltwater Pump 1-1 vault. These two areas have the highest dependence on fire brigade response since they have the highest fire ignition frequency for areas that do not have automatic fire suppression. The inspectors evaluated the risk-significance using half the nominal credit for manual fire suppression as a result of the finding. Using Tables 5.4, 5.5, and 5.6 of IMC 0609, both fire scenarios screened as very low safety significance. Since the two fire scenarios were considered worst-case for the finding, the inspectors determined that the finding was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to promptly correct diesel engine generator lube oil instrument line crack**
The inspectors identified a noncited violation for the failure to promptly correct a cracked lube oil instrument sensing line, as required by 10 CFR Part 50, Appendix B, Criterion XVI. On August 29, 2004, operators observed a lube oil leak from the weld connecting the outlet of Valve DEG-2-1084 to instrument tubing. Approximately one month later, the leak had increased and it was discovered that the circumferential crack was 180 degrees through-wall on the weld. As a result, there was an increased potential for diesel engine generator (DEG) 2-3 to trip on low lube oil level. The finding had problem identification and resolution crosscutting aspects associated with operations and engineering personnel not recognizing the significance of the degraded condition and not implementing timely corrective actions.

This finding impacted the Mitigating Systems Cornerstone for reliability of systems that respond to initiating events to prevent undesirable consequences, and it affects the equipment performance attribute. The finding was more than minor using Example 4.f of Inspection Manual Chapter 0612, Appendix E. Similar to Example 4.f, the inspectors determined that there was impact to DEG 2-3 operability. Using the SDP Phase 1 screening worksheets in Appendix A of Inspection Manual Chapter 0609, the finding was determined to have potentially greater than very low safety significance because the failure could have resulted in an actual loss of diesel engine Generator 2-3 during a loss of offsite power event. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated conditional core damage frequency was 1.2E-7/yr. This violation was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding
**Diesel fuel oil transfer modification did not adequately assess reliability impact**
A finding was identified for modifying the diesel fuel oil transfer system without properly assessing the resulting net affect on reliability from introducing a new failure potential associated with new active components. As a result, the licensee rejected a small design change, which would have eliminated the failure mode when it was recognized that failure of the new pressure control valves could fail the train. Because the failure potential was not fully assessed, the licensee decided not to implement a change that would have eliminated the impact of the failure, nor were the pressure control valves subject to any preventive maintenance to ensure their reliability. This issue was entered into the licensee's corrective action program under Action Request A0630383.

The failure to properly assess the net effect on system reliability and risk due to the positive and negative effects of this modification, or to mitigate or eliminate a new failure mode created by the modification was a performance deficiency. This issue is more than minor because it affected the design control attribute of the Mitigating Systems cornerstone objective to assure the reliability and capability of equipment needed for accident mitigation. This finding was determined to be of very low safety significance (Green) during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18 based on test results.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**No procedure for cross-tying trains of the diesel fuel oil transfer system**
A noncited violation was identified for not having a procedure to cross-tie fuel oil transfer trains in response to certain failures, contrary to the design and licensing basis of the system. The design and license basis of the diesel fuel oil transfer system credited the capability to cross-tie trains in order to meet requirements to maintain the system function and be able to withstand a worst-case single failure. The team identified that the licensee did not have a procedure or training to accomplish this task. Failure to incorporate design and licensing requirements into plant procedures was a violation of 10 CFR Part 50, Appendix B, Criterion III. This issue was entered into the licensee's corrective action program under Action Requests A0630010 and A0630015.

The failure to have a procedure needed to meet the design and license basis of the fuel oil transfer system was a performance deficiency. This finding was more than minor because it impacted the Mitigating Systems cornerstone objective of procedure quality to ensure the capability of the system, in that, the system would not be capable of supplying the emergency diesel generators for the required 7-day mission time in the event of a single failure. The team concluded that this would not result in a loss of function in accordance with Generic Letter 91-18; since procedures direct monitoring of fuel capacity, operators would be aware of the need for action for the following reasons: 1) there should be a relatively long time available to detect and correct the problem (in excess of 24 hours), 2) the expected actions are not complex, and 3) existing procedures require monitoring of the remaining fuel oil capacity during extended diesel runs. Therefore, this finding was determined to be of very low safety significance (Green) in Phase 1 of the significance determination process. The licensee took prompt compensatory measures to ensure the full mission time could be met.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Diesel fuel oil storage tank calculation did not adequately account for vortexing**
A noncited violation was identified for inadequate design control because the licensee did not properly account for vortex prevention in the calculation used to determine the usable volume in the diesel fuel oil storage tank, which could cause the pump to ingest air. The licensee was unable to locate a technical basis for this part of the calculation. The team independently calculated that 4.1 inches was necessary, compared to the 2.0 inches used in the calculation. The licensee performed a similar calculation and reached the same conclusion, which reduced the tanks' unusable volumes by a little less than 1,000 gallons in this 50,000 gallon tank. Failure to properly account for the unusable fuel oil storage tank volume necessary to prevent vortexing was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0629779.

The failure to properly evaluate and document the unusable volume of the diesel fuel oil storage tank needed to prevent vortexing and ingesting air into the transfer pump was a performance deficiency. Through calculations, the licensee was able to demonstrate that there was sufficient available margin in both the tank capacity and the existing technical specification requirement to account for this without affecting operability or necessitating a technical specification change. This finding affected the Mitigating Systems cornerstone. The issue is more than minor because it was similar to Example 3.i of Appendix E to Manual Chapter 0609, since it was necessary to re-perform a calculation to determine whether the existing condition was acceptable. The finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process, since there was available margin in the tank capacity and technical specification minimum required volume and it was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to use the highest pressure in calculation to verify adequate auxiliary feedwater flow**
A noncited violation was identified for inadequate design control, because Calculation STA-135, "Auxiliary Feedwater System," Revision 2, which was intended to demonstrate that the auxiliary feedwater pumps have adequate capacity to meet their design basis, did not correctly identify the highest pressure under which the pumps needed to function. Specifically, the calculation did not account for the dynamic pressure loss between the feedwater inlet ring and the main steam safety valves. The licensee was able to perform an analysis that concluded the pumps had sufficient flow margin at the new pressure. Failure to properly translate the peak pressure against which the auxiliary feedwater pumps must deliver the required flow rate was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0630804.

The failure to document the capability of the auxiliary feedwater pumps to deliver the required flow at the maximum possible pressure was a performance deficiency. The issue is more than minor because a calculation was needed to determine whether the existing condition was acceptable, consistent with Example 3.i of Appendix E to Manual Chapter 0609. This issue affected the Mitigating Systems cornerstone. Because there was available margin in the pump capacity, this issue was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18. Therefore, the finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate power operated relief valve accumulator calculation**
A noncited violation was identified for inadequately translating design requirements into calculations used to demonstrate the capabilities of the pressurizer power operated relief valve backup accumulators. The calculation was found to contain a number of non-conservative errors and did not contain the most current acceptance criteria from accident analyses. As a result, this calculation failed to demonstrate that the backup nitrogen accumulators could operate the pressurizer power operated relief valves for the required number of cycles. Failure to properly demonstrate that design requirements for the number of power operated relief valve cycles needed to respond to an inadvertent safety injection actuation were satisfied through a design calculation was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Requests A0631420, A0630719 and A0630740.

The failure to adequately assess and document the capability of the backup accumulators to provide enough gas to operate the power operated relief valves through the required number of cycles was a performance deficiency. This issue was greater than minor because it was similar to Example 3.i in Manual Chapter 0612, Appendix E, in that, calculations had to be performed to demonstrate that the system could satisfy the accident analyses. This finding affected the Mitigating System cornerstone. This finding screened as having very low safety significance (Green) during a Phase 1 significance determination process, since the issue was confirmed to not have resulted in a loss of function in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Analyses did not demonstrate proper load sequencing with timer anomalies**
A noncited violation was identified for failure to demonstrate that load sequencing would satisfy regulatory requirements. The team identified that a single postulated fault occurring during load sequencing with offsite power available could restart load sequencing timers in all three engineered safety features buses and result in a more limiting scenario than previously analyzed by the licensee. This could result in overlaping starting transients for motors that were intended to start separately, which was not evaluated in existing calculations. The combined effects of this could cause later starting times for safety-related loads, potentially affecting system performance assumed in accident analyses. Failure to demonstrate that the system could perform as required considering a single fault was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request 0630036.

This failure to demonstrate through analyses that the electrical distribution system was capable of performing its required function following a single postulated fault was a performance deficiency. This issue was more than minor because it affected the Mitigating System cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a design basis accident. The licensee was subsequently able to demonstrate that there would be no loss of safety function even considering the effects of a fault as described above. Therefore, this finding was determined to be of very low safety significance (Green) in Phase I of the significance determination process.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding

**Incomplete action for setting auxiliary feedwater pump minimum flow values**
The team identified a Green finding for inadequate response to industry operating experience regarding establishing minimum flow for the auxiliary feedwater pumps. The team concluded that the licensee recognized that the conditions reported in NRC Bulletin 88-04 were present in auxiliary feedwater pumps because of low settings in the minimum flow lines, but failed to take appropriate actions to minimize and manage, or to eliminate, the potential for pump damage.

This finding represented a performance deficiency because the licensee did not adequately address a degradation mechanism identified in NRC Bulletin 88-04, as required by the station's operating experience program. As a result, the auxiliary feedwater pumps continued to be operated with insufficient minimum flow to avoid unusual wear and aging without establishing increased monitoring and maintenance, or other compensating actions.

This issue was more than minor because it affected the equipment reliability objective of the Mitigating Systems cornerstone. This issue screened as Green during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18. This issue will be treated as a finding in accordance with Manual Chapter 0612: FIN 05000275, 323/2005006-08, Inadequate Response to Operating Experience for Auxiliary Feedwater Minimum Flow.

Inspection Report# : 2005006*(pdf)*

**Significance:** G Jan 27, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Corrective Actions to Prevent Recurrence for Pressurizer Safety Valve Out-of-Tolerance Lift Setpoints**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI for failure to take corrective actions to prevent recurrence for a significant condition adverse to quality. On January 27, 2005, Pacific Gas and Electric Co. identified that the Unit 2 pressurizer safety valve lift setpoints were determined to be significantly out-of-tolerance, as compared to historical and industry-wide experience. However, Pacific Gas and Electric Co. failed to identify the root cause and propose any corrective actions to prevent recurrence, despite a history of pressurizer safety valve lift setpoints being out-of-tolerance.

The finding impacted the Mitigating Systems Cornerstone and was determined to be more than minor because it impacted the cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance since it did not represent an actual loss of safety function, represent an actual loss of a safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. Specifically, analysis demonstrated that the two valves having lift setpoints 4.4 and 3.6 percent low would not adversely affect the proper lift of the power-operated relief valves, and would not result in a spurious lift of the pressurizer safety valves during a normal transient.

Inspection Report# : 2005003*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Wire and Connect Test Equipment Resulted in Vital Bus De-Energization**
A self-revealing, noncited violation was identified for the failure to set up phase sequence test equipment according to procedure, as required by 10 CFR Part 50, Appendix B, Criterion V. This failure resulted in the momentary de-energization of Vital 4kV Bus G and the auto-start of Diesel Engine Generator 2-1. Subsequent investigation by Pacific Gas & Electric Company (PG&E) revealed that the primary side of the test transformer was wired in a wye configuration instead of a delta configuration. This wiring configuration introduced a direct short to ground, which caused the second level undervoltage relay to sense a degraded bus voltage for Vital 4kV Bus G. Subsequently, the relay removed the auxiliary power supply from Bus G and caused DEG 2-1 to start and load onto the bus. This finding involved a human performance cross-cutting aspect for the failure to wire the phase sequence test equipment properly for Vital 4kV Bus G and DEG 2-1.

The finding impacted the Mitigating Systems Cornerstone for ensuring the availability and capability of systems that respond to initiating events to prevent undesirable consequences that was associated with a pre-event human error performance. Considering Example 4.b of IMC 0612, Appendix E, the finding is greater than minor since the incorrect wiring and connection of the test equipment resulted in a vital bus de-energization and the actuation of DEG 2-1. Using Checklist 4 of Inspection Manual Chapter (IMC) 0609, Appendix G, Attachment 1, the finding did not result in the Technical Specifications for AC and DC power sources not being met and the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade PG&E's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade PG&E's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance .
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self-Revealing

Item Type: NCV NonCited Violation
**Inadequate ASCO Valve Qualification Causes Plant Trip**
A self revealing violation of 10 CFR 50.49(f) was identified for the failure to maintain approximately 70 safety related solenoid operated valves in an environmentally qualified condition. On February 9, 2002, an age related ASCO solenoid operated valve failure caused a loss of steam generator feedwater event and a Unit 2 manual plant trip. Further, the licensee did not promptly evaluate the extent of condition of the ASCO failure (coil insulation failure), which delayed the identification of elastomer qualification issues for approximately 1 year. In a related finding, the team identified that the licensee had missed earlier opportunities to identify ASCO elastomer qualification issues, in that they failed to thoroughly evaluate several pertinent NRC information notices and previous valve failures. The failure to: 1) properly establish equipment qualification limits; 2) thoroughly evaluate plant events and failures; and 3) properly evaluate industry operating experience constituted performance concerns. PG&E entered this issue into their corrective action program as Action Request 0613008. These issues have cross-cutting aspects in the area of problem identification and resolution because the original problem investigation did not identify the full scope of the cause and extent of condition, delaying some important corrective actions for approximately 1 year.

This finding was greater than minor because, if left uncorrected, these deficiencies would become a more significant safety concern by increasing the failure rate as the components age. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated delta-CDF for the finding is 2.2E-8/yr. This violation was of very low risk significance.
Inspection Report# : 2004005*(pdf)*

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# Barrier Integrity

**Significance:** G Sep 08, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Implement Adequate Work Control for Activities That Can Affect The Control Room Boundary**
A self-revealing noncited violation of Technical Specifications 5.4.1.a was identified for the failure to implement adequate work controls for painting activities in the area of control room ventilation equipment. Subsequently, the conduct of painting in the supply duct for Control Room Supply Fan S-38 resulted in operating fans drawing in the paint fumes into the control room. The work planning did not identify that the established ventilation path would result in the paint fumes entering the control room. The finding has crosscutting aspects associated with human performance in the planning of the work activity.

This finding impacted the Barrier Integrity Cornerstone and was determined to be more than minor because if left uncorrected the finding could result in a more significant safety concern involving control of work activities that could affect the control room atmosphere. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspector considered that the issue represented an administrative control function for preventing paint fumes from entering the control room and the protection of the control room ventilation system charcoal filters. This issue was discussed with a senior reactor analyst and determined that the appropriate safety significance evaluation was through management review. The management review considered Pacific Gas and Electric Company's control of painting materials in and around the control room envelope, any potential impact on the charcoal filters used to maintain the radiological barrier in the event of an accident, and any potential impact on licensee personnel. Based on the introduction of paint fumes into the control room did not adversely affect the control room operators' ability to operate the plant, there was not an actual degradation of the control room boundary and the charcoal filters remained operable, the finding was determined to be of very low safety significance.

Inspection Report# : 2005004*(pdf)*

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to properly pre-plan CRVS maintenance when opening the control room boundary**
Two examples of a self-revealing violation of Technical Specification 5.4.1.a were identified for failure to pre-plan maintenance associated with the Control Room Ventilation System (CRVS). On January 4 and February 1, 2005, both trains of CRVS were rendered inoperable for short periods of time when the system was opened to the atmosphere for maintenance without pre-planning the administrative controls prescribed by Technical Specification 3.7.10, because of personnel error. Technical Specification 3.7.10 states that the control room boundary may be opened intermittently under administrative controls, and that if two trains of CRVS are inoperable because of the control room boundary being open, then the system must be restored to operability within 24 hours. The Bases for Technical Specification 3.7.10 states that proper administrative controls to invoke this Technical Specification exception consist of stationing a dedicated individual who is in continuous communication with the control room, who has a method of rapidly closing the control room boundary, and has been specifically trained on these duties. These administrative controls were not in place when the control room boundary was inadvertently opened on January 4 and February 1, 2005. A human performance crosscutting aspect was identified for failure to pre-plan maintenance associated with the CRVS that resulted in the control room boundary being opened without administrative controls.

This issue is more than minor and affects the Barrier Integrity Cornerstone, because it represents partial losses of function of the CRVS on two occasions. On January 4 (for 15 minutes) and February 1, 2005, (four hours) both trains of CRVS were rendered inoperable because of an opening in the CRVS boundary which would have prohibited pressurization of the control room. Although Technical Specification 3.7.10 allowed this condition for up to 24 hours, it only allows opening of the control room boundary under strict administrative controls, that were not in place. This issue screens to Green in accordance with Item 1 of the Containment Barriers Cornerstone Phase 1 review, because it constitutes a CRVS issue only, and is therefore of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Mislabel of Neutron Flux Detector Resulted in Neutronic Decoupling of Detector From the Core**
The inspectors identified a noncited violation for the failure to develop a core offload sequence that maintained the source range neutron flux monitors operable, as required by 10 CFR Part 50, Appendix B, Criterion V. Inaccurate labeling of two neutron detectors in the core offload planning tool resulted in the development of a core offload sequence that when implemented resulted in one of the detectors becoming neutronically uncoupled from the core during core alterations. A human performance crosscutting aspect was identified for the labeling error in the core offload planning. A second human performance crosscutting aspect was identified for the failure to ascertain the cause of the downward trend when first identified by the inspectors.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute for procedure quality which could impact cladding. The finding is more than minor when compared to Example 4.e of Inspection Manual Chapter 0612, Appendix E. Similar to the example, Procedure OP B-8DS1, Step 5.2.1, described a responding nuclear instrument as having at least one fuel assembly face-adjacent or diagonally adjacent to the detector. Due to a labeling error in the core offload planning tool, the core offload sequence was developed in a manner that caused a neutron detector (Detector N-52) not to have an adjacent fuel assembly. Using Checklist 4 of Inspection Manual Chapter 0609, Appendix G, Attachment 1, the finding was determined not to increase the likelihood of a loss of reactor coolant system inventory, degrade Pacific Gas & Electric Company's ability to terminate a leak path or add reactor coolant system inventory when needed, or degrade Pacific Gas & Electric Company's ability to recover decay heat removal once it is lost. Therefore, the finding was screened as having very low safety significance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Containment Fan Cooler Unit Reverse Rotation**
The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly correct reverse rotation of containment fan cooler units (CFCUs) for both Units 1 and 2. PG&E observed reverse rotation of CFCUs for approximately 13 years, as a result of the CFCU backdraft dampers sticking partially open. The purpose of the backdraft dampers is to prevent reverse rotation of the CFCUs, which could cause the fan motor to trip on overcurrent when the CFCUs are started following a loss of coolant accident. Prior to Refueling Outage 2R12, 2 CFCUs in Unit 1 and 3 CFCUs in Unit 2 exhibited reverse rotation. One of the CFCUs in Unit 2 was considered inoperable due to reverse rotation and another was only considered operable if it was running.

The finding impacts the Barrier Integrity Cornerstone to provide reasonable assurance that physical design barriers protect the public from radio nuclide releases caused by accidents or events and is associated with the barrier performance attribute. The finding is more than minor when considering Example 3.g of IMC 0612, Appendix E. Similar to the example, PG&E observed reverse rotation of CFCUs for 13 years, and the reverse rotation impacted the operability of the CFCUs. Using the SDP Phase 1 Screening Worksheet from IMC 0609, the finding was determined to be of very low safety significance since it was determined that there was not an actual loss of defense-in-depth in containment pressure control or hydrogen control .
Inspection Report# : 2004005*(pdf)*

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# Emergency Preparedness

**Significance:** G Dec 31, 2004
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Establish Compensatory Measures to Ensure the Implementation of the Diablo Canyon Emergency Plan**
The inspectors identified a violation of 10 CFR 50.54(q) and 50.47.b(4) for the failure to maintain the seismic force monitors during the periods, June 16-19,1999, December 1-4, 2000, April 25-27, 2002, May 25-29, 2002, November 6-8, 2003, December 30-31, 2003, and August 9-10, 2004, such that the emergency plan designed to meet planning standard (4) in 10 CFR 50.47(b) could be promptly implemented. Specifically, PG&E failed to provide a means for the emergency director to promptly classify seismic events at the notification of unusual

event, alert or site area emergency levels, while the seismic force monitor utilized by the operators (emergency director) was out of service or being replaced. This finding had a human performance cross-cutting aspect associated with identifying compensatory measures to address the removal of the earthquake force monitors.

This performance deficiency impacted the emergency preparedness cornerstone because PG&E did not meet an emergency planning requirement and the cause was reasonably within PG&E 's control and should have been prevented. It is greater than minor because it has a potential to impact safety and because it was not a record keeping or administrative issue or an insignificant procedural error. This deficiency could have affected the EP Cornerstone objective of ensuring the capability to implement measures to protect the health and safety of the public during an emergency, and is associated with attributes of facilities and equipment, and offsite emergency preparedness. The finding is evaluated using the Emergency Preparedness "Failure to Comply" flowchart of the SDP and is a violation of 10 CFR 50.54(q) and planning standard 50.47(b)(4), which states, in part, that a standard emergency action level and classification system... is in use Utilizing the Failure to Comply Flow Chart in Manual Chapter 0609, the performance deficiency does not result in a failure of the risk significant planning standard (RSPS) or a degraded RSPS in that the unavailability of the seismic monitors would not prevent the declaration of a Site Area Emergency, Alert or Notification of Unusual Event .
Inspection Report# : 2004005*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Perform an Adequate Survey to Evaluate Radiological Hazards**
A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified when the licensee failed to perform an adequate survey to evaluate the radiological hazards associated with venting the steam generator exhaust into containment during the Unit 2 refueling outage. On February 7, 2003, the licensee failed to take air samples to account for the decay of tellurium-132 into iodine-132 in the steam generator exhaust prior to venting into the containment building. Consequently, fifty-two workers in containment received unplanned and unintended low-level intakes (less than 10 millirem) of iodine-132. This issue has been entered into the licensee's corrective action program as Action Request No. A0628334.

The failure to perform a survey to evaluate radiological hazards is a performance deficiency. The finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of Exposure Control and involved unplanned and unintended dose to workers that resulted from actions contrary to NRC requirements. Therefore the Occupational Radiation Safety Significance Determination Process was used to analyze the significance of the finding which was determined to be of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.
Inspection Report# : 2004009*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Lock a High Radiation Area with Dose Rates Greater than 1 Rem per Hour**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent unauthorized entry of a portion of the whole body into a high radiation area with dose rates greater than 1 rem per hour. Specifically, on November 14, 2004, PG&E failed to use an effective locking mechanism on the lower access flaps of the primary steam generator shield doors. The ineffective locking mechanism was discovered two days later when workers went to remove suction hoses. This could have allowed an individual to expose the arm above the elbow to dose rates greater than 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

**Significance:** G Dec 31, 2004
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Access a High Radiation Area with Dose Rates Greater than 1 Rem per Hour with the Correct Radiation Work Permit**
A self-revealing NCV of Technical Specification 5.7.2 was reviewed as a result of PG&E's failure to prevent two individuals from entering a high radiation area with dose rates greater than 1 rem per hour on the incorrect radiation work permit. Two individuals entered an area with dose rates of 6 rem per hour in Reactor Coolant Pump Cubicle 2-4 using a radiation work permit which only allowed entry into areas with dose

rates up to 1 rem per hour. This finding was placed into PG&E's corrective action program.

The finding is greater than minor because it is associated with one of the cornerstone attributes (exposure control) and affected the cornerstone objective because it could have resulted in unplanned, unintended radiation dose. The inspector determined that the finding was of very low significance because (1) it was not an ALARA finding, (2) it was not an overexposure, (3) it did have a substantial potential for overexposure, and (4) it did not compromise the ability to assess doses. This finding also had crosscutting aspects associated with human performance.
Inspection Report# : 2004005*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to control radioactive material contained in certain generally-licensed devices in accordance with 10 CFR 31.5**
The team identified a non-cited violation of 10 CFR 31.5(c) because the licensee failed to maintain a program for generally-licensed radioactive devices used for reactor operations in accordance with the regulatory requirements. The licensee failed to implement a program for the use of generally-licensed devices used for monitoring personnel, and consequently failed to maintain and test 14 radioactive sources housed within the generally-licensed devices. Specifically, the licensee had not (1) conducted contamination leak tests on the device and the 10-millicurie Nickel-63 source housed in each device at the required frequency and (2) assigned an individual with the regulatory knowledge or authority to ensure compliance with 10 CFR 31.5. This issue has been entered into the licensee's corrective action program as Action Request A0628345.

The licensee's failure to control generally-licensed devices containing radioactive material in accordance with 10 CFR 31.5 was a performance deficiency. The finding was more than minor because it affected the Public Radiation Safety cornerstone attribute and affected the associated cornerstone objective. In order to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain, the licensee is required to leak test each generally-licensed device. Using the Public Radiation Safety Significance Determination Process, the finding had very low safety significance (Green) because: (1) it was not a transportation issue, (2) public exposure was not more than 5 millirem, and (3) there were not more than five occurrences. This finding also had crosscutting aspects associated with the effectiveness of problem identification and resolution.
Inspection Report# : 2004009*(pdf)*

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

Last modified : November 30, 2005

# Diablo Canyon 2
# 4Q/2005 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Nov 29, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Emergency Core Cooling System Check Valve Back-Leakage**
An NRC-identified non-cited violation of 10 CFR Part 50, Criterion XVI, was identified for the failure to promptly correct Emergency Core Cooling System (ECCS) check valve back-leakage. Since 2000, Units 1 and 2 have experienced ECCS check valve back-leakage. Pacific Gas and Electric Company (PG&E) has failed to adequately take into consideration industry experience and provide for timely corrective actions regarding ECCS check valve back-leakage and its potential to cause gas-binding of ECCS pumps and/or water hammer of ECCS piping. This issue was entered into PG&E's corrective action program as Action Requests A0526037 and A0610421.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because it did not represent an actual loss of safety function, represent an actual loss of safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. The cause of the finding is related to the cross-cutting element of problem identification and resolution in that PG&E did not adequately evaluate and implement timely corrective actions to ECCS check valve back-leakage.

Inspection Report# : 2005005*(pdf)*

**Significance:** G Jul 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Assure That Appropriate Quality Standards Are Specified and Included in Design Documents and That Deviations are Controlled**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that appropriate quality standards are specified and included in the design documents and that deviations from such standards are controlled. Specifically, Pacific Gas and Electric Company failed to control the quality of work performed by contractors to ensure adequate cable bend radius for the newly installed vital battery chargers. Pacific Gas and Electric Company subsequently reworked to restore the proper bend radius. The quality control documents for cable terminations and installation have been modified to ensure that cable bend radius is assessed.

This finding impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is more than minor since it is similar to Inspection Manual Chapter 0612, Appendix E, Example 3.a, in that all vital battery chargers must have their connections and cables reworked for long term reliability. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspectors determined that 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. Therefore, the finding was determined to be of very low safety significance. The cause of the finding is related to the crosscutting element of human performance in that maintenance personnel failed to ensure the adequate cable bend radius for vital battery chargers.
Inspection Report# : 2005004*(pdf)*

**Significance:** G Jun 17, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify Non-conservative Containment Recirculation Sump Valve Differential Pressure**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company initially screened industry operating experience regarding the potential for

containment recirculation sump valves failing to open following certain small-break loss-of-coolant accidents as not being applicable to Diablo Canyon Power Plant. Upon questioning from the inspectors, the industry operating experience was found to be applicable and the calculation concerning containment recirculation sump valves were determined to be nonconforming but the valves remained operable. Additionally, the inspectors questioned Pacific Gas and Electric Company regarding the need for a prompt operability assessment for the valves. For corrective actions, Pacific Gas and Electric Company planned to revise the calculation associated with the differential pressure across the containment recirculation sump valves and base future testing of the valves from the new calculation.

The finding impacted the Mitigating Systems Cornerstone and was determined to be more than minor since it impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding affected the cornerstone attribute of design control, and the failure to recognize the applicability of the industry operating experience would allow the non-conservative design and testing of the containment recirculation sump valves to continue to exist. Using the Significance Determination Process Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance since the finding is a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18, Revision 1. This finding had cross-cutting aspects in the area of problem identification and resolution.
Inspection Report# : 2005004*(pdf)*

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to correct fire program violation concerning qualifications of Operations Responders in support of the fire brigade**
The inspectors identified a noncited violation of Technical Specification 5.4.1.d for failure to implement procedures for Fire Protection Implementation, because of failure to provide adequate training for operations fire responders. Procedure OM8, "Fire Protection Program," Revision 2B, Section 7.8 states, in part, that quality problems associated with the Fire Protection Program shall be documented and resolved in accordance with Procedure OM7 "Corrective Action," Revision 2B. Section 9.5.1 of the Final Safety Analysis Report states that measures are established to ensure conditions adverse to fire protection are identified, reported and corrected, and that administrative procedures are established to implement this requirement. Specifically, Pacific Gas & Electric Company failed to adequately resolve a condition adverse to fire protection in accordance with Procedure OM7. As of March 1, 2005, operations responders were not required to participate in fire drills for initial qualification or maintenance of qualification, as was noted as a qualification deficiency in Non-cited Violation 50-275;323/2003-08-01, and Action Request (AR) A0600934.

The performance deficiency associated with this finding is a failure to adequately implement the fire protection program with respect to the qualifications of the fire brigade operations responder. The finding impacted the mitigating systems cornerstone and was more than minor since there was an adverse impact to a fire protection defense-in-depth element. The finding is greater than minor because the reactor safety mitigating systems cornerstone objective attribute to provide protection against external factors was affected. Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," does not address fire brigade performance deficiencies. Regional management review concluded this finding was of very low safety significance because it affected the fire prevention and administrative controls category and represented a training deficiency. This finding has problem identification and resolution cross cutting aspects for the failure to correct operations responder training deficiencies.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Mar 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to promptly correct diesel engine generator lube oil instrument line crack**
The inspectors identified a noncited violation for the failure to promptly correct a cracked lube oil instrument sensing line, as required by 10 CFR Part 50, Appendix B, Criterion XVI. On August 29, 2004, operators observed a lube oil leak from the weld connecting the outlet of Valve DEG-2-1084 to instrument tubing. Approximately one month later, the leak had increased and it was discovered that the circumferential crack was 180 degrees through-wall on the weld. As a result, there was an increased potential for diesel engine generator (DEG) 2-3 to trip on low lube oil level. The finding had problem identification and resolution crosscutting aspects associated with operations and engineering personnel not recognizing the significance of the degraded condition and not implementing timely corrective actions.

This finding impacted the Mitigating Systems Cornerstone for reliability of systems that respond to initiating events to prevent undesirable consequences, and it affects the equipment performance attribute. The finding was more than minor using Example 4.f of Inspection Manual Chapter 0612, Appendix E. Similar to Example 4.f, the inspectors determined that there was impact to DEG 2-3 operability. Using the SDP Phase 1 screening worksheets in Appendix A of Inspection Manual Chapter 0609, the finding was determined to have potentially greater than very low safety significance because the failure could have resulted in an actual loss of diesel engine Generator 2-3 during a loss of offsite power event. An NRC Senior Reactor Analyst performed a Phase 3 significance determination and the estimated conditional core damage frequency was 1.2E-7/yr. This violation was of very low safety significance.

Inspection Report# : 2005002*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding
**Diesel fuel oil transfer modification did not adequately assess reliability impact**
A finding was identified for modifying the diesel fuel oil transfer system without properly assessing the resulting net affect on reliability from introducing a new failure potential associated with new active components. As a result, the licensee rejected a small design change, which would have eliminated the failure mode when it was recognized that failure of the new pressure control valves could fail the train. Because the failure potential was not fully assessed, the licensee decided not to implement a change that would have eliminated the impact of the failure, nor were the pressure control valves subject to any preventive maintenance to ensure their reliability. This issue was entered into the licensee's corrective action program under Action Request A0630383.

The failure to properly assess the net effect on system reliability and risk due to the positive and negative effects of this modification, or to mitigate or eliminate a new failure mode created by the modification was a performance deficiency. This issue is more than minor because it affected the design control attribute of the Mitigating Systems cornerstone objective to assure the reliability and capability of equipment needed for accident mitigation. This finding was determined to be of very low safety significance (Green) during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18 based on test results.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**No procedure for cross-tying trains of the diesel fuel oil transfer system**
A noncited violation was identified for not having a procedure to cross-tie fuel oil transfer trains in response to certain failures, contrary to the design and licensing basis of the system. The design and license basis of the diesel fuel oil transfer system credited the capability to cross-tie trains in order to meet requirements to maintain the system function and be able to withstand a worst-case single failure. The team identified that the licensee did not have a procedure or training to accomplish this task. Failure to incorporate design and licensing requirements into plant procedures was a violation of 10 CFR Part 50, Appendix B, Criterion III. This issue was entered into the licensee's corrective action program under Action Requests A0630010 and A0630015.

The failure to have a procedure needed to meet the design and license basis of the fuel oil transfer system was a performance deficiency. This finding was more than minor because it impacted the Mitigating Systems cornerstone objective of procedure quality to ensure the capability of the system, in that, the system would not be capable of supplying the emergency diesel generators for the required 7-day mission time in the event of a single failure. The team concluded that this would not result in a loss of function in accordance with Generic Letter 91-18; since procedures direct monitoring of fuel capacity, operators would be aware of the need for action for the following reasons: 1) there should be a relatively long time available to detect and correct the problem (in excess of 24 hours), 2) the expected actions are not complex, and 3) existing procedures require monitoring of the remaining fuel oil capacity during extended diesel runs. Therefore, this finding was determined to be of very low safety significance (Green) in Phase 1 of the significance determination process. The licensee took prompt compensatory measures to ensure the full mission time could be met.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Diesel fuel oil storage tank calculation did not adequately account for vortexing**
A noncited violation was identified for inadequate design control because the licensee did not properly account for vortex prevention in the calculation used to determine the usable volume in the diesel fuel oil storage tank, which could cause the pump to ingest air. The licensee was unable to locate a technical basis for this part of the calculation. The team independently calculated that 4.1 inches was necessary, compared to the 2.0 inches used in the calculation. The licensee performed a similar calculation and reached the same conclusion, which reduced the tanks' unusable volumes by a little less than 1,000 gallons in this 50,000 gallon tank. Failure to properly account for the unusable fuel oil storage tank volume necessary to prevent vortexing was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0629779.

The failure to properly evaluate and document the unusable volume of the diesel fuel oil storage tank needed to prevent vortexing and ingesting air into the transfer pump was a performance deficiency. Through calculations, the licensee was able to demonstrate that there was sufficient available margin in both the tank capacity and the existing technical specification requirement to account for this without affecting operability or necessitating a technical specification change. This finding affected the Mitigating Systems cornerstone. The issue is more than minor because it was similar to Example 3.i of Appendix E to Manual Chapter 0609, since it was necessary to re-perform a calculation to determine whether the existing condition was acceptable. The finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process, since there was available margin in the tank capacity and technical specification minimum required volume and it was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to use the highest pressure in calculation to verify adequate auxiliary feedwater flow**
A noncited violation was identified for inadequate design control, because Calculation STA-135, "Auxiliary Feedwater System," Revision 2, which was intended to demonstrate that the auxiliary feedwater pumps have adequate capacity to meet their design basis, did not correctly identify the highest pressure under which the pumps needed to function. Specifically, the calculation did not account for the dynamic pressure loss between the feedwater inlet ring and the main steam safety valves. The licensee was able to perform an analysis that concluded the pumps had sufficient flow margin at the new pressure. Failure to properly translate the peak pressure against which the auxiliary feedwater pumps must deliver the required flow rate was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request A0630804.

The failure to document the capability of the auxiliary feedwater pumps to deliver the required flow at the maximum possible pressure was a performance deficiency. The issue is more than minor because a calculation was needed to determine whether the existing condition was acceptable, consistent with Example 3.i of Appendix E to Manual Chapter 0609. This issue affected the Mitigating Systems cornerstone. Because there was available margin in the pump capacity, this issue was confirmed not to involve a loss of function of the system in accordance with Generic Letter 91-18. Therefore, the finding was determined to be of very low safety significance (Green) during Phase 1 of the significance determination process.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate power operated relief valve accumulator calculation**
A noncited violation was identified for inadequately translating design requirements into calculations used to demonstrate the capabilities of the pressurizer power operated relief valve backup accumulators. The calculation was found to contain a number of non-conservative errors and did not contain the most current acceptance criteria from accident analyses. As a result, this calculation failed to demonstrate that the backup nitrogen accumulators could operate the pressurizer power operated relief valves for the required number of cycles. Failure to properly demonstrate that design requirements for the number of power operated relief valve cycles needed to respond to an inadvertent safety injection actuation were satisfied through a design calculation was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Requests A0631420, A0630719 and A0630740.

The failure to adequately assess and document the capability of the backup accumulators to provide enough gas to operate the power operated relief valves through the required number of cycles was a performance deficiency. This issue was greater than minor because it was similar to Example 3.i in Manual Chapter 0612, Appendix E, in that, calculations had to be performed to demonstrate that the system could satisfy the accident analyses. This finding affected the Mitigating System cornerstone. This finding screened as having very low safety significance (Green) during a Phase 1 significance determination process, since the issue was confirmed to not have resulted in a loss of function in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Analyses did not demonstrate proper load sequencing with timer anomalies**
A noncited violation was identified for failure to demonstrate that load sequencing would satisfy regulatory requirements. The team identified that a single postulated fault occurring during load sequencing with offsite power available could restart load sequencing timers in all three engineered safety features buses and result in a more limiting scenario than previously analyzed by the licensee. This could result in overlaping starting transients for motors that were intended to start separately, which was not evaluated in existing calculations. The combined effects of this could cause later starting times for safety-related loads, potentially affecting system performance assumed in accident analyses. Failure to demonstrate that the system could perform as required considering a single fault was a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." This issue was entered into the licensee's corrective action program under Action Request 0630036.

This failure to demonstrate through analyses that the electrical distribution system was capable of performing its required function following a single postulated fault was a performance deficiency. This issue was more than minor because it affected the Mitigating System cornerstone objective of ensuring availability, reliability, and capability of systems needed to respond to a design basis accident. The licensee was subsequently able to demonstrate that there would be no loss of safety function even considering the effects of a fault as described above. Therefore, this finding was determined to be of very low safety significance (Green) in Phase I of the significance determination process.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Feb 15, 2005
Identified By: NRC
Item Type: FIN Finding

**Incomplete action for setting auxiliary feedwater pump minimum flow values**
The team identified a Green finding for inadequate response to industry operating experience regarding establishing minimum flow for the auxiliary feedwater pumps. The team concluded that the licensee recognized that the conditions reported in NRC Bulletin 88-04 were present in auxiliary feedwater pumps because of low settings in the minimum flow lines, but failed to take appropriate actions to minimize and manage, or to eliminate, the potential for pump damage.

This finding represented a performance deficiency because the licensee did not adequately address a degradation mechanism identified in NRC Bulletin 88-04, as required by the station's operating experience program. As a result, the auxiliary feedwater pumps continued to be operated with insufficient minimum flow to avoid unusual wear and aging without establishing increased monitoring and maintenance, or other compensating actions. This issue was more than minor because it affected the equipment reliability objective of the Mitigating Systems cornerstone. This issue screened as Green during a Phase 1 significance determination process, since the performance deficiency was confirmed not to result in a loss of function in accordance with Generic Letter 91-18.
Inspection Report# : 2005006*(pdf)*

**Significance:** G Jan 27, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Provide Corrective Actions to Prevent Recurrence for Pressurizer Safety Valve Out-of-Tolerance Lift Setpoints**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI for failure to take corrective actions to prevent recurrence for a significant condition adverse to quality. On January 27, 2005, Pacific Gas and Electric Co. identified that the Unit 2 pressurizer safety valve lift setpoints were determined to be significantly out-of-tolerance, as compared to historical and industry-wide experience. However, Pacific Gas and Electric Co. failed to identify the root cause and propose any corrective actions to prevent recurrence, despite a history of pressurizer safety valve lift setpoints being out-of-tolerance.

The finding impacted the Mitigating Systems Cornerstone and was determined to be more than minor because it impacted the cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance since it did not represent an actual loss of safety function, represent an actual loss of a safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. Specifically, analysis demonstrated that the two valves having lift setpoints 4.4 and 3.6 percent low would not adversely affect the proper lift of the power-operated relief valves, and would not result in a spurious lift of the pressurizer safety valves during a normal transient. The cause of the finding is related to the crosscutting element of problem identification and resolution in that Pacific Gas and Electric Company failed to identify the root cause and propose corrective actions for pressurizer safety valve lift setpoint out-of-tolerance.

Inspection Report# : 2005003*(pdf)*

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# Barrier Integrity

**Significance:** G Sep 08, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Implement Adequate Work Control for Activities That Can Affect The Control Room Boundary**
A self-revealing noncited violation of Technical Specifications 5.4.1.a was identified for the failure to implement adequate work controls for painting activities in the area of control room ventilation equipment. Subsequently, the conduct of painting in the supply duct for Control Room Supply Fan S-38 resulted in operating fans drawing in the paint fumes into the control room. The work planning did not identify that the established ventilation path would result in the paint fumes entering the control room. The finding has crosscutting aspects associated with human performance in the planning of the work activity.

This finding impacted the Barrier Integrity Cornerstone and was determined to be more than minor because if left uncorrected the finding could result in a more significant safety concern involving control of work activities that could affect the control room atmosphere. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspector considered that the issue represented an administrative control function for preventing paint fumes from entering the control room and the protection of the control room ventilation system charcoal filters. This issue was discussed with a senior reactor analyst and determined that the appropriate safety significance evaluation was through management review. The management review considered Pacific Gas and Electric Company's control of painting materials in and around the control room envelope, any potential impact on the charcoal filters used to maintain the radiological barrier in the event of an accident, and any potential impact on licensee personnel. Based on the introduction of paint fumes into the control room did not adversely affect the control room operators' ability to operate the plant, there was not an actual degradation of the control room boundary and the charcoal filters remained operable, the finding was determined to be of very low safety significance.

Inspection Report# : 2005004*(pdf)*

**Significance:** G Mar 31, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to properly pre-plan CRVS maintenance when opening the control room boundary**
Two examples of a self-revealing, non-cited violation of Technical Specification 5.4.1.a were identified for failure to adequately plan maintenance associated with the Control Room Ventilation System. On January 4 and February 1, 2005, both trains of the Control Room Ventilation System were inadvertently rendered inoperable for short periods of time when the system boundary was opened for maintenance. In each case, the maintenance activity was not appropriately planned to ensure the administrative controls prescribed by Technical Specification 3.7.10 were met and/or the appropriate components were identified.

This issue is more than minor because the issue affects the Barrier Integrity Cornerstone and represented a partial loss of function of the Control Room Ventilation System for both train boundaries being open. The issue was evaluated utilizing Inspection Manual Chapter 0609, "Significance Determination Process," Appendix A, Item 1 for the Containment Barriers Cornerstone. The Phase 1 review identified that the finding only represents a degradation of the radiological barrier function for the control room and was therefore of very low safety significance. A human performance crosscutting aspect was identified for the inadequate planning and communications involving the work activities on the Control Room Ventilation System.

Inspection Report# : 2005002*(pdf)*

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## Emergency Preparedness

**Significance: SL-IV** Oct 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Accurately Assess and Report Performance Indicator Data**
The inspector identified a noncited violation of 10 CFR Part 50.9 because Pacific Gas and Electric Company (PG&E) failed to provide complete and accurate information in a submittal of data for the emergency preparedness drill and exercise performance indicator. Specifically, PG&E staff failed to identify three missed opportunities for emergency notification accuracy during the second calendar quarter of 2005. PG&E took prompt action to correct the second quarter data, which resulted in the drill and exercise performance indicator color to cross from GREEN to WHITE. PG&E also initiated a 100 percent review of the second and third quarter drill and exercise performance indicator data and discovered one additional administrative error in the third quarter performance indicator data, which had been previously evaluated, but not yet reported to the NRC. PG&E had previously initiated a root cause evaluation in its corrective action program to determine the reason for the declining indicator and, subsequently, initiated another root cause evaluation to determine the reason for the failure to adequately evaluate and report the performance indicator data.

Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. Supplement 7, Section D.3, of the NRC Enforcement Policy describes this finding as a Severity Level IV violation. The issue is significant because it indicates a declining trend in the attention to detail shown by senior licensed operators in performing emergency notifications to the state and local authorities. This issue is documented in PG&E's corrective action program as Nonconformance Report N0002200. The finding had human performance cross-cutting aspects for the failure to provide accurate performance indicator data.
Inspection Report# : 2005005*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Post A Radiation Area**
The inspector identified a non-cited violation of 10 CFR 20.1902 because Pacific Gas and Electric Company (PG&E) failed to post a radiation area. Specifically, PG&E did not post an area within Vault 26 in which the radiation dose rates were approximately 30 millirem per hour at 30 centimeters from the surfaces of radioactive material storage containers. The finding was entered into PG&E's corrective action program as Action Request A0652226 and planned corrective action is still being evaluated.

The finding was more than minor because it was associated with one of the cornerstone attributes (exposure control and monitoring) and the finding affected the Occupational Radiation Safety cornerstone objective, in that uninformed workers could unknowingly accrue additional radiation dose. The inspector determined that the finding had no more than very low safety significance because: (1) it did not involve ALARA planning and controls, (2) there was no personnel overexposure, (3) there was no substantial potential for personnel overexposure, and (4) the

finding did not compromise PG&E's ability to assess dose. The finding also has cross-cutting aspects related to problem identification and resolution, in that a similar violation was previously identified during Inspection 50-275/02-04; 50-323/02-04.
Inspection Report# : 2005005*(pdf)*

**Significance:** G Jan 14, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Perform an Adequate Survey to Evaluate Radiological Hazards**
A self-revealing non-cited violation of 10 CFR 20.1501(a) was identified when the licensee failed to perform an adequate survey to evaluate the radiological hazards associated with venting the steam generator exhaust into containment during the Unit 2 refueling outage. On February 7, 2003, the licensee failed to take air samples to account for the decay of tellurium-132 into iodine-132 in the steam generator exhaust prior to venting into the containment building. Consequently, fifty-two workers in containment received unplanned and unintended low-level intakes (less than 10 millirem) of iodine-132. This issue has been entered into the licensee's corrective action program as Action Request No. A0628334.

The failure to perform a survey to evaluate radiological hazards is a performance deficiency. The finding is more than minor because it affected the Occupational Radiation Safety cornerstone objective to protect worker health and safety from radiation and radioactive materials. This finding was associated with the cornerstone attribute of Exposure Control and involved unplanned and unintended dose to workers that resulted from actions contrary to NRC requirements. Therefore the Occupational Radiation Safety Significance Determination Process was used to analyze the significance of the finding which was determined to be of very low safety significance because it did not involve: (1) ALARA planning and controls, (2) an overexposure, (3) a substantial potential for overexposure, or (4) an impaired ability to assess dose.
Inspection Report# : 2004009*(pdf)*

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## Public Radiation Safety

**Significance:** G Jan 14, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to control radioactive material contained in certain generally-licensed devices in accordance with 10 CFR 31.5**
The team identified a non-cited violation of 10 CFR 31.5(c) because the licensee failed to maintain a program for generally-licensed radioactive devices used for reactor operations in accordance with the regulatory requirements. The licensee failed to implement a program for the use of generally-licensed devices used for monitoring personnel, and consequently failed to maintain and test 14 radioactive sources housed within the generally-licensed devices. Specifically, the licensee had not (1) conducted contamination leak tests on the device and the 10-millicurie Nickel-63 source housed in each device at the required frequency and (2) assigned an individual with the regulatory knowledge or authority to ensure compliance with 10 CFR 31.5. This issue has been entered into the licensee's corrective action program as Action Request A0628345.

The licensee's failure to control generally-licensed devices containing radioactive material in accordance with 10 CFR 31.5 was a performance deficiency. The finding was more than minor because it affected the Public Radiation Safety cornerstone attribute and affected the associated cornerstone objective. In order to ensure adequate protection of the public health and safety from exposure to radioactive materials released into the public domain, the licensee is required to leak test each generally-licensed device. Using the Public Radiation Safety Significance Determination Process, the finding had very low safety significance (Green) because: (1) it was not a transportation issue, (2) public exposure was not more than 5 millirem, and (3) there were not more than five occurrences. This finding also had crosscutting aspects associated with the effectiveness of problem identification and resolution.
Inspection Report# : 2004009*(pdf)*

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

Last modified : March 03, 2006

# Diablo Canyon 2
# 1Q/2006 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Nov 29, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Emergency Core Cooling System Check Valve Back-Leakage**
An NRC-identified non-cited violation of 10 CFR Part 50, Criterion XVI, was identified for the failure to promptly correct Emergency Core Cooling System (ECCS) check valve back-leakage. Since 2000, Units 1 and 2 have experienced ECCS check valve back-leakage. Pacific Gas and Electric Company (PG&E) has failed to adequately take into consideration industry experience and provide for timely corrective actions regarding ECCS check valve back-leakage and its potential to cause gas-binding of ECCS pumps and/or water hammer of ECCS piping. This issue was entered into PG&E's corrective action program as Action Requests A0526037 and A0610421.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because it did not represent an actual loss of safety function, represent an actual loss of safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. The cause of the finding is related to the cross-cutting element of problem identification and resolution in that PG&E did not adequately evaluate and implement timely corrective actions to ECCS check valve back-leakage.

Inspection Report# : 2005005*(pdf)*

**Significance:** G Jul 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Assure That Appropriate Quality Standards Are Specified and Included in Design Documents and That Deviations are Controlled**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that appropriate quality standards are specified and included in the design documents and that deviations from such standards are controlled. Specifically, Pacific Gas and Electric Company failed to control the quality of work performed by contractors to ensure adequate cable bend radius for the newly installed vital battery chargers. Pacific Gas and Electric Company subsequently reworked to restore the proper bend radius. The quality control documents for cable terminations and installation have been modified to ensure that cable bend radius is assessed.

This finding impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is more than minor since it is similar to Inspection Manual Chapter 0612, Appendix E, Example 3.a, in that all vital battery chargers must have their connections and cables reworked for long term reliability. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspectors determined that 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. Therefore, the finding was determined to be of very low safety significance. The cause of the finding is related to the crosscutting element of human performance in that maintenance personnel failed to ensure the adequate cable bend radius for vital battery chargers.
Inspection Report# : 2005004*(pdf)*

**Significance:** G Jun 17, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify Non-conservative Containment Recirculation Sump Valve Differential Pressure**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company initially screened industry operating experience regarding the potential for

containment recirculation sump valves failing to open following certain small-break loss-of-coolant accidents as not being applicable to Diablo Canyon Power Plant. Upon questioning from the inspectors, the industry operating experience was found to be applicable and the calculation concerning containment recirculation sump valves were determined to be nonconforming but the valves remained operable. Additionally, the inspectors questioned Pacific Gas and Electric Company regarding the need for a prompt operability assessment for the valves. For corrective actions, Pacific Gas and Electric Company planned to revise the calculation associated with the differential pressure across the containment recirculation sump valves and base future testing of the valves from the new calculation.

The finding impacted the Mitigating Systems Cornerstone and was determined to be more than minor since it impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding affected the cornerstone attribute of design control, and the failure to recognize the applicability of the industry operating experience would allow the non-conservative design and testing of the containment recirculation sump valves to continue to exist. Using the Significance Determination Process Phase 1 Screening Worksheet of Inspection Manual Chapter 0609, the finding was determined to be of very low safety significance since the finding is a design or qualification deficiency confirmed not to result in loss of function per Generic Letter 91-18, Revision 1. This finding had cross-cutting aspects in the area of problem identification and resolution.
Inspection Report# : 2005004*(pdf)*

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# Barrier Integrity

**Significance:** G Sep 08, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Implement Adequate Work Control for Activities That Can Affect The Control Room Boundary**
A self-revealing noncited violation of Technical Specifications 5.4.1.a was identified for the failure to implement adequate work controls for painting activities in the area of control room ventilation equipment. Subsequently, the conduct of painting in the supply duct for Control Room Supply Fan S-38 resulted in operating fans drawing in the paint fumes into the control room. The work planning did not identify that the established ventilation path would result in the paint fumes entering the control room. The finding has crosscutting aspects associated with human performance in the planning of the work activity.

This finding impacted the Barrier Integrity Cornerstone and was determined to be more than minor because if left uncorrected the finding could result in a more significant safety concern involving control of work activities that could affect the control room atmosphere. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspector considered that the issue represented an administrative control function for preventing paint fumes from entering the control room and the protection of the control room ventilation system charcoal filters. This issue was discussed with a senior reactor analyst and determined that the appropriate safety significance evaluation was through management review. The management review considered Pacific Gas and Electric Company's control of painting materials in and around the control room envelope, any potential impact on the charcoal filters used to maintain the radiological barrier in the event of an accident, and any potential impact on licensee personnel. Based on the introduction of paint fumes into the control room did not adversely affect the control room operators' ability to operate the plant, there was not an actual degradation of the control room boundary and the charcoal filters remained operable, the finding was determined to be of very low safety significance.

Inspection Report# : 2005004*(pdf)*

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# Emergency Preparedness

**Significance: SL-IV** Oct 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Accurately Assess and Report Performance Indicator Data**
The inspector identified a noncited violation of 10 CFR Part 50.9 because Pacific Gas and Electric Company (PG&E) failed to provide complete and accurate information in a submittal of data for the emergency preparedness drill and exercise performance indicator. Specifically, PG&E staff failed to identify three missed opportunities for emergency notification accuracy during the second calendar quarter of 2005. PG&E took prompt action to correct the second quarter data, which resulted in the drill and exercise performance indicator color to cross from GREEN to WHITE. PG&E also initiated a 100 percent review of the second and third quarter drill and exercise performance indicator data and discovered one additional administrative error in the third quarter performance indicator data, which had been previously evaluated, but not yet reported to the NRC. PG&E had previously initiated a root cause evaluation in its corrective action program to determine the reason for the declining indicator and, subsequently, initiated another root cause evaluation to determine the reason for the failure to adequately evaluate and report the performance indicator data.

Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. Supplement 7, Section D.3, of the NRC Enforcement Policy describes this finding as a Severity Level IV violation. The issue is significant

because it indicates a declining trend in the attention to detail shown by senior licensed operators in performing emergency notifications to the state and local authorities. This issue is documented in PG&E's corrective action program as Nonconformance Report N0002200. The finding had human performance cross-cutting aspects for the failure to provide accurate performance indicator data.
Inspection Report# : 2005005*(pdf)*

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## Occupational Radiation Safety

**Significance:** G Dec 31, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Post A Radiation Area**
The inspector identified a non-cited violation of 10 CFR 20.1902 because Pacific Gas and Electric Company (PG&E) failed to post a radiation area. Specifically, PG&E did not post an area within Vault 26 in which the radiation dose rates were approximately 30 millirem per hour at 30 centimeters from the surfaces of radioactive material storage containers. The finding was entered into PG&E's corrective action program as Action Request A0652226 and planned corrective action is still being evaluated.

The finding was more than minor because it was associated with one of the cornerstone attributes (exposure control and monitoring) and the finding affected the Occupational Radiation Safety cornerstone objective, in that uninformed workers could unknowingly accrue additional radiation dose. The inspector determined that the finding had no more than very low safety significance because: (1) it did not involve ALARA planning and controls, (2) there was no personnel overexposure, (3) there was no substantial potential for personnel overexposure, and (4) the finding did not compromise PG&E's ability to assess dose. The finding also has cross-cutting aspects related to problem identification and resolution, in that a similar violation was previously identified during Inspection 50-275/02-04; 50-323/02-04.
Inspection Report# : 2005005*(pdf)*

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## Public Radiation Safety

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

Last modified : May 25, 2006

# Diablo Canyon 2
# 2Q/2006 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Apr 20, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate refueling procedure for draining and depressurizing the reactor coolant system**
An NRC-identified, non-cited violation of Technical Specification 5.4.1.a was determined for an inadequate procedure, Procedure OP A-2:II, “Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel,” Revision 33A. Specifically, on April 20, 2006, while operators depressurized the reactor coolant system (RCS), with water level 2 ft below the reactor vessel flange, the two required level instruments, wide-range reactor vessel refueling level indication system and LI-400, read 15 inches higher than actual reactor vessel water level. The inspectors determined that the procedure was not adequate because prior operating experience had not been incorporated into the procedure that demonstrated the level instruments would read non-conservatively during RCS depressurization. Also, Procedure OP A-2:II did not have criteria that alerted operators to abnormal level instrument deviations that may be caused by phenomenon outside of the level deviations expected by the RCS depressurization. Pacific Gas and Electric Company (PG&E) has planned to evaluate potential changes to Procedure OP A-2:II and RCS water level instrumentation when used during RCS depressurization. This issue was entered into PG&E's corrective action program as Action Requests A0664484, A0672419, and A0672422.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, Attachment 1, Checklist 3, the finding is determined to be of very low safety significance since an optional set of instrumentation provided accurate RCS level indication and there was no loss of RCS inventory control. The finding had a cross-cutting aspect in the area of human performance for resources because PG&E failed to ensure the adequacy of procedures used for reactor vessel level monitoring to ensure nuclear safety.
Inspection Report# : 2006003*(pdf)*

**Significance:** G Nov 29, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Emergency Core Cooling System Check Valve Back-Leakage**
An NRC-identified non-cited violation of 10 CFR Part 50, Criterion XVI, was identified for the failure to promptly correct Emergency Core Cooling System (ECCS) check valve back-leakage. Since 2000, Units 1 and 2 have experienced ECCS check valve back-leakage. Pacific Gas and Electric Company (PG&E) has failed to adequately take into consideration industry experience and provide for timely corrective actions regarding ECCS check valve back-leakage and its potential to cause gas-binding of ECCS pumps and/or water hammer of ECCS piping. This issue was entered into PG&E's corrective action program as Action Requests A0526037 and A0610421.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because it did not represent an actual loss of safety function, represent an actual loss of safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. The cause of the finding is related to the cross-cutting element of problem identification and resolution in that PG&E did not adequately evaluate and implement timely corrective actions to ECCS check valve back-leakage.

Inspection Report# : 2005005*(pdf)*

**Significance:** G Jul 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Assure That Appropriate Quality Standards Are Specified and Included in Design Documents and That Deviations are Controlled**
The inspectors identified an noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to assure that

appropriate quality standards are specified and included in the design documents and that deviations from such standards are controlled. Specifically, Pacific Gas and Electric Company failed to control the quality of work performed by contractors to ensure adequate cable bend radius for the newly installed vital battery chargers. Pacific Gas and Electric Company subsequently reworked to restore the proper bend radius. The quality control documents for cable terminations and installation have been modified to ensure that cable bend radius is assessed.

This finding impacted the Mitigating Systems Cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is more than minor since it is similar to Inspection Manual Chapter 0612, Appendix E, Example 3.a, in that all vital battery chargers must have their connections and cables reworked for long term reliability. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspectors determined that 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. Therefore, the finding was determined to be of very low safety significance. The cause of the finding is related to the crosscutting element of human performance in that maintenance personnel failed to ensure the adequate cable bend radius for vital battery chargers.
Inspection Report# : 2005004*(pdf)*

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# Barrier Integrity

**Significance:** G May 03, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to follow welding procedures**
An NRC-identified, non-cited violation of Technical Specification 5.4.1 was identified because Pacific Gas and Electric Company (PG&E) failed to follow the procedure for ensuring that welding preheat temperatures were verified prior to welding. Specifically, during the replacement of Component Cooling Water Valves 279 and 280, which provide cooling to the reactor vessel support pads, PG&E failed to verify that the minimum welding preheat temperature of 50°F was met and could not demonstrate that the ambient temperature was greater than 50°F. PG&E entered the finding into their corrective action program as Action Request A0665588.

The finding was greater than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone and impacted the cornerstone objective of providing reasonable assurance that physical design barriers, in this case the reactor coolant system, protect the public from radio-nuclide releases caused by accidents or events. The finding was determined to be of very low safety significance based on management review of the plant conditions at the time the performance deficiency occurred (defueled) and the condition was evaluated prior to the plant entering Mode 5.
Inspection Report# : 2006003*(pdf)*

**Significance:** G Sep 08, 2005
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Implement Adequate Work Control for Activities That Can Affect The Control Room Boundary**
A self-revealing noncited violation of Technical Specifications 5.4.1.a was identified for the failure to implement adequate work controls for painting activities in the area of control room ventilation equipment. Subsequently, the conduct of painting in the supply duct for Control Room Supply Fan S-38 resulted in operating fans drawing in the paint fumes into the control room. The work planning did not identify that the established ventilation path would result in the paint fumes entering the control room. The finding has crosscutting aspects associated with human performance in the planning of the work activity.

This finding impacted the Barrier Integrity Cornerstone and was determined to be more than minor because if left uncorrected the finding could result in a more significant safety concern involving control of work activities that could affect the control room atmosphere. Using the Significance Determination Process Phase 1 Screening Worksheet in Appendix A of Inspection Manual Chapter 0609, the inspector considered that the issue represented an administrative control function for preventing paint fumes from entering the control room and the protection of the control room ventilation system charcoal filters. This issue was discussed with a senior reactor analyst and determined that the appropriate safety significance evaluation was through management review. The management review considered Pacific Gas and Electric Company's control of painting materials in and around the control room envelope, any potential impact on the charcoal filters used to maintain the radiological barrier in the event of an accident, and any potential impact on licensee personnel. Based on the introduction of paint fumes into the control room did not adversely affect the control room operators' ability to operate the plant, there was not an actual degradation of the control room boundary and the charcoal filters remained operable, the finding was determined to be of very low safety significance.

Inspection Report# : 2005004*(pdf)*

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# Emergency Preparedness

**Significance: SL-IV** Oct 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Accurately Assess and Report Performance Indicator Data**
The inspector identified a noncited violation of 10 CFR Part 50.9 because Pacific Gas and Electric Company (PG&E) failed to provide complete and accurate information in a submittal of data for the emergency preparedness drill and exercise performance indicator. Specifically, PG&E staff failed to identify three missed opportunities for emergency notification accuracy during the second calendar quarter of 2005. PG&E took prompt action to correct the second quarter data, which resulted in the drill and exercise performance indicator color to cross from GREEN to WHITE. PG&E also initiated a 100 percent review of the second and third quarter drill and exercise performance indicator data and discovered one additional administrative error in the third quarter performance indicator data, which had been previously evaluated, but not yet reported to the NRC. PG&E had previously initiated a root cause evaluation in its corrective action program to determine the reason for the declining indicator and, subsequently, initiated another root cause evaluation to determine the reason for the failure to adequately evaluate and report the performance indicator data.

Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. Supplement 7, Section D.3, of the NRC Enforcement Policy describes this finding as a Severity Level IV violation. The issue is significant because it indicates a declining trend in the attention to detail shown by senior licensed operators in performing emergency notifications to the state and local authorities. This issue is documented in PG&E's corrective action program as Nonconformance Report N0002200. The finding had human performance cross-cutting aspects for the failure to provide accurate performance indicator data.
Inspection Report# : 2005005*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Apr 18, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey to identify the magnitude and extent of radiation levels to identify radiological hazards**
The inspectors identified a non-cited violation of 10 CFR 20.1501(a) because Pacific Gas and Electric Company (PG&E) failed to survey to determine the extent and magnitude of radiation levels and evaluate the radiological hazards. Specifically, on April 18, 2006, the inspectors identified elevated radiation levels near two chemical volume control system valves located in a hallway on the 100-foot elevation of Unit 2. PG&E confirmed elevated radiation levels near the valves were as high as 200 millirem per hour on contact and 28 millirem per hour at 30 centimeters. PG&E surveyed the area and entered the finding into their corrective action program as Action Request A0665039.

The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Exposure Control and Monitoring and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because workers could have unknowingly received additional radiation exposure. When going through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not an as low as is reasonably achievable finding. There was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The finding also had cross-cutting aspects associated with human performance because because adequate resources were not established for the survey requirements.
Inspection Report# : 2006003*(pdf)*

**Significance:** G Nov 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Post A Radiation Area**
The inspector identified a non-cited violation of 10 CFR 20.1902 because Pacific Gas and Electric Company (PG&E) failed to post a radiation area. Specifically, PG&E did not post an area within Vault 26 in which the radiation dose rates were approximately 30 millirem per hour at 30 centimeters from the surfaces of radioactive material storage containers. The finding was entered into PG&E's corrective action program as Action Request A0652226 and planned corrective action is still being evaluated.

The finding was more than minor because it was associated with one of the cornerstone attributes (exposure control and monitoring) and the finding affected the Occupational Radiation Safety cornerstone objective, in that uninformed workers could unknowingly accrue additional radiation dose. The inspector determined that the finding had no more than very low safety significance because: (1) it did not involve ALARA planning and controls, (2) there was no personnel overexposure, (3) there was no substantial potential for personnel overexposure, and (4) the finding did not compromise PG&E's ability to assess dose. The finding also has cross-cutting aspects related to problem identification and resolution, in that a similar violation was previously identified during Inspection 50-275/02-04; 50-323/02-04.
Inspection Report# : 2005005*(pdf)*

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## Public Radiation Safety

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

Last modified : August 25, 2006

# Diablo Canyon 2
# 3Q/2006 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 25, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Include Floor Drains Credited in the Flood Analysis Into the Maintenance Rule Program**
An NRC-identified, noncited violation of 10 CFR 50.65(b) was determined for the failure of engineering staff to include the auxiliary feedwater pump room floor drains within the scope of Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance at the Diablo Canyon Power Plant. Specifically, Calculation 76060, "Flooding Analysis G Area and Auxiliary Building," Revision 1, assumes that at least two of the three floor drains in the auxiliary feedwater pump rooms would be able to remove up to 316 gpm of water in the event of a flood. Despite their credited function in the flood analysis, engineering staff did not scope them into their monitoring program. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0678658.

The finding is greater than minor because it is associated with the Mitigating Systems cornerstone attribute of protection against external factors and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with operating experience because engineering personnel did not effectively incorporate pertinent industry operating experience into their program for evaluating the effectiveness of maintenance performed on AFW pump room floor drains.
Inspection Report# : 2006004*(pdf)*

**Significance:** G Aug 23, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Identify that the Correct Equipment Necessary for Implementing EOP for Inadequate Core Cooling Was Not Pre-staged**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company failed to promptly identify that it had prestaged the wrong equipment (a flange hose connection with the wrong tread pattern) necessary to cross-connect the fire main water system to the auxiliary feedwater system during a loss of core cooling event. This performance deficiency was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0676729.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function

of a single train for greater than its TS allowed outage time, did not represent an actual loss of one or more risk-significant non-TS trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a crosscutting aspect in the area of human performance associated with resources because the licensee did not ensure that equipment needed to perform an EOP was available and adequate to assure nuclear safety.
Inspection Report# : 2006004*(pdf)*

**Significance:** G Apr 20, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate refueling procedure for draining and depressurizing the reactor coolant system**
An NRC-identified, non-cited violation of Technical Specification 5.4.1.a was determined for an inadequate procedure, Procedure OP A-2:II, “Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel,” Revision 33A. Specifically, on April 20, 2006, while operators depressurized the reactor coolant system (RCS), with water level 2 ft below the reactor vessel flange, the two required level instruments, wide-range reactor vessel refueling level indication system and LI-400, read 15 inches higher than actual reactor vessel water level. The inspectors determined that the procedure was not adequate because prior operating experience had not been incorporated into the procedure that demonstrated the level instruments would read non-conservatively during RCS depressurization. Also, Procedure OP A-2:II did not have criteria that alerted operators to abnormal level instrument deviations that may be caused by phenomenon outside of the level deviations expected by the RCS depressurization. Pacific Gas and Electric Company (PG&E) has planned to evaluate potential changes to Procedure OP A-2:II and RCS water level instrumentation when used during RCS depressurization. This issue was entered into PG&E's corrective action program as Action Requests A0664484, A0672419, and A0672422.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, Attachment 1, Checklist 3, the finding is determined to be of very low safety significance since an optional set of instrumentation provided accurate RCS level indication and there was no loss of RCS inventory control. The finding had a cross-cutting aspect in the area of human performance for resources because PG&E failed to ensure the adequacy of procedures used for reactor vessel level monitoring to ensure nuclear safety.
Inspection Report# : 2006003*(pdf)*

**Significance:** G Nov 29, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Correct Emergency Core Cooling System Check Valve Back-Leakage**
An NRC-identified non-cited violation of 10 CFR Part 50, Criterion XVI, was identified for the failure to promptly correct Emergency Core Cooling System (ECCS) check valve back-leakage. Since 2000, Units 1 and 2 have experienced ECCS check valve back-leakage. Pacific Gas and Electric Company (PG&E) has failed to adequately take into consideration industry experience and provide for timely corrective actions regarding ECCS check valve back-leakage and its potential to cause gas-binding of ECCS pumps and/or water hammer of ECCS piping. This issue was entered into PG&E's corrective action program as Action Requests A0526037 and A0610421.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because it did not represent an actual loss of safety function, represent an actual loss of safety function for a single train for greater than the Technical Specification allowed outage time, or screen as potentially risk significant due to seismic, fire, flooding, or severe weather initiating events. The cause of the finding is related to the cross-cutting element of problem identification and resolution in that PG&E did not adequately evaluate and implement timely corrective actions to ECCS check valve back-leakage.

Inspection Report# : 2005005*(pdf)*

# Barrier Integrity

**Significance:** G May 03, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to follow welding procedures**
An NRC-identified, non-cited violation of Technical Specification 5.4.1 was identified because Pacific Gas and Electric Company (PG&E) failed to follow the procedure for ensuring that welding preheat temperatures were verified prior to welding. Specifically, during the replacement of Component Cooling Water Valves 279 and 280, which provide cooling to the reactor vessel support pads, PG&E failed to verify that the minimum welding preheat temperature of 50°F was met and could not demonstrate that the ambient temperature was greater than 50°F. PG&E entered the finding into their corrective action program as Action Request A0665588.

The finding was greater than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone and impacted the cornerstone objective of providing reasonable assurance that physical design barriers, in this case the reactor coolant system, protect the public from radio-nuclide releases caused by accidents or events. The finding was determined to be of very low safety significance based on management review of the plant conditions at the time the performance deficiency occurred (defueled) and the condition was evaluated prior to the plant entering Mode 5.
Inspection Report# : 2006003*(pdf)*

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# Emergency Preparedness

**Significance: SL-IV** Oct 20, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Accurately Assess and Report Performance Indicator Data**
The inspector identified a noncited violation of 10 CFR Part 50.9 because Pacific Gas and Electric Company (PG&E) failed to provide complete and accurate information in a submittal of data for the emergency preparedness drill and exercise performance indicator. Specifically, PG&E staff failed to identify three missed opportunities for emergency notification accuracy during the second calendar quarter of 2005. PG&E took prompt action to correct the second quarter data, which resulted in the drill and exercise performance indicator color to cross from GREEN to WHITE. PG&E also initiated a 100 percent review of the second and third quarter drill and exercise performance indicator data and discovered one additional administrative error in the third quarter performance indicator data, which had been previously evaluated, but not yet reported to the NRC. PG&E had previously initiated a root cause evaluation in its corrective action program to determine the reason for the declining indicator and, subsequently, initiated another root cause evaluation to determine the reason for the failure to adequately evaluate and report the performance indicator data.

Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated using the traditional enforcement process. Supplement 7, Section D.3, of the NRC Enforcement Policy describes this finding as a Severity Level IV violation. The issue is significant because it indicates a declining trend in the attention to detail shown by senior licensed operators in performing emergency notifications to the state and local authorities. This issue is documented in PG&E's corrective action program as Nonconformance Report N0002200. The finding had human performance cross-cutting aspects for the failure to provide accurate performance indicator data.
Inspection Report# : 2005005*(pdf)*

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# Occupational Radiation Safety

**Significance:** G Apr 18, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey to identify the magnitude and extent of radiation levels to identify radiological hazards**
The inspectors identified a non-cited violation of 10 CFR 20.1501(a) because Pacific Gas and Electric Company (PG&E) failed to survey to determine the extent and magnitude of radiation levels and evaluate the radiological hazards. Specifically, on April 18, 2006, the inspectors identified elevated radiation levels near two chemical volume control system valves located in a hallway on the 100-foot elevation of Unit 2. PG&E confirmed elevated radiation levels near the valves were as high as 200 millirem per hour on contact and 28 millirem per hour at 30 centimeters. PG&E surveyed the area and entered the finding into their corrective action program as Action Request A0665039.

The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Exposure Control and Monitoring and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because workers could have unknowingly received additional radiation exposure. When going through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not an as low as is reasonably achievable finding. There was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The finding also had cross-cutting aspects associated with human performance because because adequate resources were not established for the survey requirements.
Inspection Report# : 2006003*(pdf)*

**Significance:** G Nov 15, 2005
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Post A Radiation Area**
The inspector identified a non-cited violation of 10 CFR 20.1902 because Pacific Gas and Electric Company (PG&E) failed to post a radiation area. Specifically, PG&E did not post an area within Vault 26 in which the radiation dose rates were approximately 30 millirem per hour at 30 centimeters from the surfaces of radioactive material storage containers. The finding was entered into PG&E's corrective action program as Action Request A0652226 and planned corrective action is still being evaluated.

The finding was more than minor because it was associated with one of the cornerstone attributes (exposure control and monitoring) and the finding affected the Occupational Radiation Safety cornerstone objective, in that uninformed workers could unknowingly accrue additional radiation dose. The inspector determined that the finding had no more than very low safety significance because: (1) it did not involve ALARA planning and controls, (2) there was no personnel overexposure, (3) there was no substantial potential for personnel overexposure, and (4) the finding did not compromise PG&E's ability to assess dose. The finding also has cross-cutting aspects related to problem identification and resolution, in that a similar violation was previously identified during Inspection 50-275/02-04; 50-323/02-04.
Inspection Report# : 2005005*(pdf)*

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# Public Radiation Safety

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# Physical Protection

Physical Protection information not publicly available.

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# Miscellaneous

Last modified : December 21, 2006

# Diablo Canyon 2
# 4Q/2006 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 25, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Include Floor Drains Credited in the Flood Analysis Into the Maintenance Rule Program**
An NRC-identified, noncited violation of 10 CFR 50.65(b) was determined for the failure of engineering staff to include the auxiliary feedwater pump room floor drains within the scope of Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance at the Diablo Canyon Power Plant. Specifically, Calculation 76060, "Flooding Analysis G Area and Auxiliary Building," Revision 1, assumes that at least two of the three floor drains in the auxiliary feedwater pump rooms would be able to remove up to 316 gpm of water in the event of a flood. Despite their credited function in the flood analysis, engineering staff did not scope them into their monitoring program. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0678658.

The finding is greater than minor because it is associated with the Mitigating Systems cornerstone attribute of protection against external factors and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with operating experience because engineering personnel did not effectively incorporate pertinent industry operating experience into their program for evaluating the effectiveness of maintenance performed on AFW pump room floor drains.
Inspection Report# : 2006004 *(pdf)*

**Significance:** G Aug 23, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Identify that the Correct Equipment Necessary for Implementing EOP for Inadequate Core Cooling Was Not Pre-staged**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company failed to promptly identify that it had prestaged the wrong equipment (a flange hose connection with the wrong tread pattern) necessary to cross-connect the fire main water system to the auxiliary feedwater system during a loss of core cooling event. This performance deficiency was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0676729.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function

of a single train for greater than its TS allowed outage time, did not represent an actual loss of one or more risk-significant non-TS trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a crosscutting aspect in the area of human performance associated with resources because the licensee did not ensure that equipment needed to perform an EOP was available and adequate to assure nuclear safety.
Inspection Report# : 2006004 *(pdf)*

**Significance:** G Apr 20, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate refueling procedure for draining and depressurizing the reactor coolant system**
An NRC-identified, non-cited violation of Technical Specification 5.4.1.a was determined for an inadequate procedure, Procedure OP A-2:II, “Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel,” Revision 33A. Specifically, on April 20, 2006, while operators depressurized the reactor coolant system (RCS), with water level 2 ft below the reactor vessel flange, the two required level instruments, wide-range reactor vessel refueling level indication system and LI-400, read 15 inches higher than actual reactor vessel water level. The inspectors determined that the procedure was not adequate because prior operating experience had not been incorporated into the procedure that demonstrated the level instruments would read non-conservatively during RCS depressurization. Also, Procedure OP A-2:II did not have criteria that alerted operators to abnormal level instrument deviations that may be caused by phenomenon outside of the level deviations expected by the RCS depressurization. Pacific Gas and Electric Company (PG&E) has planned to evaluate potential changes to Procedure OP A-2:II and RCS water level instrumentation when used during RCS depressurization. This issue was entered into PG&E's corrective action program as Action Requests A0664484, A0672419, and A0672422.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, Attachment 1, Checklist 3, the finding is determined to be of very low safety significance since an optional set of instrumentation provided accurate RCS level indication and there was no loss of RCS inventory control. The finding had a cross-cutting aspect in the area of human performance for resources because PG&E failed to ensure the adequacy of procedures used for reactor vessel level monitoring to ensure nuclear safety.
Inspection Report# : 2006003 *(pdf)*

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## Barrier Integrity

**Significance:** G May 03, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to follow welding procedures**
An NRC-identified, non-cited violation of Technical Specification 5.4.1 was identified because Pacific Gas and Electric Company (PG&E) failed to follow the procedure for ensuring that welding preheat temperatures were verified prior to welding. Specifically, during the replacement of Component Cooling Water Valves 279 and 280, which provide cooling to the reactor vessel support pads, PG&E failed to verify that the minimum welding preheat temperature of 50°F was met and could not demonstrate that the ambient temperature was greater than 50°F. PG&E entered the finding into their corrective action program as Action Request A0665588.

The finding was greater than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone and impacted the cornerstone objective of providing reasonable assurance that physical design barriers, in this case the reactor coolant system, protect the public from radio-nuclide releases caused by accidents or events. The finding was determined to be of very low safety significance based on management review of the plant conditions at the time the performance deficiency occurred (defueled) and the condition was evaluated prior to the plant entering Mode 5.
Inspection Report# : 2006003 *(pdf)*

# Emergency Preparedness

**Significance: N/A** Apr 06, 2006
Identified By: NRC
Item Type: FIN Finding
**Acceptable performance in addressing performance indicator monitoring and accuracy**
The U.S. Nuclear Regulatory Commission (NRC) performed this supplemental inspection to assess the licensee’s evaluation associated with the failure to provide complete and accurate performance indicator data to the NRC. This performance issue was previously characterized as having low to moderate risk significance (White) in NRC Inspection Report 05000275, 05000323/2006005. During this supplemental inspection, performed in accordance with Inspection Procedure 95001, the inspector determined that the licensee conducted comprehensive evaluations of the missed performance indicator data and the failure to submit complete and accurate performance indicator information to the NRC. The licensee’s evaluations identified the primary root cause of the performance issue to be inconsistent standards, procedures, and policies which hindered implementation of the emergency plan, limited and inequitable emergency planning training, and the use of inexperienced emergency planning personnel. To determine the scope of the performance indicator issue, the licensee had a panel of subject matter experts review programs to identify similar error precursors. These experts identified programs that met the criteria. These programs were entered into the licensee's corrective action program and required that self-assessments be performed. The licensee also issued Action Requests to other performance indicator monitors to determine if other performance indicators were not meeting station goals or have a high potential or risk of not meeting them. In addition, procedures were revised to clarify procedure details. Given the licensee’s acceptable performance in addressing the performance indicator data monitoring and accuracy, the white finding associated with this issue will only be considered in assessing plant performance for a total of four quarters in accordance with the guidance in Inspection Manual Chapter 0305, “Operating Reactor Assessment Program." Implementation of the licensee’s corrective actions will be reviewed during a future inspection.
Inspection Report# : 2006010 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Apr 18, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey to identify the magnitude and extent of radiation levels to identify radiological hazards**
The inspectors identified a non-cited violation of 10 CFR 20.1501(a) because Pacific Gas and Electric Company (PG&E) failed to survey to determine the extent and magnitude of radiation levels and evaluate the radiological hazards. Specifically, on April 18, 2006, the inspectors identified elevated radiation levels near two chemical volume control system valves located in a hallway on the 100-foot elevation of Unit 2. PG&E confirmed elevated radiation levels near the valves were as high as 200 millirem per hour on contact and 28 millirem per hour at 30 centimeters. PG&E surveyed the area and entered the finding into their corrective action program as Action Request A0665039.

The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Exposure Control and Monitoring and affected the cornerstone objective to ensure the adequate protection of a worker’s health and safety from exposure to radiation because workers could have unknowingly received additional radiation exposure. When going through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not an as low as is reasonably achievable finding. There was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The finding also had cross-cutting aspects associated with human performance because because adequate resources were not established for the survey requirements.
Inspection Report# : 2006003 *(pdf)*

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# Public Radiation Safety

**Significance:** G Aug 29, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Survey Material Unconditionally Released**
The team reviewed a self-revealing, non-cited violation of 10 CFR 20.1501(a) that resulted in an unconditional release of radioactive material from the radiologically controlled area. Specifically, the contents of a vehicle cab were not removed and surveyed, resulting in the release of a contaminated safety harness from the radiologically controlled area. The safety harness remained in the protected area. The licensee determined the inadequate survey of the vehicle and its contents was caused by inadequate procedural guidance. As corrective action, the licensee plans to revise Procedure RCP D-614, "Release of Solid Materials from Radiologically Controlled Areas," Revision 9, to include instructions for the removal of such items from vehicles and the survey to detect contamination.

The failure to adequately survey a contaminated item to prevent its release from the radiologically controlled area is a performance deficiency. This finding is greater than minor because it was associated with a Public Radiation Safety cornerstone attribute (material release) and it affected the associated cornerstone objective in that the failure to control radioactive material decreases the licensee's assurance that the public will not receive unnecessary dose. Using the Public Radiation Safety Significance Determination Process, the team determined that the finding had very low safety significance because: (1) the finding was a radioactive material control finding, (2) it was not a transportation finding, (3) it did not result in public dose greater than 0.005 rem, and (4) radioactive material was not released from the protected area more than five times. Additionally, this finding has a cross-cutting aspect in the area of human performance associated with resources because the licensee did not have complete procedures, in that, the procedures did not provide sufficiently detailed guidance to ensure the surveying of vehicle contents prior to removal of the vehicle from the radiologically controlled area.
Inspection Report# : 2006013 *(pdf)*

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# Physical Protection

Physical Protection information not publicly available.

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# Miscellaneous

**Significance: N/A** Jun 22, 2006
Identified By: NRC
Item Type: FIN Finding
**Biennial problem identification and resolution assessment for 2006**
The team reviewed approximately 280 action requests, apparent cause evaluations, and root cause analyses, as well as supporting documents to assess problem identification and resolution activities. In general, the corrective action program procedures and processes were effective, thresholds for identifying issues were low, and corrective actions were adequate to address conditions adverse to quality. Notwithstanding the above, a number of self-revealing and NRC identified findings in each of these attributes of your problem identification and resolution program were noted over the past two years. Many of these findings were related to equipment deficiencies, some of which resulted in inoperable safety-related equipment. The team noted improvement in all three areas when comparing the results of this and more recent inspections when compared to inspections two years ago.

Based on the interviews conducted, the team concluded that a positive safety conscious work environment existed at Diablo Canyon Power Plant. The team determined that employees felt free to raise safety concerns to station managers and supervisors, the employee concerns program, and the NRC. However, the team noted two isolated incidents regarding the environment that did not foster openly raising safety concerns. The licensee had already taken actions to address the

concerns. All the interviewees believed that potential safety issues were being addressed.
Inspection Report# : 2006012 *(pdf)*

Last modified : March 01, 2007

# Diablo Canyon 2
# 1Q/2007 Plant Inspection Findings

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## Initiating Events

**Significance:** G May 16, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Preserve Corrective Action for Thimble Tube Wear**
A self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for the failure to apply adequate design control measures regarding the installation of thimble tubes with chrome-plated bands. Specifically, Pacific Gas and Electric Company installed thimble tubes with chrome-plated bands at the fuel assembly bottom nozzle/lower core plate interface to address flow-induced vibration wear. Due to the failure of engineering personnel to account for the chrome-plated bands in the thimble tube relocation procedure, the chrome-plated band on Thimble Tube L-13 was removed from its designed location at the fuel assembly bottom nozzle, thereby increasing the potential for thimble tube through-wall wear. This issue was entered into Pacific Gas and Electric Company's corrective action program as Nonconformance Report N0002211.

The finding is greater than minor because it is associated with the Initiating Events Cornerstone attribute of design control and affects the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because, assuming the worst-case degradation, the finding would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage or affect mitigating systems. Specifically, the inspectors verified the worst-case leakage, i.e., guillotine break, from a thimble tube at the fuel assembly bottom nozzle/lower core plate interface to be approximately 7 gpm versus the Technical Specification reactor coolant system identified leakage limit of 10 gpm. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because Pacific Gas and Electric Company removed a corrective action to prevent recurrence of significant thimble tube wear.
Inspection Report# : 2006005 *(pdf)*

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## Mitigating Systems

**Significance:** G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update 480 V Switchgear Heat Dissipation Calculaiton**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was determined for the failure of engineering personnel to appropriately update the heat dissipation calculation for vital 480 V switchgear rooms. Since 1994, Calculation 90-DC, "Heat Dissipation of Electrical Equipment – 480 V Switchgear," Revision 4, had not been updated with changes in analyzed bus electrical loading. The calculation was input to other ventilation calculations to determine air flow balancing to 480 V switchgear and inverter rooms. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Requests A0688992 and A0689527.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specifications allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather

initiating event.
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Sep 25, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Include Floor Drains Credited in the Flood Analysis Into the Maintenance Rule Program**
An NRC-identified, noncited violation of 10 CFR 50.65(b) was determined for the failure of engineering staff to include the auxiliary feedwater pump room floor drains within the scope of Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance at the Diablo Canyon Power Plant. Specifically, Calculation 76060, "Flooding Analysis G Area and Auxiliary Building," Revision 1, assumes that at least two of the three floor drains in the auxiliary feedwater pump rooms would be able to remove up to 316 gpm of water in the event of a flood. Despite their credited function in the flood analysis, engineering staff did not scope them into their monitoring program. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0678658.

The finding is greater than minor because it is associated with the Mitigating Systems cornerstone attribute of protection against external factors and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with operating experience because engineering personnel did not effectively incorporate pertinent industry operating experience into their program for evaluating the effectiveness of maintenance performed on AFW pump room floor drains.
Inspection Report# : 2006004 *(pdf)*

**Significance:** G Aug 23, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Identify that the Correct Equipment Necessary for Implementing EOP for Inadequate Core Cooling Was Not Pre-staged**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company failed to promptly identify that it had prestaged the wrong equipment (a flange hose connection with the wrong tread pattern) necessary to cross-connect the fire main water system to the auxiliary feedwater system during a loss of core cooling event. This performance deficiency was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0676729.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of one or more risk-significant non-TS trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a crosscutting aspect in the area of human performance associated with resources because the licensee did not ensure that equipment needed to perform an EOP was available and adequate to assure nuclear safety.
Inspection Report# : 2006004 *(pdf)*

**Significance:** G Apr 20, 2006
Identified By: NRC
Item Type: NCV NonCited Violation

**Inadequate refueling procedure for draining and depressurizing the reactor coolant system**
An NRC-identified, non-cited violation of Technical Specification 5.4.1.a was determined for an inadequate procedure, Procedure OP A-2:II, “Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in Vessel,” Revision 33A. Specifically, on April 20, 2006, while operators depressurized the reactor coolant system (RCS), with water level 2 ft below the reactor vessel flange, the two required level instruments, wide-range reactor vessel refueling level indication system and LI-400, read 15 inches higher than actual reactor vessel water level. The inspectors determined that the procedure was not adequate because prior operating experience had not been incorporated into the procedure that demonstrated the level instruments would read non-conservatively during RCS depressurization. Also, Procedure OP A-2:II did not have criteria that alerted operators to abnormal level instrument deviations that may be caused by phenomenon outside of the level deviations expected by the RCS depressurization. Pacific Gas and Electric Company (PG&E) has planned to evaluate potential changes to Procedure OP A-2:II and RCS water level instrumentation when used during RCS depressurization. This issue was entered into PG&E's corrective action program as Action Requests A0664484, A0672419, and A0672422.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Appendix G, Attachment 1, Checklist 3, the finding is determined to be of very low safety significance since an optional set of instrumentation provided accurate RCS level indication and there was no loss of RCS inventory control. The finding had a cross-cutting aspect in the area of human performance for resources because PG&E failed to ensure the adequacy of procedures used for reactor vessel level monitoring to ensure nuclear safety.
Inspection Report# : 2006003 *(pdf)*

**Significance:** G Feb 09, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Change to Auxiliary Saltwater Pump Routine Surveillance Test Acceptance Criteria Inadequate Change to Auxiliary Saltwater Pump Routine Surveillance Test Acceptance Criteria**
GREEN. An NRC-identified, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of engineering personnel to apply adequate design control measures. Specifically, on February 9, 2006, engineering personnel changed the acceptance criteria in the auxiliary saltwater pump surveillance test from greater than zero packing leak-off to zero packing leak-off with packing gland temperature less than 120°F. The acceptance criteria change was based on engineering judgment even though vendor documentation called for greater than zero packing leak-off to prevent packing and pump shaft damage. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0684631.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to be of very low safety significance because it did not represent an actual loss of system safety function, did not represent an actual loss of a single train for greater than its Technical Specification allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance because engineering personnel failed to provide up-to-date design documentation to support a design change in surveillance test acceptance criteria.
Inspection Report# : 2006005 *(pdf)*

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## Barrier Integrity

**Significance:** G Jan 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Inappropriate Temporary Modification to Control Room Condenser**

An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was determined for the failure of maintenance personnel to make modifications to the Control Room Condenser CR-38 filter mount consistent with the component's design documentation and Procedure CF4.ID7, "Temporary Modifications," Revision 18. Specifically, on August 15, 2006, maintenance personnel used vice-grip pliers, C-clamps, and plastic tie-wraps to secure in place the filter mount, which was significantly corroded. The modification had not been documented or analyzed at the time it was placed into service. After subsequent engineering reviews, the condenser was considered inoperable due to the loss of seismic qualification. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0688202.

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of design control for the control room barrier and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding did not represent degradation of the barrier function of the control room against radiological hazards, smoke, or toxic atmosphere. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, in that maintenance personnel failed to adequately identify the degraded condition of the control room condenser when it was initially discovered.
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Sep 29, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Evaluate Operability of Auxiliary Building Ventilation Control Panels**
An NRC-identified, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," was determined for the failure of engineering and operations personnel to promptly identify and correct a condition adverse to quality. On two occasions between September 29 and November 9, 2006, operations and engineering personnel (1) failed to address operability when using manual actions in place of automatic actions associated with the auxiliary building ventilation system and (2) failed to fully address the impact of debris between the circuit card and the panel connections of the auxiliary building ventilation system. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0678429.

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of structure, system, and component and barrier performance and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radio-nuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding only represents a degradation of the radiological barrier function provided for the auxiliary building. This finding has a cross-cutting aspect in the area of problem identification and resolution because operations and engineering personnel did not adequately evaluate operability of the auxiliary building ventilation system due to the failure to fully encompass all aspects of the degraded conditions and corresponding compensatory measures.

Inspection Report# : 2006005 *(pdf)*

**Significance:** G May 03, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to follow welding procedures**
An NRC-identified, non-cited violation of Technical Specification 5.4.1 was identified because Pacific Gas and Electric Company (PG&E) failed to follow the procedure for ensuring that welding preheat temperatures were verified prior to welding. Specifically, during the replacement of Component Cooling Water Valves 279 and 280, which provide cooling to the reactor vessel support pads, PG&E failed to verify that the minimum welding preheat temperature of 50°F was met and could not demonstrate that the ambient temperature was greater than 50°F. PG&E entered the finding into their corrective action program as Action Request A0665588.

The finding was greater than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone and impacted the cornerstone objective of providing reasonable assurance that physical design barriers, in this

case the reactor coolant system, protect the public from radio-nuclide releases caused by accidents or events. The finding was determined to be of very low safety significance based on management review of the plant conditions at the time the performance deficiency occurred (defueled) and the condition was evaluated prior to the plant entering Mode 5.
Inspection Report# : 2006003 *(pdf)*

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# Emergency Preparedness

**Significance: N/A** Apr 06, 2006
Identified By: NRC
Item Type: FIN Finding
**Acceptable performance in addressing performance indicator monitoring and accuracy**
The U.S. Nuclear Regulatory Commission (NRC) performed this supplemental inspection to assess the licensee's evaluation associated with the failure to provide complete and accurate performance indicator data to the NRC. This performance issue was previously characterized as having low to moderate risk significance (White) in NRC Inspection Report 05000275, 05000323/2006005. During this supplemental inspection, performed in accordance with Inspection Procedure 95001, the inspector determined that the licensee conducted comprehensive evaluations of the missed performance indicator data and the failure to submit complete and accurate performance indicator information to the NRC. The licensee's evaluations identified the primary root cause of the performance issue to be inconsistent standards, procedures, and policies which hindered implementation of the emergency plan, limited and inequitable emergency planning training, and the use of inexperienced emergency planning personnel. To determine the scope of the performance indicator issue, the licensee had a panel of subject matter experts review programs to identify similar error precursors. These experts identified programs that met the criteria. These programs were entered into the licensee's corrective action program and required that self-assessments be performed. The licensee also issued Action Requests to other performance indicator monitors to determine if other performance indicators were not meeting station goals or have a high potential or risk of not meeting them. In addition, procedures were revised to clarify procedure details. Given the licensee's acceptable performance in addressing the performance indicator data monitoring and accuracy, the white finding associated with this issue will only be considered in assessing plant performance for a total of four quarters in accordance with the guidance in Inspection Manual Chapter 0305, "Operating Reactor Assessment Program." Implementation of the licensee's corrective actions will be reviewed during a future inspection.
Inspection Report# : 2006010 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Apr 18, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to survey to identify the magnitude and extent of radiation levels to identify radiological hazards**
The inspectors identified a non-cited violation of 10 CFR 20.1501(a) because Pacific Gas and Electric Company (PG&E) failed to survey to determine the extent and magnitude of radiation levels and evaluate the radiological hazards. Specifically, on April 18, 2006, the inspectors identified elevated radiation levels near two chemical volume control system valves located in a hallway on the 100-foot elevation of Unit 2. PG&E confirmed elevated radiation levels near the valves were as high as 200 millirem per hour on contact and 28 millirem per hour at 30 centimeters. PG&E surveyed the area and entered the finding into their corrective action program as Action Request A0665039.

The finding was greater than minor because it was associated with the Occupational Radiation Safety Cornerstone attribute of Exposure Control and Monitoring and affected the cornerstone objective to ensure the adequate protection of a worker's health and safety from exposure to radiation because workers could have unknowingly received additional radiation exposure. When going through the Occupational Radiation Safety Significance Determination Process, the finding was determined to be of very low safety significance because it was not an as low as is reasonably achievable finding. There was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The finding also had cross-cutting aspects associated with human performance because because adequate resources were not established for the survey requirements.

Inspection Report# : 2006003 *(pdf)*

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## Public Radiation Safety

**Significance:** G Aug 29, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Survey Material Unconditionally Released**
The team reviewed a self-revealing, non-cited violation of 10 CFR 20.1501(a) that resulted in an unconditional release of radioactive material from the radiologically controlled area. Specifically, the contents of a vehicle cab were not removed and surveyed, resulting in the release of a contaminated safety harness from the radiologically controlled area. The safety harness remained in the protected area. The licensee determined the inadequate survey of the vehicle and its contents was caused by inadequate procedural guidance. As corrective action, the licensee plans to revise Procedure RCP D-614, "Release of Solid Materials from Radiologically Controlled Areas," Revision 9, to include instructions for the removal of such items from vehicles and the survey to detect contamination.

The failure to adequately survey a contaminated item to prevent its release from the radiologically controlled area is a performance deficiency. This finding is greater than minor because it was associated with a Public Radiation Safety cornerstone attribute (material release) and it affected the associated cornerstone objective in that the failure to control radioactive material decreases the licensee's assurance that the public will not receive unnecessary dose. Using the Public Radiation Safety Significance Determination Process, the team determined that the finding had very low safety significance because: (1) the finding was a radioactive material control finding, (2) it was not a transportation finding, (3) it did not result in public dose greater than 0.005 rem, and (4) radioactive material was not released from the protected area more than five times. Additionally, this finding has a cross-cutting aspect in the area of human performance associated with resources because the licensee did not have complete procedures, in that, the procedures did not provide sufficiently detailed guidance to ensure the surveying of vehicle contents prior to removal of the vehicle from the radiologically controlled area.
Inspection Report# : 2006013 *(pdf)*

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## Physical Protection

Physical Protection information not publicly available.

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## Miscellaneous

**Significance: N/A** Jun 22, 2006
Identified By: NRC
Item Type: FIN Finding
**Biennial problem identification and resolution assessment for 2006**
The team reviewed approximately 280 action requests, apparent cause evaluations, and root cause analyses, as well as supporting documents to assess problem identification and resolution activities. In general, the corrective action program procedures and processes were effective, thresholds for identifying issues were low, and corrective actions were adequate to address conditions adverse to quality. Notwithstanding the above, a number of self-revealing and NRC identified findings in each of these attributes of your problem identification and resolution program were noted over the past two years. Many of these findings were related to equipment deficiencies, some of which resulted in inoperable safety-related equipment. The team noted improvement in all three areas when comparing the results of this and more recent inspections when compared to inspections two years ago.

Based on the interviews conducted, the team concluded that a positive safety conscious work environment existed at Diablo

Canyon Power Plant. The team determined that employees felt free to raise safety concerns to station managers and supervisors, the employee concerns program, and the NRC. However, the team noted two isolated incidents regarding the environment that did not foster openly raising safety concerns. The licensee had already taken actions to address the concerns. All the interviewees believed that potential safety issues were being addressed.
Inspection Report# : 2006012 *(pdf)*

Last modified : June 01, 2007

# Diablo Canyon 2
# 2Q/2007 Plant Inspection Findings

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## Initiating Events

**Significance:** G Aug 31, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Preserve Corrective Action for Thimble Tube Wear**
A self-revealing, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified for the failure to apply adequate design control measures regarding the installation of thimble tubes with chrome-plated bands. Specifically, Pacific Gas and Electric Company installed thimble tubes with chrome-plated bands at the fuel assembly bottom nozzle/lower core plate interface to address flow-induced vibration wear. Due to the failure of engineering personnel to account for the chrome-plated bands in the thimble tube relocation procedure, the chrome-plated band on Thimble Tube L-13 was removed from its designed location at the fuel assembly bottom nozzle, thereby increasing the potential for thimble tube through-wall wear. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Nonconformance Report N0002211.

The finding is greater than minor because it is associated with the Initiating Events Cornerstone attribute of design control and affects the associated cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because, assuming the worst-case degradation, the finding would not result in exceeding the Technical Specification limit for identified reactor coolant system leakage or affect mitigating systems. Specifically, the inspectors verified the worst-case leakage, i.e., guillotine break, from a thimble tube at the fuel assembly bottom nozzle/lower core plate interface to be approximately 7 gpm versus the Technical Specification reactor coolant system identified leakage limit of 10 gpm. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because Pacific Gas and Electric Company removed a corrective action to prevent recurrence of significant thimble tube wear.
Inspection Report# : 2006005 *(pdf)*

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## Mitigating Systems

**Significance:** G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update 480 V Switchgear Heat Dissipation Calculaiton**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of engineering personnel to appropriately update the heat dissipation calculation for vital 480 V switchgear rooms. Since 1994, Calculation 90-DC, “Heat Dissipation of Electrical Equipment – 480 V Switchgear,” Revision 4, had not been updated with changes in analyzed bus electrical loading. The calculation was input to other ventilation calculations to determine air flow balancing to 480 V switchgear and inverter rooms. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Requests A0688992 and A0689527.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train

for greater than its Technical Specifications allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Jan 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Use Correct Design Inputs in Determination of a Potential for Choking Flow/Cavitation Across the Auxiliary Service Water Throttled Butterfly Valves**
Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to translate design basis information into specifications and procedures. The team identified that a nonconservative flow rate was used as an input in engineering design calculations resulting in the potential for choked flow at the discharge valves for the Unit 1 auxiliary service water system. Choked flow turbulence is a wear concern for these components, and can result in system failure. The licensee entered this finding into their corrective action program as Action Requests A0678338 and A0678472.

The finding is more than minor because the error affected the Mitigating System Cornerstone objective (Design Control attribute) of ensuring availability, reliability, and capability of the auxiliary service water systems to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet, the issue screened as having very low safety significance because 1) did not represent a loss of system safety function; and 2) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment; and did not screen as potentially risk significant because of a seismic, flooding, or sever weather initiating event.

Inspection Report# : 2006011 *(pdf)*

**Significance:** G Jan 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Consider Instrument Uncertainty in Surveillance Requirements for Technical Specifications LCO 3.7.9**
Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to demonstrate that the acceptance criteria for surveillance tests had conservatively accounted for uncertainties in determination of the minimum allowed ultimate heat sink temperature. Specifically, the team identified that the acceptance criteria specified in the Surveillance Test Procedure STP I-1A, Routine Shift Checks Required by the Licensee, Revision 101, did not correctly account for instrument uncertainty. The licensee entered this finding into their corrective action program as Action Request A0682398.

The finding is more than minor because the error affected the Mitigating System cornerstone objective (Design Control attribute) of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet, the issue screened as having very low safety significance because 1) did not represent a loss of system safety function; and 2) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment; and did not screen as potentially risk significant because of a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2006011 *(pdf)*

**Significance:** G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Change to Auxiliary Saltwater Pump Routine Surveillance Test Acceptance Criteria**
GREEN. An NRC-identified, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was determined for the failure of engineering personnel to apply adequate design control measures. Specifically, on February 9, 2006, engineering personnel changed the acceptance criteria in the auxiliary saltwater pump surveillance test from greater than zero packing leak-off to zero packing leak-off with packing gland temperature less than 120°F.

The acceptance criteria change was based on engineering judgment even though vendor documentation called for greater than zero packing leak-off to prevent packing and pump shaft damage. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0684631.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to be of very low safety significance because it did not represent an actual loss of system safety function, did not represent an actual loss of a single train for greater than its Technical Specification allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance because engineering personnel failed to provide up-to-date design documentation to support a design change in surveillance test acceptance criteria.
Inspection Report# : 2006005 *(pdf)*

**Significance:** G Sep 25, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Include Floor Drains Credited in the Flood Analysis Into the Maintenance Rule Program**
An NRC-identified, noncited violation of 10 CFR 50.65(b) was determined for the failure of engineering staff to include the auxiliary feedwater pump room floor drains within the scope of Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance at the Diablo Canyon Power Plant. Specifically, Calculation 76060, "Flooding Analysis G Area and Auxiliary Building," Revision 1, assumes that at least two of the three floor drains in the auxiliary feedwater pump rooms would be able to remove up to 316 gpm of water in the event of a flood. Despite their credited function in the flood analysis, engineering staff did not scope them into their monitoring program. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0678658.

The finding is greater than minor because it is associated with the Mitigating Systems cornerstone attribute of protection against external factors and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, did not represent an actual loss of one or more risk-significant non-Technical Specification trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a cross-cutting aspect in the area of problem identification and resolution associated with operating experience because engineering personnel did not effectively incorporate pertinent industry operating experience into their program for evaluating the effectiveness of maintenance performed on AFW pump room floor drains.
Inspection Report# : 2006004 *(pdf)*

**Significance:** G Aug 23, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Promptly Identify that the Correct Equipment Necessary for Implementing EOP for Inadequate Core Cooling Was Not Pre-staged**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," for the failure to promptly identify a condition adverse to quality. Specifically, Pacific Gas and Electric Company failed to promptly identify that it had prestaged the wrong equipment (a flange hose connection with the wrong tread pattern) necessary to cross-connect the fire main water system to the auxiliary feedwater system during a loss of core cooling event. This performance deficiency was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0676729.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter

0609, “Significance Determination Process,” Phase 1 Worksheet, the inspectors determined that this finding is of very low safety significance because the condition did not represent a loss of system safety function, did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time, did not represent an actual loss of one or more risk-significant non-TS trains of equipment for greater than 24 hours, and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding has a crosscutting aspect in the area of human performance associated with resources because the licensee did not ensure that equipment needed to perform an EOP was available and adequate to assure nuclear safety.
Inspection Report# : 2006004 *(pdf)*

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# Barrier Integrity

**Significance:** G Jan 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Inappropriate Temporary Modification to Control Room Condenser**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of maintenance personnel to make modifications to the Control Room Condenser CR-38 filter mount consistent with the component’s design documentation and Procedure CF4.ID7, “Temporary Modifications,” Revision 18. Specifically, on August 15, 2006, maintenance personnel used vice-grip pliers, C-clamps, and plastic tie-wraps to secure in place the filter mount, which was significantly corroded. The modification had not been documented or analyzed at the time it was placed into service. After subsequent engineering reviews, the condenser was considered inoperable due to the loss of seismic qualification. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0688202.

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of design control for the control room barrier and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding did not represent degradation of the barrier function of the control room against radiological hazards, smoke, or toxic atmosphere. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, in that maintenance personnel failed to adequately identify the degraded condition of the control room condenser when it was initially discovered.
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Sep 29, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Evaluate Operability of Auxiliary Building Ventilation Control Panels**
An NRC-identified, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Actions,” was determined for the failure of engineering and operations personnel to promptly identify and correct a condition adverse to quality. On two occasions between September 29 and November 9, 2006, operations and engineering personnel (1) failed to address operability when using manual actions in place of automatic actions associated with the auxiliary building ventilation system and (2) failed to fully address the impact of debris between the circuit card and the panel connections of the auxiliary building ventilation system. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0678429.

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of structure, system, and component and barrier performance and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radio-nuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding only represents a degradation of the radiological barrier function provided for the auxiliary building. This finding has a cross-cutting aspect in the area of problem identification and resolution because operations and engineering personnel did not adequately evaluate operability of the auxiliary building ventilation system due to the failure to fully encompass all aspects of the degraded conditions and corresponding compensatory measures.

Inspection Report# : 2006005 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

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# Public Radiation Safety

**Significance:** G Aug 29, 2006
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Survey Material Unconditionally Released**
The team reviewed a self-revealing, non-cited violation of 10 CFR 20.1501(a) that resulted in an unconditional release of radioactive material from the radiologically controlled area. Specifically, the contents of a vehicle cab were not removed and surveyed, resulting in the release of a contaminated safety harness from the radiologically controlled area. The safety harness remained in the protected area. The licensee determined the inadequate survey of the vehicle and its contents was caused by inadequate procedural guidance. As corrective action, the licensee plans to revise Procedure RCP D-614, “Release of Solid Materials from Radiologically Controlled Areas,” Revision 9, to include instructions for the removal of such items from vehicles and the survey to detect contamination.

The failure to adequately survey a contaminated item to prevent its release from the radiologically controlled area is a performance deficiency. This finding is greater than minor because it was associated with a Public Radiation Safety cornerstone attribute (material release) and it affected the associated cornerstone objective in that the failure to control radioactive material decreases the licensee’s assurance that the public will not receive unnecessary dose. Using the Public Radiation Safety Significance Determination Process, the team determined that the finding had very low safety significance because: (1) the finding was a radioactive material control finding, (2) it was not a transportation finding, (3) it did not result in public dose greater than 0.005 rem, and (4) radioactive material was not released from the protected area more than five times. Additionally, this finding has a cross-cutting aspect in the area of human performance associated with resources because the licensee did not have complete procedures, in that, the procedures did not provide sufficiently detailed guidance to ensure the surveying of vehicle contents prior to removal of the vehicle from the radiologically controlled area.
Inspection Report# : 2006013 *(pdf)*

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# Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

Last modified : August 24, 2007

# Diablo Canyon 2
# 3Q/2007 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Scope Reactor Cavity and Containment Structure Sump Level Indication Systems Into the Maintenance Rule Program**
GREEN. The inspectors identified a Green, noncited violation of 10 CFR 50.65(b) was identified for the failure of engineering personnel to include the reactor cavity and containment structure sump level indication systems into the scope of its program for monitoring the effectiveness of maintenance. Specifically, between April 14, 2007 and May 17, 2007, Units 1 and 2 experienced multiple failures of the reactor cavity and containment structure sump level indications. These systems are required by the plant's Technical Specifications in order to promptly identify and take actions for reactor coolant system leaks before they can potentially develop into a loss of coolant accident. Additionally, the inspectors discovered that Emergency Operating Procedure ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired," Revision 18, utilized the containment structure sump level indication for mitigative actions. Based on the fact that the systems are used to mitigate a loss of coolant accident and were used in the emergency operating procedures, the inspectors determined that the systems should have been included in Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0696295.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance, associated with the decision-making component, in that Pacific Gas and Electric Company failed to use conservative assumptions in evaluating the function and use of the sump level indications in mitigating the effects of design basis accidents (H.1(b)).
Inspection Report# : 2007003 *(pdf)*

**Significance:** G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update 480 V Switchgear Heat Dissipation Calculaiton**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was determined for the failure of engineering personnel to appropriately update the heat dissipation calculation for vital 480 V switchgear rooms. Since 1994, Calculation 90-DC, "Heat Dissipation of Electrical Equipment – 480 V Switchgear," Revision 4, had not been updated with changes in analyzed bus electrical loading. The calculation was input to other ventilation calculations to determine air flow balancing to 480 V switchgear and inverter rooms. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Requests A0688992 and A0689527.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of

procedure quality and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specifications allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Jan 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Use Correct Design Inputs in Determination of a Potential for Choking Flow/Cavitation Across the Auxiliary Service Water Throttled Butterfly Valves**
Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to translate design basis information into specifications and procedures. The team identified that a nonconservative flow rate was used as an input in engineering design calculations resulting in the potential for choked flow at the discharge valves for the Unit 1 auxiliary service water system. Choked flow turbulence is a wear concern for these components, and can result in system failure. The licensee entered this finding into their corrective action program as Action Requests A0678338 and A0678472.

The finding is more than minor because the error affected the Mitigating System Cornerstone objective (Design Control attribute) of ensuring availability, reliability, and capability of the auxiliary service water systems to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet, the issue screened as having very low safety significance because 1) did not represent a loss of system safety function; and 2) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment; and did not screen as potentially risk significant because of a seismic, flooding, or sever weather initiating event.

Inspection Report# : 2006011 *(pdf)*

**Significance:** G Jan 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Consider Instrument Uncertainty in Surveillance Requirements for Technical Specifications LCO 3.7.9**
Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to demonstrate that the acceptance criteria for surveillance tests had conservatively accounted for uncertainties in determination of the minimum allowed ultimate heat sink temperature. Specifically, the team identified that the acceptance criteria specified in the Surveillance Test Procedure STP I-1A, Routine Shift Checks Required by the Licensee, Revision 101, did not correctly account for instrument uncertainty. The licensee entered this finding into their corrective action program as Action Request A0682398.

The finding is more than minor because the error affected the Mitigating System cornerstone objective (Design Control attribute) of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet, the issue screened as having very low safety significance because 1) did not represent a loss of system safety function; and 2) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment; and did not screen as potentially risk significant because of a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2006011 *(pdf)*

**Significance:** G Dec 31, 2006
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Change to Auxiliary Saltwater Pump Routine Surveillance Test Acceptance Criteria**

GREEN. An NRC-identified, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of engineering personnel to apply adequate design control measures. Specifically, on February 9, 2006, engineering personnel changed the acceptance criteria in the auxiliary saltwater pump surveillance test from greater than zero packing leak-off to zero packing leak-off with packing gland temperature less than 120°F. The acceptance criteria change was based on engineering judgment even though vendor documentation called for greater than zero packing leak-off to prevent packing and pump shaft damage. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0684631.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to be of very low safety significance because it did not represent an actual loss of system safety function, did not represent an actual loss of a single train for greater than its Technical Specification allowed outage time, and the finding did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance because engineering personnel failed to provide up-to-date design documentation to support a design change in surveillance test acceptance criteria.
Inspection Report# : 2006005 *(pdf)*

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# Barrier Integrity

**Significance:** G Jan 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Inappropriate Temporary Modification to Control Room Condenser**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of maintenance personnel to make modifications to the Control Room Condenser CR-38 filter mount consistent with the component’s design documentation and Procedure CF4.ID7, “Temporary Modifications,” Revision 18. Specifically, on August 15, 2006, maintenance personnel used vice-grip pliers, C-clamps, and plastic tie-wraps to secure in place the filter mount, which was significantly corroded. The modification had not been documented or analyzed at the time it was placed into service. After subsequent engineering reviews, the condenser was considered inoperable due to the loss of seismic qualification. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0688202.

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of design control for the control room barrier and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding did not represent degradation of the barrier function of the control room against radiological hazards, smoke, or toxic atmosphere. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, in that maintenance personnel failed to adequately identify the degraded condition of the control room condenser when it was initially discovered.
Inspection Report# : 2007002 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

## Public Radiation Safety

## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

## Miscellaneous

Last modified : December 07, 2007

# Diablo Canyon 2
# 4Q/2007 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Scope Reactor Cavity and Containment Structure Sump Level Indication Systems Into the Maintenance Rule Program**
GREEN. The inspectors identified a Green, noncited violation of 10 CFR 50.65(b) was identified for the failure of engineering personnel to include the reactor cavity and containment structure sump level indication systems into the scope of its program for monitoring the effectiveness of maintenance. Specifically, between April 14, 2007 and May 17, 2007, Units 1 and 2 experienced multiple failures of the reactor cavity and containment structure sump level indications. These systems are required by the plant's Technical Specifications in order to promptly identify and take actions for reactor coolant system leaks before they can potentially develop into a loss of coolant accident. Additionally, the inspectors discovered that Emergency Operating Procedure ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired," Revision 18, utilized the containment structure sump level indication for mitigative actions. Based on the fact that the systems are used to mitigate a loss of coolant accident and were used in the emergency operating procedures, the inspectors determined that the systems should have been included in Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0696295.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance, associated with the decision-making component, in that Pacific Gas and Electric Company failed to use conservative assumptions in evaluating the function and use of the sump level indications in mitigating the effects of design basis accidents (H.1(b)).
Inspection Report# : 2007003 *(pdf)*

**Significance:** G Feb 16, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update 480 V Switchgear Heat Dissipation Calculaiton**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was determined for the failure of engineering personnel to appropriately update the heat dissipation calculation for vital 480 V switchgear rooms. Since 1994, Calculation 90-DC, "Heat Dissipation of Electrical Equipment – 480 V Switchgear," Revision 4, had not been updated with changes in analyzed bus electrical loading. The calculation was input to other ventilation calculations to determine air flow balancing to 480 V switchgear and inverter rooms. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Requests A0688992 and A0689527.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of

procedure quality and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specifications allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2007002 *(pdf)*

**Significance:** G Jan 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Use Correct Design Inputs in Determination of a Potential for Choking Flow/Cavitation Across the Auxiliary Service Water Throttled Butterfly Valves**
Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to translate design basis information into specifications and procedures. The team identified that a nonconservative flow rate was used as an input in engineering design calculations resulting in the potential for choked flow at the discharge valves for the Unit 1 auxiliary service water system. Choked flow turbulence is a wear concern for these components, and can result in system failure. The licensee entered this finding into their corrective action program as Action Requests A0678338 and A0678472.

The finding is more than minor because the error affected the Mitigating System Cornerstone objective (Design Control attribute) of ensuring availability, reliability, and capability of the auxiliary service water systems to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet, the issue screened as having very low safety significance because 1) did not represent a loss of system safety function; and 2) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment; and did not screen as potentially risk significant because of a seismic, flooding, or sever weather initiating event.

Inspection Report# : 2006011 *(pdf)*

**Significance:** G Jan 11, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Consider Instrument Uncertainty in Surveillance Requirements for Technical Specifications LCO 3.7.9**
Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control, for the failure to demonstrate that the acceptance criteria for surveillance tests had conservatively accounted for uncertainties in determination of the minimum allowed ultimate heat sink temperature. Specifically, the team identified that the acceptance criteria specified in the Surveillance Test Procedure STP I-1A, Routine Shift Checks Required by the Licensee, Revision 101, did not correctly account for instrument uncertainty. The licensee entered this finding into their corrective action program as Action Request A0682398.

The finding is more than minor because the error affected the Mitigating System cornerstone objective (Design Control attribute) of ensuring availability, reliability, and capability of systems needed to respond to initiating events to prevent undesired consequences. Using the Manual Chapter 0609, Significance Determination Process, Phase 1 screening worksheet, the issue screened as having very low safety significance because 1) did not represent a loss of system safety function; and 2) did not represent an actual loss of safety function of one or more non-technical specification trains of equipment; and did not screen as potentially risk significant because of a seismic, flooding, or severe weather initiating event.
Inspection Report# : 2006011 *(pdf)*

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# Barrier Integrity

**Significance:** G Jan 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Inappropriate Temporary Modification to Control Room Condenser**
An NRC-identified, noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was determined for the failure of maintenance personnel to make modifications to the Control Room Condenser CR-38 filter mount consistent with the component’s design documentation and Procedure CF4.ID7, “Temporary Modifications,” Revision 18. Specifically, on August 15, 2006, maintenance personnel used vice-grip pliers, C-clamps, and plastic tie-wraps to secure in place the filter mount, which was significantly corroded. The modification had not been documented or analyzed at the time it was placed into service. After subsequent engineering reviews, the condenser was considered inoperable due to the loss of seismic qualification. This issue was entered into Pacific Gas and Electric Company’s corrective action program as Action Request A0688202.

The finding is greater than minor because it is associated with the Barrier Integrity Cornerstone attribute of design control for the control room barrier and affects the associated cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, “Significance Determination Process,” Phase 1 Worksheet, the finding is determined to have very low safety significance because the finding did not represent degradation of the barrier function of the control room against radiological hazards, smoke, or toxic atmosphere. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, in that maintenance personnel failed to adequately identify the degraded condition of the control room condenser when it was initially discovered.
Inspection Report# : 2007002 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : February 04, 2008

# Diablo Canyon 2
# 1Q/2008 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Demonstrate that the Unit 2 Containment Atmosphere Particulate Radioactivity Monitor Performance was Being Effectively Controlled per 10 CFR 50.65(a)(2)**
The inspectors identified a noncited violation of 10 CFR 50.65(a)(2), after Pacific Gas and Electric Company failed to effectively control performance monitoring of the Unit 2 containment atmosphere particulate radiation monitor through appropriate preventive maintenance. Eight functional failures of the radiation monitor occurred between November 2006 and January 2008. The licensee did not categorize any of these failures as Maintenance Rule functional failures.

This finding is greater than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and it affects the cornerstone objective to ensure the availability, reliability, and capability of the systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1, Appendix A. The inspectors determined that this finding was of very low safety significance because this is not a design or qualification deficiency, does not represent a loss of a system safety function or safety function of a single train, and does not screen as potentially risk significant due to external events. The inspectors also determined that this finding has a crosscutting aspect in the area of human performance associated with the work practices component because engineering staff failed to follow the November 2006 revision to the licensee maintenance rule procedure that would have required each failure to be counted as a maintenance rule functional failure. Engineering staff incorrectly concluded that the revision was not applicable to the radiation monitors and therefore did not implement the change.

Inspection Report# : 2008002 *(pdf)*

**Significance:** G Feb 17, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Integrity of an Auxiliary Building Fire Door**
On February 17, 2008, the inspectors identified a noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," after Pacific Gas and Electric failed to maintain the integrity of an auxiliary building fire door. The inspectors identified that the latching mechanism on Fire Door 348 was degraded and not engaged. The unlatched fire door resulted in a reduction in fire confinement capability. The door was required to provide a 1½-hour fire barrier between two plant fire areas. The licensee had several prior opportunities to identify the degraded fire door. Security and operations personnel passed through the affected fire area several times each day.

This finding is greater than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. Using Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors determined this finding is within the fire confinement category and the fire barrier was moderately degraded because the door latch was not functional. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water based fire suppression system was in place in the exposing fire area. This finding was entered into the corrective action program as Action Request A0719774. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because plant personnel did not maintain a low threshold for identifying issues.

Inspection Report# : 2008002 *(pdf)*

**Significance:** G Jun 30, 2007
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Scope Reactor Cavity and Containment Structure Sump Level Indication Systems Into the Maintenance Rule Program**
GREEN. The inspectors identified a Green, noncited violation of 10 CFR 50.65(b) was identified for the failure of engineering personnel to include the reactor cavity and containment structure sump level indication systems into the scope of its program for monitoring the effectiveness of maintenance. Specifically, between April 14, 2007 and May 17, 2007, Units 1 and 2 experienced multiple failures of the reactor cavity and containment structure sump level indications. These systems are required by the plant's Technical Specifications in order to promptly identify and take actions for reactor coolant system leaks before they can potentially develop into a loss of coolant accident. Additionally, the inspectors discovered that Emergency Operating Procedure ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired," Revision 18, utilized the containment structure sump level indication for mitigative actions. Based on the fact that the systems are used to mitigate a loss of coolant accident and were used in the emergency operating procedures, the inspectors determined that the systems should have been included in Pacific Gas and Electric Company's program for monitoring the effectiveness of maintenance. This issue was entered into Pacific Gas and Electric Company's corrective action program as Action Request A0696295.

The finding is greater than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance and affects the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheet, the finding is determined to have very low safety significance since it did not represent a loss of system safety function, an actual loss of safety function of a single train for greater than its Technical Specification allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance, associated with the decision-making component, in that Pacific Gas and Electric Company failed to use conservative assumptions in evaluating the function and use of the sump level indications in mitigating the effects of design basis accidents (H.1(b)).
Inspection Report# : 2007003 *(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Feb 13, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedures, per Technical Specification 5.4.1**
The inspectors identified a noncited violation of Technical Specification 5.4.1 for failure to follow a licensee procedure. Specifically, while touring the Unit 2 spent fuel pool on February 13, 2008, the inspectors observed workers performing fuel inspections on the fuel bridge. The inspectors noted that the physical location of a continuous air monitor, an AMS-4, was in the southeast corner of the floor. Ventilation flow in this area was north to south with negative ventilation centered on the spent fuel pool. Section 2.2 of Procedure RCP D-430 states, in part, the purpose

of the continuous air monitors was to alert personnel to changes in radiological conditions and that locations are selected based on their potential as contributors to airborne activity. The location of the continuous air monitor was not appropriate to alert the workers of changing radiological conditions. During review of this occurrence, the inspectors were made aware of a similar issue. Specifically, Action Request A0666110 was opened on May 3, 2006, to evaluate the adequacy of AMS 4 placement in the fuel building during fuel moves. This action request was currently open with a resolution date of December 15, 2008.

This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the failure to monitor for radioactive material in the air had the potential to increase personnel dose. This occurrence involves workers unplanned, unintended or potential for such dose; therefore, this finding was evaluated using the occupational radiation safety significance determination process. The inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable planning or work control issue; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. This finding also has a crosscutting aspect in the area of problem identification and resolution, corrective action component, because the licensee failed to take timely corrective actions to address safety issues.

Inspection Report# : 2008002 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

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## Miscellaneous

Last modified : June 05, 2008

# Diablo Canyon 2
# 2Q/2008 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Demonstrate that the Unit 2 Containment Atmosphere Particulate Radioactivity Monitor Performance was Being Effectively Controlled per 10 CFR 50.65(a)(2)**
The inspectors identified a noncited violation of 10 CFR 50.65(a)(2), after Pacific Gas and Electric Company failed to effectively control performance monitoring of the Unit 2 containment atmosphere particulate radiation monitor through appropriate preventive maintenance. Eight functional failures of the radiation monitor occurred between November 2006 and January 2008. The licensee did not categorize any of these failures as Maintenance Rule functional failures.

This finding is greater than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and it affects the cornerstone objective to ensure the availability, reliability, and capability of the systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1, Appendix A. The inspectors determined that this finding was of very low safety significance because this is not a design or qualification deficiency, does not represent a loss of a system safety function or safety function of a single train, and does not screen as potentially risk significant due to external events. The inspectors also determined that this finding has a crosscutting aspect in the area of human performance associated with the work practices component because engineering staff failed to follow the November 2006 revision to the licensee maintenance rule procedure that would have required each failure to be counted as a maintenance rule functional failure. Engineering staff incorrectly concluded that the revision was not applicable to the radiation monitors and therefore did not implement the change. [H4(b)]

Inspection Report# : 2008002 *(pdf)*

**Significance:** G Feb 17, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Integrity of an Auxiliary Building Fire Door**
On February 17, 2008, the inspectors identified a noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," after Pacific Gas and Electric failed to maintain the integrity of an auxiliary building fire door. The inspectors identified that the latching mechanism on Fire Door 348 was degraded and not engaged. The unlatched fire door resulted in a reduction in fire confinement capability. The door was required to provide a 1½-hour fire barrier between two plant fire areas. The licensee had several prior opportunities to identify the degraded fire door. Security and operations personnel passed through the affected fire area several times each day.

This finding is greater than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. Using Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors determined this finding is within the fire confinement category and the fire barrier was moderately degraded because the door latch was not functional. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water based fire suppression system was in place in the exposing fire area. This finding was entered into the corrective action program as Action Request A0719774. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because plant personnel did not maintain a low threshold for identifying issues.

Inspection Report# : 2008002 *(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Feb 13, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedures, per Technical Specification 5.4.1**
The inspectors identified a noncited violation of Technical Specification 5.4.1 for failure to follow a licensee procedure. Specifically, while touring the Unit 2 spent fuel pool on February 13, 2008, the inspectors observed workers performing fuel inspections on the fuel bridge. The inspectors noted that the physical location of a continuous air monitor, an AMS-4, was in the southeast corner of the floor. Ventilation flow in this area was north to south with negative ventilation centered on the spent fuel pool. Section 2.2 of Procedure RCP D-430 states, in part, the purpose of the continuous air monitors was to alert personnel to changes in radiological conditions and that locations are selected based on their potential as contributors to airborne activity. The location of the continuous air monitor was not appropriate to alert the workers of changing radiological conditions. During review of this occurrence, the inspectors were made aware of a similar issue. Specifically, Action Request A0666110 was opened on May 3, 2006, to evaluate the adequacy of AMS 4 placement in the fuel building during fuel moves. This action request was currently open with a resolution date of December 15, 2008.

This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the failure to monitor for radioactive material in the air had the potential to increase personnel dose. This occurrence involves workers unplanned, unintended or potential for such dose; therefore, this finding was evaluated using the occupational radiation safety significance determination process. The inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable planning or work control issue; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. This finding also has a crosscutting aspect in the area of problem identification and resolution, corrective action component, because the licensee failed to take timely corrective actions to address safety issues.

Inspection Report# : 2008002 *(pdf)*

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# Public Radiation Safety

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# Physical Protection

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# Miscellaneous

Last modified : August 29, 2008

# Diablo Canyon 2
# 3Q/2008 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a Safety Assessment for Following Discovery of Explosive Gas in the Auxiliary and Containment Buildings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric personnel failed to perform a safety assessment prior to implementing a temporary procedure on July 20, 2008, to transfer an explosive gas mixture from the waste gas system to the Unit 2 vent. The explosive mixture of oxygen and hydrogen was discovered in the Unit 2 reactor coolant drain tank, waste gas surge tank, and interconnecting piping. The licensee also identified that the Unit 2 pressurizer relief tank vapor space exceeded the lower flammable limits. The explosive and flammable gas created a condition outside the plant design bases and was inconsistent with safety analysis. Plant Procedure TS3.ID2, “Licensing Basis Impact Evaluation,” required the licensee to have performed a safety assessment prior to conducting activities outside the design bases and inconsistent with safety analysis. The licensee entered this condition into the corrective actions system as Action Request A0741069.

This finding is greater than minor because explosive and flammable gas within the containment and auxiliary buildings affected the Initiating Events Cornerstone objective to limit the likelihood of events that may upset plant stability and challenge critical safety functions during power operations and protect against external factors such as fire and explosions. The inspectors used Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” to analyze the significances of the finding. The inspectors determined this finding was a fire prevention and administrative controls category due to the failure to meet the equipment control guideline for combustible gas flammability limits. The inspectors concluded that that this finding is of very low safety significance because the condition represented a low degradation rating due to the lack of a direct ignition source. This finding has a crosscutting aspect in human performance in the area of Decision Making because the licensee failed to use the systematic process provided in Procedure TS3.ID when making a safety significant or risk-significant decision when faced with the unexpected explosive gas mixture within containment and auxiliary building plant systems.
Inspection Report# : 2008004 *(pdf)*

**Significance:** G Jun 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Operability Procedure**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after the licensee failed to adhere to several requirements in Administrative Procedure OM7.ID12, “Operability Determination,” Revision 11. Specifically, the licensee identified that it did not perform a prompt operability assessment for a condition adverse to quality until approximately 1 year after the immediate operability determination was performed. Also, the inspectors identified that when the prompt operability assessment was performed, it relied inappropriately on engineering judgment, for a complex issue, without an adequately documented basis for that judgment. The adverse condition was an
identified nonconformance related to the design basis because both units were operating at a full power average temperature less than the design value. The licensee has entered this into their corrective action program as Action Request A0723331 which details their planned correction actions.

The inspectors determined that the finding was more than minor because it is similar to Inspection Manual Chapter 0612, Appendix E, Minor Example 3(j) in that operability was questioned and both the licensee and the vendor had to perform significant work and analysis in order to fully address the operability impact of a low average temperature on operating Unit 1. In accordance with Inspection Manual Chapter 0609, “Significance Determination Process,” Attachment 4, Phase 1 - Initial Screening and Characterization of Findings, the inspectors concluded the finding was of very low safety significance (Green) because it did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions when it decided that engineering judgment alone was a sufficient basis for operability without a supporting plant specific analysis [H.1(b)] .

Inspection Report# : 2008003 *(pdf)*

**Significance:** G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Demonstrate that the Unit 2 Containment Atmosphere Particulate Radioactivity Monitor Performance was Being Effectively Controlled per 10 CFR 50.65(a)(2)**
The inspectors identified a noncited violation of 10 CFR 50.65(a)(2), after Pacific Gas and Electric Company failed to effectively control performance monitoring of the Unit 2 containment atmosphere particulate radiation monitor through appropriate preventive maintenance. Eight functional failures of the radiation monitor occurred between November 2006 and January 2008. The licensee did not categorize any of these failures as Maintenance Rule functional failures.

This finding is greater than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and it affects the cornerstone objective to ensure the availability, reliability, and capability of the systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1, Appendix A. The inspectors determined that this finding was of very low safety significance because this is not a design or qualification deficiency, does not represent a loss of a system safety function or safety function of a single train, and does not screen as potentially risk significant due to external events. The inspectors also determined that this finding has a crosscutting aspect in the area of human performance associated with the work practices component because engineering staff failed to follow the November 2006 revision to the licensee maintenance rule procedure that would have required each failure to be counted as a maintenance rule functional failure. Engineering staff incorrectly concluded that the revision was not applicable to the radiation monitors and therefore did not implement the change. [H4(b)]

Inspection Report# : 2008002 *(pdf)*

**Significance:** G Feb 17, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Integrity of an Auxiliary Building Fire Door**
On February 17, 2008, the inspectors identified a noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," after Pacific Gas and Electric failed to maintain the integrity of an auxiliary building fire door. The inspectors identified that the latching mechanism on Fire Door 348 was degraded and not engaged. The unlatched fire door resulted in a reduction in fire confinement capability. The door was required to provide a 1½-hour fire barrier between two plant fire areas. The licensee had several prior opportunities to identify the degraded fire door. Security and operations personnel passed through the affected fire area several times each day.

This finding is greater than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. Using Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors determined this finding is within the fire confinement category and the fire barrier was moderately degraded because the door latch was not functional. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water based fire suppression system was in place in the exposing fire area. This finding was entered into the corrective action program as Action Request A0719774. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because plant personnel did not maintain a low threshold for identifying issues.

Inspection Report# : 2008002 *(pdf)*

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# Barrier Integrity

**Significance:** G Sep 30, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Procedure Resulting in Inoperable Auxiliary Building Ventilation System**
The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1, "Procedures," after Pacific Gas and Electric personnel failed to provide adequate work instructions for removal of equipment from service, resulting in the inoperability of both Unit 2 auxiliary building ventilation system trains, a condition prohibited by plant technical specifications. The work instruction did not provide a step for properly realigning the system to maintain operability of one train. The licensee entered this condition into the corrective actions system as Notification 50070612.

This finding was more than minor because the loss of both ventilation trains affected the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events and the inadequate procedure affected the attribute of procedure quality. The finding was of very low safety significance because it only represented a degradation of the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance with a Work Practices component because Pacific Gas & Electric staff failed to perform an adequate prejob brief to address questions regarding the sequence of steps and operators proceeded with the clearance in the face of uncertainty.

Inspection Report# : 2008004 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

**Significance:** G Feb 13, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedures, per Technical Specification 5.4.1**
The inspectors identified a noncited violation of Technical Specification 5.4.1 for failure to follow a licensee procedure. Specifically, while touring the Unit 2 spent fuel pool on February 13, 2008, the inspectors observed workers performing fuel inspections on the fuel bridge. The inspectors noted that the physical location of a continuous air monitor, an AMS-4, was in the southeast corner of the floor. Ventilation flow in this area was north to south with negative ventilation centered on the spent fuel pool. Section 2.2 of Procedure RCP D-430 states, in part, the purpose of the continuous air monitors was to alert personnel to changes in radiological conditions and that locations are selected based on their potential as contributors to airborne activity. The location of the continuous air monitor was not appropriate to alert the workers of changing radiological conditions. During review of this occurrence, the inspectors were made aware of a similar issue. Specifically, Action Request A0666110 was opened on May 3, 2006, to evaluate the adequacy of AMS 4 placement in the fuel building during fuel moves. This action request was currently open with a resolution date of December 15, 2008.

This finding is greater than minor because it is associated with the occupational radiation safety program and process attribute and affected the cornerstone objective, in that the failure to monitor for radioactive material in the air had the potential to increase personnel dose. This occurrence involves workers unplanned, unintended or potential for such dose; therefore, this finding was evaluated using the occupational radiation safety significance determination process. The inspectors determined that this finding was of very low safety significance because it did not involve: (1) an as low as is reasonably achievable planning or work control issue; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess dose. This finding also has a crosscutting aspect in the area of problem identification and resolution, corrective action component, because the licensee failed to take timely corrective actions to address safety issues.

Inspection Report# : 2008002 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : November 26, 2008

# Diablo Canyon 2
# 4Q/2008 Plant Inspection Findings

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## Initiating Events

**Significance:** G Sep 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a Safety Assessment for Following Discovery of Explosive Gas in the Auxiliary and Containment Buildings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric personnel failed to perform a safety assessment prior to implementing a temporary procedure on July 20, 2008, to transfer an explosive gas mixture from the waste gas system to the Unit 2 vent. The explosive mixture of oxygen and hydrogen was discovered in the Unit 2 reactor coolant drain tank, waste gas surge tank, and interconnecting piping. The licensee also identified that the Unit 2 pressurizer relief tank vapor space exceeded the lower flammable limits. The explosive and flammable gas created a condition outside the plant design bases and was inconsistent with safety analysis. Plant Procedure TS3.ID2, “Licensing Basis Impact Evaluation,” required the licensee to have performed a safety assessment prior to conducting activities outside the design bases and inconsistent with safety analysis. The licensee entered this condition into the corrective actions system as Action Request A0741069.

This finding is greater than minor because explosive and flammable gas within the containment and auxiliary buildings affected the Initiating Events Cornerstone objective to limit the likelihood of events that may upset plant stability and challenge critical safety functions during power operations and protect against external factors such as fire and explosions. The inspectors used Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” to analyze the significances of the finding. The inspectors determined this finding was a fire prevention and administrative controls category due to the failure to meet the equipment control guideline for combustible gas flammability limits. The inspectors concluded that that this finding is of very low safety significance because the condition represented a low degradation rating due to the lack of a direct ignition source. This finding has a crosscutting aspect in human performance in the area of Decision Making because the licensee failed to use the systematic process provided in Procedure TS3.ID when making a safety significant or risk-significant decision when faced with the unexpected explosive gas mixture within containment and auxiliary building plant systems.
Inspection Report# : 2008004 *(pdf)*

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## Mitigating Systems

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Evaluation Following the Loss of Design Control for the 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria III, Design Control, after Pacific Gas and Electric Company failed to maintain and verify that the 500 kilo-Volt offsite power system met the design basis. The licensee was unable to retrieve design documents demonstrating in compliance with General Design Criteria 17. On October 28, 2008, the licensee completed an evaluation of the missing documentation and concluded that the 500 kilo-Volt system was in compliance with the plant design basis by a “road map” of pre-existing analysis supporting other plant requirements. Licensee personnel failed to verify the applicability of “road map” analysis as required by design control measures. The inspectors concluded that the licensee’s evaluation of the 500 kilo-Volt offsite power design measures was inadequate. The licensee assumed a 30 minute delay time was needed to align the

500 kilo-Volt offsite power to the plant vital buses. The licensee assumed no reactor coolant inventory loss would occur through the reactor coolant pump seals during the delay time. The inspectors determined that this assumption was incorrect and the supporting “road map” document was not appropriate to the application. The inspectors estimated about 12 gallons per minute of reactor coolant inventory would be lost through the seals during first 8 minutes and then increase to about 88 gallons per minute until seal injection could be reestablished. The licensee entered this condition into the corrective action program as Notification 50083989.

The finding is greater than minor because the failure to maintain design measures of the 500 kilo-Volt offsite power system is associated with the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors used Inspection Manual Chanter 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the significance of this finding. The inspectors concluded the finding is of low safety significance because the condition was not a design or qualification deficiency conformed to result in loss of operability or functionality. While this performance deficiency represented a latent issue, the inspectors determined that the licensee had reasonable opportunity to identify the problem during the evaluation of the impact of missing design documents during October 2008. The inspectors concluded that the performance deficiency was representative current licensee performance and assigned a crosscutting component of problem identification and resolution and corrective action program aspect because Pacific Gas and Electric Company did not thoroughly evaluate the missing design basis such that the resolutions address causes and extent of conditions, as necessary [P.1.(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Operation of 230 kV Offsite Power System Outside the Design Basis**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria III, Design Control, after Pacific Gas and Electric failed to adequately translate the design basis for the 230 kV preferred offsite power system into specifications and procedures. Between November 3 and 7, 2008, the licensee operated with both units aligned to a single startup transformer. This created a situation where a dual unit trip or trip on one unit and accident on the other unit could result in loss of the preferred immediate offsite power source offsite power to both units.
The finding is greater than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the significance of this finding. The inspectors concluded that the finding is a design deficiency that did result in loss of operability. However, the inspectors concluded the finding is of very low safety significance because the actual loss of safety function of the 230 kV offsite power system was less than the Technical Specification allowed outage time. The inspectors also concluded that the finding did not represent a loss of safety function for greater than 24 hours or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors determined that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the operability of the offsite power circuit prior to removing the Unit 2 startup transformer from service [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Implement Effective Actions to Correct an Adverse Trend**
The inspectors identified a finding after Pacific Gas and Electric was ineffective in addressing an adverse trend in missed quality control inspection hold points. Licensee Procedure OM7, “Corrective Action Program,” required that the licensee evaluate problems commensurate with their significance, determine the cause, and conduct a proper

evaluation and resolution of repeat occurrences. The procedure further required that corrective actions are completed in a timely manner consistent with the problem significance. On May 19, 2007, Pacific Gas and Electric identified an adverse trend of missing quality control inspection hold points and requested that an apparent cause evaluation be performed. On July 11, 2007, this adverse trend was also evaluated by the Quality Verification Department as part of an assessment of Refueling Outage 14 maintenance. In March 2008, the licensee completed the evaluations and corrective actions. During the subsequent Unit 2 refueling outage, the Quality Verification Department identified over 11 additional missing quality inspection hold points. The inspectors identified that the licensee's corrective actions were ineffective to correct the adverse trend in missing quality control inspection hold points. Pacific Gas and Electric Company entered this finding into the corrective action program as Notification 50135175.
The finding was more than minor because, if left uncorrected, the failure to perform inspections has the potential to lead to a more significant safety concern. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded that this finding was of very low safety significance because the uncorrected adverse trend did not represent a loss of system safety function, the loss of safety function of a single train for greater than its Technical Specification allowed outage time, actual loss of safety function of one or more non-Technical Specification trains greater than 24 hours, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating. The finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to thoroughly evaluate the adverse trend and take corrective actions that addressed the cause and extent of condition [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to provide adequate design control measures for verifying the emergency diesel generators meet the design basis. The inspectors requested to review the design control measures that Pacific Gas and Electric maintained to demonstrate compliance with General Design Criteria 17, "Electric Power Systems," design basis. The licensee was not able to retrieve the requested design control measures for the onsite electrical power systems. The licensee provided unit specific diesel loading calculations. The inspectors identified that the licensee failed to include all design basis accidents, a single limiting failure, consider bus frequency and voltage fluctuations, motor starting currents, or manually initiated loads in the calculation. In response to the inspectors' observations, the licensee performed an operability evaluation. The inspectors reviewed the evaluation and concluded that the emergency diesel generators remained operable and capable of performing their intended safety function. The licensee has entered this issue into the corrective action program as Notification 50163396.
This finding is greater than minor because the design control attribute of the Mitigating Systems Cornerstone and the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences are affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality. The inspectors did not assign a crosscutting aspect because the finding represented a latent design issue. Pacific Gas and Electric revised the calculations in September 2006 and did not have a recent opportunity to identify this issue.

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct Violations of the Seismically-Induced Systems Interaction Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly implement housekeeping procedures to prevent seismically-induced system

interactions. Specifically, the team identified two instances during a plant walk down where transient equipment was staged in the vicinity of safety-related equipment identified as seismically-induced system interaction targets. This transient equipment had not been analyzed to assess the risk to these safety-related components. Following identification by the team, licensee staff secured and analyzed the transient equipment. Licensee staff entered this finding into the corrective action program as Notifications 50084856 and 50084761.

The failure of plant personnel to follow the requirements to properly secure or analyze equipment in close proximity to sensitive equipment was a performance deficiency. The finding was more than minor because it was similar to Inspection Manual Chapter 0612, "Power Reactor Inspection Reports'" Appendix E, Example 3.j., in that it was indicative of a significant programmatic deficiency in the licensee's Seismically-Induced System Interactions Program that could lead to worse errors if uncorrected. Specifically, a change in program ownership in 2006 resulted in a degradation of the sensitivity of plant personnel to the risk of seismically-induced system interactions due to transient materials, insufficient training of plant personnel on the program, and an absence of quality records over an approximately two-year period. Using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance because it did not result in an actual loss of a system safety function, did not result in a loss of a single train of safety equipment for greater than its technical specification allowed outage time, did not involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a cross-cutting aspect in the area of human performance associated with the work practices area component because the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel failed to follow procedures [H.4(b)].

Inspection Report# : 2008008 *(pdf)*

**Significance:** G Jun 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Operability Procedure**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after the licensee failed to adhere to several requirements in Administrative Procedure OM7.ID12, "Operability Determination," Revision 11. Specifically, the licensee identified that it did not perform a prompt operability assessment for a condition adverse to quality until approximately 1 year after the immediate operability determination was performed. Also, the inspectors identified that when the prompt operability assessment was performed, it relied inappropriately on engineering judgment, for a complex issue, without an adequately documented basis for that judgment. The adverse condition was an identified nonconformance related to the design basis because both units were operating at a full power average temperature less than the design value. The licensee has entered this into their corrective action program as Action Request A0723331 which details their planned correction actions.

The inspectors determined that the finding was more than minor because it is similar to Inspection Manual Chapter 0612, Appendix E, Minor Example 3(j) in that operability was questioned and both the licensee and the vendor had to perform significant work and analysis in order to fully address the operability impact of a low average temperature on operating Unit 1. In accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 4, Phase 1 - Initial Screening and Characterization of Findings, the inspectors concluded the finding was of very low safety significance (Green) because it did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions when it decided that engineering judgment alone was a sufficient basis for operability without a supporting plant specific analysis [H.1(b)] .

Inspection Report# : 2008003 *(pdf)*

**Significance:** G Mar 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Demonstrate that the Unit 2 Containment Atmosphere Particulate Radioactivity Monitor Performance was Being Effectively Controlled per 10 CFR 50.65(a)(2)**
The inspectors identified a noncited violation of 10 CFR 50.65(a)(2), after Pacific Gas and Electric Company failed to effectively control performance monitoring of the Unit 2 containment atmosphere particulate radiation monitor through appropriate preventive maintenance. Eight functional failures of the radiation monitor occurred between November 2006 and January 2008. The licensee did not categorize any of these failures as Maintenance Rule functional failures.

This finding is greater than minor because it is associated with the mitigating systems cornerstone attribute of equipment performance and it affects the cornerstone objective to ensure the availability, reliability, and capability of the systems that respond to initiating events to prevent undesirable consequences. The inspectors evaluated the significance of this finding using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1, Appendix A. The inspectors determined that this finding was of very low safety significance because this is not a design or qualification deficiency, does not represent a loss of a system safety function or safety function of a single train, and does not screen as potentially risk significant due to external events. The inspectors also determined that this finding has a crosscutting aspect in the area of human performance associated with the work practices component because engineering staff failed to follow the November 2006 revision to the licensee maintenance rule procedure that would have required each failure to be counted as a maintenance rule functional failure. Engineering staff incorrectly concluded that the revision was not applicable to the radiation monitors and therefore did not implement the change. [H4(b)]

Inspection Report# : 2008002 *(pdf)*

**Significance:** G Feb 17, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Integrity of an Auxiliary Building Fire Door**
On February 17, 2008, the inspectors identified a noncited violation of Technical Specification 5.4.1.d, "Fire Protection Program," after Pacific Gas and Electric failed to maintain the integrity of an auxiliary building fire door. The inspectors identified that the latching mechanism on Fire Door 348 was degraded and not engaged. The unlatched fire door resulted in a reduction in fire confinement capability. The door was required to provide a 1½-hour fire barrier between two plant fire areas. The licensee had several prior opportunities to identify the degraded fire door. Security and operations personnel passed through the affected fire area several times each day.

This finding is greater than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. Using Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors determined this finding is within the fire confinement category and the fire barrier was moderately degraded because the door latch was not functional. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water based fire suppression system was in place in the exposing fire area. This finding was entered into the corrective action program as Action Request A0719774. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because plant personnel did not maintain a low threshold for identifying issues.

Inspection Report# : 2008002 *(pdf)*

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## Barrier Integrity

**Significance:** G Sep 30, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Procedure Resulting in Inoperable Auxiliary Building Ventilation System**
The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1, "Procedures," after Pacific Gas and Electric personnel failed to provide adequate work instructions for removal of equipment from service, resulting in the inoperability of both Unit 2 auxiliary building ventilation system trains, a condition prohibited by plant technical specifications. The work instruction did not provide a step for properly realigning the system to maintain operability of one train. The licensee entered this condition into the corrective actions system as Notification 50070612.

This finding was more than minor because the loss of both ventilation trains affected the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events and the inadequate procedure affected the attribute of procedure quality. The finding was of very low safety significance because it only represented a degradation of the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance with a Work Practices component because Pacific Gas & Electric staff failed to perform an adequate prejob brief to address questions regarding the sequence of steps and operators proceeded with the clearance in the face of uncertainty.

Inspection Report# : 2008004 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

**Significance:** G Feb 13, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Conduct an Adequate Evaluation**
Part 20.1501(a) of Title 10 of the Code of Federal Regulations requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in 10 CFR Part 20 and that are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. Pursuant to 10 CFR 20.1003, a "survey" means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation. Part 20.1201(a) of Title 10 of the Code of Federal Regulations states, in part, that the licensee shall control the occupational dose to individual adults to specified limits.

Contrary to this requirement, as of February 13, 2008, the licensee failed to perform an evaluation of the radiological conditions and the potential hazards during fuel pool activities that was necessary to comply with 10 CFR 20.1201(a) and that were reasonable under the circumstances to evaluate the concentrations of radioactive materials, and the potential radiological hazards that could be present. Specifically, the licensee failed to evaluate the placement of an AMS-4 monitor to ensure that surveys taken were appropriate for monitoring the concentrations of radioactive material and the potential radiological hazards that could be present. Because this failure to perform radiological surveys is of very low safety significance and has been entered into the licensee's corrective action program as Action Request A0719338, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000323/2008002 03, Failure to Conduct an Adequate Evaluation.

Inspection Report# : 2008002 *(pdf)*

## Public Radiation Safety

**Significance: SL-IV** Dec 11, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report**
The team identified a non-cited violation of 10 CFR 50.71(e) for the failure of the licensee to periodically (every 24 months) update its Final Safety Analysis Report Update with all changes made in the facility or procedures. Specifically, in July 2005, the licensee stopped using the boric acid evaporator system as described in the Final Safety Analysis Report Update, Section 11.2.6, and did not submit an update to the NRC regarding this operational change. This issue was entered into the licensee's corrective action program as Notification 50116337 and licensee representatives stated an update would be submitted.

The team determined that the failure to update the Final Safety Analysis Report Update to reflect changes made to the facility was a performance deficiency. This issue is subject to traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. The finding is characterized as a Severity Level IV, non-cited violation in accordance with NRC Enforcement Policy, Supplement I, Example D.6, in that, the erroneous information in the Final Safety Analysis Report Update was not used to make an unacceptable change to the facility or procedures.
Inspection Report# : 2008009 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Take Appropriate Actions to Correct an Identified Adverse Trend**
The team identified a finding for failure to take adequate corrective actions to correct adverse trends in control of radioactive and potentially contaminated material as required by the corrective action program. Specifically, between May 2005 and June 2008, the licensee on two occasions identified and failed to correct adverse trends in the control of radioactive and potentially contaminated material. Licensee staff entered this finding into the corrective action program as Notification 50085121.

The finding was more than minor because it affected the Public Radiation Safety cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Using Inspection Manual Chapter 0609 Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to have very low safety significance because the dose impact to a member of the public was less than or equal to 0.005 rem total effective dose equivalent . The finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action area component; because the licensee failed to thoroughly evaluate problems such that the resolution addressed the cause [P.1(c)].

Inspection Report# : 2008008 *(pdf)*

## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

# Miscellaneous

Last modified : April 07, 2009

# Diablo Canyon 2
# 1Q/2009 Plant Inspection Findings

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## Initiating Events

**Significance:** G Sep 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a Safety Assessment for Following Discovery of Explosive Gas in the Auxiliary and Containment Buildings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric personnel failed to perform a safety assessment prior to implementing a temporary procedure on July 20, 2008, to transfer an explosive gas mixture from the waste gas system to the Unit 2 vent. The explosive mixture of oxygen and hydrogen was discovered in the Unit 2 reactor coolant drain tank, waste gas surge tank, and interconnecting piping. The licensee also identified that the Unit 2 pressurizer relief tank vapor space exceeded the lower flammable limits. The explosive and flammable gas created a condition outside the plant design bases and was inconsistent with safety analysis. Plant Procedure TS3.ID2, "Licensing Basis Impact Evaluation," required the licensee to have performed a safety assessment prior to conducting activities outside the design bases and inconsistent with safety analysis. The licensee entered this condition into the corrective actions system as Action Request A0741069.

This finding is greater than minor because explosive and flammable gas within the containment and auxiliary buildings affected the Initiating Events Cornerstone objective to limit the likelihood of events that may upset plant stability and challenge critical safety functions during power operations and protect against external factors such as fire and explosions. The inspectors used Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," to analyze the significances of the finding. The inspectors determined this finding was a fire prevention and administrative controls category due to the failure to meet the equipment control guideline for combustible gas flammability limits. The inspectors concluded that that this finding is of very low safety significance because the condition represented a low degradation rating due to the lack of a direct ignition source. This finding has a crosscutting aspect in human performance in the area of Decision Making because the licensee failed to use the systematic process provided in Procedure TS3.ID when making a safety significant or risk-significant decision when faced with the unexpected explosive gas mixture within containment and auxiliary building plant systems.
Inspection Report# : 2008004 *(pdf)*

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## Mitigating Systems

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Operation of 230 kV Offsite Power System Outside the Design Basis**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria III, Design Control, after Pacific Gas and Electric failed to adequately translate the design basis for the 230 kV preferred offsite power system into specifications and procedures. Between November 3 and 7, 2008, the licensee operated with both units aligned to a single startup transformer. This created a situation where a dual unit trip or trip on one unit and accident on the other unit could result in loss of the preferred immediate offsite power source offsite power to both units.
The finding is greater than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the

Significance of Reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded that the finding is a design deficiency that did result in loss of operability. However, the inspectors concluded the finding is of very low safety significance because the actual loss of safety function of the 230 kV offsite power system was less than the Technical Specification allowed outage time. The inspectors also concluded that the finding did not represent a loss of safety function for greater than 24 hours or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors determined that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the operability of the offsite power circuit prior to removing the Unit 2 startup transformer from service [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Implement Effective Actions to Correct an Adverse Trend**
The inspectors identified a finding after Pacific Gas and Electric was ineffective in addressing an adverse trend in missed quality control inspection hold points. Licensee Procedure OM7, "Corrective Action Program," required that the licensee evaluate problems commensurate with their significance, determine the cause, and conduct a proper evaluation and resolution of repeat occurrences. The procedure further required that corrective actions are completed in a timely manner consistent with the problem significance. On May 19, 2007, Pacific Gas and Electric identified an adverse trend of missing quality control inspection hold points and requested that an apparent cause evaluation be performed. On July 11, 2007, this adverse trend was also evaluated by the Quality Verification Department as part of an assessment of Refueling Outage 14 maintenance. In March 2008, the licensee completed the evaluations and corrective actions. During the subsequent Unit 2 refueling outage, the Quality Verification Department identified over 11 additional missing quality inspection hold points. The inspectors identified that the licensee's corrective actions were ineffective to correct the adverse trend in missing quality control inspection hold points. Pacific Gas and Electric Company entered this finding into the corrective action program as Notification 50135175.
The finding was more than minor because, if left uncorrected, the failure to perform inspections has the potential to lead to a more significant safety concern. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded that this finding was of very low safety significance because the uncorrected adverse trend did not represent a loss of system safety function, the loss of safety function of a single train for greater than its Technical Specification allowed outage time, actual loss of safety function of one or more non-Technical Specification trains greater than 24 hours, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating. The finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to thoroughly evaluate the adverse trend and take corrective actions that addressed the cause and extent of condition [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to provide adequate design control measures for verifying the emergency diesel generators meet the design basis. The inspectors requested to review the design control measures that Pacific Gas and Electric maintained to demonstrate compliance with General Design Criteria 17, "Electric Power Systems," design basis. The licensee was not able to retrieve the requested design control measures for the onsite electrical power systems. The licensee provided unit specific diesel loading calculations. The inspectors identified that the licensee failed to include all design basis accidents, a single limiting failure, consider bus frequency and voltage fluctuations, motor starting currents, or manually initiated loads in the calculation. In response to the inspectors' observations, the licensee performed an operability evaluation. The inspectors reviewed the evaluation and concluded that the

emergency diesel generators remained operable and capable of performing their intended safety function. The licensee has entered this issue into the corrective action program as Notification 50163396.
This finding is greater than minor because the design control attribute of the Mitigating Systems Cornerstone and the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences are affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality. The inspectors did not assign a crosscutting aspect because the finding represented a latent design issue. Pacific Gas and Electric revised the calculations in September 2006 and did not have a recent opportunity to identify this issue.

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct Violations of the Seismically-Induced Systems Interaction Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly implement housekeeping procedures to prevent seismically-induced system interactions. Specifically, the team identified two instances during a plant walk down where transient equipment was staged in the vicinity of safety-related equipment identified as seismically-induced system interaction targets. This transient equipment had not been analyzed to assess the risk to these safety-related components. Following identification by the team, licensee staff secured and analyzed the transient equipment. Licensee staff entered this finding into the corrective action program as Notifications 50084856 and 50084761.

The failure of plant personnel to follow the requirements to properly secure or analyze equipment in close proximity to sensitive equipment was a performance deficiency. The finding was more than minor because it was similar to Inspection Manual Chapter 0612, "Power Reactor Inspection Reports'" Appendix E, Example 3.j., in that it was indicative of a significant programmatic deficiency in the licensee's Seismically-Induced System Interactions Program that could lead to worse errors if uncorrected. Specifically, a change in program ownership in 2006 resulted in a degradation of the sensitivity of plant personnel to the risk of seismically-induced system interactions due to transient materials, insufficient training of plant personnel on the program, and an absence of quality records over an approximately two-year period. Using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance because it did not result in an actual loss of a system safety function, did not result in a loss of a single train of safety equipment for greater than its technical specification allowed outage time, did not involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a cross-cutting aspect in the area of human performance associated with the work practices area component because the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel failed to follow procedures [H.4(b)].

Inspection Report# : 2008008 *(pdf)*

**Significance:** G Jun 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Operability Procedure**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after the licensee failed to adhere to several requirements in Administrative Procedure OM7.ID12, "Operability Determination," Revision 11. Specifically, the licensee identified that it did not perform a prompt operability assessment for a condition adverse to quality until approximately 1 year after the immediate operability determination was performed. Also, the inspectors identified that when the prompt operability assessment was

performed, it relied inappropriately on engineering judgment, for a complex issue, without an adequately documented basis for that judgment. The adverse condition was an identified nonconformance related to the design basis because both units were operating at a full power average temperature less than the design value. The licensee has entered this into their corrective action program as Action Request A0723331 which details their planned correction actions.

The inspectors determined that the finding was more than minor because it is similar to Inspection Manual Chapter 0612, Appendix E, Minor Example 3(j) in that operability was questioned and both the licensee and the vendor had to perform significant work and analysis in order to fully address the operability impact of a low average temperature on operating Unit 1. In accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 4, Phase 1 - Initial Screening and Characterization of Findings, the inspectors concluded the finding was of very low safety significance (Green) because it did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding has a crosscutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions when it decided that engineering judgment alone was a sufficient basis for operability without a supporting plant specific analysis [H.1(b)] .

Inspection Report# : 2008003 *(pdf)*

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# Barrier Integrity

**Significance:** G Sep 30, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Procedure Resulting in Inoperable Auxiliary Building Ventilation System**
The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1, "Procedures," after Pacific Gas and Electric personnel failed to provide adequate work instructions for removal of equipment from service, resulting in the inoperability of both Unit 2 auxiliary building ventilation system trains, a condition prohibited by plant technical specifications. The work instruction did not provide a step for properly realigning the system to maintain operability of one train. The licensee entered this condition into the corrective actions system as Notification 50070612.

This finding was more than minor because the loss of both ventilation trains affected the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events and the inadequate procedure affected the attribute of procedure quality. The finding was of very low safety significance because it only represented a degradation of the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance with a Work Practices component because Pacific Gas & Electric staff failed to perform an adequate prejob brief to address questions regarding the sequence of steps and operators proceeded with the clearance in the face of uncertainty.

Inspection Report# : 2008004 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

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## Public Radiation Safety

**Significance: SL-IV** Dec 11, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report**
The team identified a non-cited violation of 10 CFR 50.71(e) for the failure of the licensee to periodically (every 24 months) update its Final Safety Analysis Report Update with all changes made in the facility or procedures. Specifically, in July 2005, the licensee stopped using the boric acid evaporator system as described in the Final Safety Analysis Report Update, Section 11.2.6, and did not submit an update to the NRC regarding this operational change. This issue was entered into the licensee's corrective action program as Notification 50116337 and licensee representatives stated an update would be submitted.

The team determined that the failure to update the Final Safety Analysis Report Update to reflect changes made to the facility was a performance deficiency. This issue is subject to traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. The finding is characterized as a Severity Level IV, non-cited violation in accordance with NRC Enforcement Policy, Supplement I, Example D.6, in that, the erroneous information in the Final Safety Analysis Report Update was not used to make an unacceptable change to the facility or procedures.
Inspection Report# : 2008009 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Take Appropriate Actions to Correct an Identified Adverse Trend**
The team identified a finding for failure to take adequate corrective actions to correct adverse trends in control of radioactive and potentially contaminated material as required by the corrective action program. Specifically, between May 2005 and June 2008, the licensee on two occasions identified and failed to correct adverse trends in the control of radioactive and potentially contaminated material. Licensee staff entered this finding into the corrective action program as Notification 50085121.

The finding was more than minor because it affected the Public Radiation Safety cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Using Inspection Manual Chapter 0609 Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to have very low safety significance because the dose impact to a member of the public was less than or equal to 0.005 rem total effective dose equivalent . The finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action area component; because the licensee failed to thoroughly evaluate problems such that the resolution addressed the cause [P.1(c)].

Inspection Report# : 2008008 *(pdf)*

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : June 05, 2009

# Diablo Canyon 2
# 2Q/2009 Plant Inspection Findings

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## Initiating Events

**Significance:** G Sep 30, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a Safety Assessment for Following Discovery of Explosive Gas in the Auxiliary and Containment Buildings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric personnel failed to perform a safety assessment prior to implementing a temporary procedure on July 20, 2008, to transfer an explosive gas mixture from the waste gas system to the Unit 2 vent. The explosive mixture of oxygen and hydrogen was discovered in the Unit 2 reactor coolant drain tank, waste gas surge tank, and interconnecting piping. The licensee also identified that the Unit 2 pressurizer relief tank vapor space exceeded the lower flammable limits. The explosive and flammable gas created a condition outside the plant design bases and was inconsistent with safety analysis. Plant Procedure TS3.ID2, “Licensing Basis Impact Evaluation,” required the licensee to have performed a safety assessment prior to conducting activities outside the design bases and inconsistent with safety analysis. The licensee entered this condition into the corrective actions system as Action Request A0741069.

This finding is greater than minor because explosive and flammable gas within the containment and auxiliary buildings affected the Initiating Events Cornerstone objective to limit the likelihood of events that may upset plant stability and challenge critical safety functions during power operations and protect against external factors such as fire and explosions. The inspectors used Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” to analyze the significances of the finding. The inspectors determined this finding was a fire prevention and administrative controls category due to the failure to meet the equipment control guideline for combustible gas flammability limits. The inspectors concluded that that this finding is of very low safety significance because the condition represented a low degradation rating due to the lack of a direct ignition source. This finding has a crosscutting aspect in human performance in the area of Decision Making because the licensee failed to use the systematic process provided in Procedure TS3.ID when making a safety significant or risk-significant decision when faced with the unexpected explosive gas mixture within containment and auxiliary building plant systems.
Inspection Report# : 2008004 *(pdf)*

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## Mitigating Systems

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition Prohibited by the Plant's Technical Specifications**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a condition prohibited by technical specifications. The licensee failed to correctly evaluate the March 18, 2009, failure of the Unit 2 control rod demand position indicators for reportability. The inspectors concluded that the failure of control rod position indicators was a condition prohibited by Technical Specification 3.17, “Rod Position Indication.”

This finding is greater than minor because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the regulations in order to perform its regulatory function. This finding affected the mitigating systems cornerstone. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV, noncited violation. The licensee entered this issue into the corrective action program as Notification 50242153. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the Unit 2 control rod demand position indicators for reportability [P.1(c)]

Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the FSARU with Current Plant Design Criteria**
The inspectors identified a noncited violation of 10 CFR Part 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with current plant design criteria. The Final Safety Analysis Report Update stated that Diablo Canyon was designed to comply with the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, published in July 1967. The inspectors identified that the Diablo Canyon Safety Evaluation Report stated that the NRC used General Design Criteria published in July 1971 to review the plant design. In addition, during the initial licensing process, the licensee stated that the plant was evaluated against the 1971 design criteria during the licensing process.

The inspectors evaluated this finding using the traditional enforcement process because the failure to update the Final Safety Analysis Report affected the NRC's ability to perform its regulatory function. The inspectors concluded that the failure to update the Final Safety Analysis Report was a Severity Level IV violation based on the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I – Reactor Operations, dated January 14, 2005, because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner [P.1(d)]

Inspection Report# : 2009003 *(pdf)*

**Significance:** G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following the Loss of Design Control for the 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, "Corrective Action," after Pacific Gas and Electric failed to adequately correct a nonconforming condition related to the adequacy of design documentation to demonstrate the acceptability of design control for the 500 kV delayed offsite power system. The licensee stated the design control documentation demonstrated that the offsite power system met the design basis was not retrievable. The licensee entered this nonconforming condition into the corrective action system. On October 28, 2008, plant engineers completed an evaluation of the nonconforming condition and concluded the delayed offsite power system design basis was demonstrated by a "road map" of pre-existing analyses created to support other plant functions. The inspectors concluded that the "road map" was less than adequate because the licensee failed to consider the affect of the loss of reactor coolant pump seal cooling and injection anticipated during the time needed to align the offsite power supply to the engineering safety feature buses. The inspectors concluded that the failure of the licensee to promptly correct the nonconforming condition and ensure that the "road map" implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance.

This finding is more than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors concluded this finding is of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition to ensure that the offsite power system design basis was met [P.1(c)](

Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility Associated with the 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a thermal hydraulic analysis to determine if prior NRC approval was required for a 30-minute delay time to align offsite power. This analysis, Calculation STA 274, "RETRAN Evaluation of GDC 17 Loss of AC Scenario," Revision 0, demonstrated that the 30-minute delayed offsite power source was acceptable. On December 31, 2008, a Pacific Gas and Electric 10 CFR 50.59 screen concluded that Calculation STA 274 was not required to be evaluated to determine if prior NRC approval was required for the delay time. On March 31, 2009, the inspectors concluded that the licensee was required to evaluate Calculation STA 274 to determine if prior NRC approval was needed. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required for the 30-minute delay time to align offsite power.

The inspectors concluded that the finding is more than minor because the changes made to the facility required prior NRC review and approval. The finding affected the Mitigating Systems Cornerstone because the change described how the delayed offsite power source met the design basis. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency that did not result in the loss of operability or functionality. Because the issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated this finding using the traditional enforcement process. This issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1 (c)]

Inspection Report# : 2009003 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Operation of 230 kV Offsite Power System Outside the Design Basis**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria III, Design Control, after Pacific Gas and Electric failed to adequately translate the design basis for the 230 kV preferred offsite power system into specifications and procedures. Between November 3 and 7, 2008, the licensee operated with both units aligned to a single startup transformer. This created a situation where a dual unit trip or trip on one unit and accident on the other unit could result in loss of the preferred immediate offsite power source offsite power to both units.
The finding is greater than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded that the finding is a design deficiency that did result in loss of operability. However, the inspectors concluded the finding is of very low safety significance because the actual loss of safety function of the 230 kV offsite power system was less than the Technical Specification allowed outage time. The inspectors also concluded that the finding did not represent a loss of safety function for greater than 24 hours or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors determined that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the operability of the offsite power circuit prior to removing the Unit 2 startup transformer from service [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Implement Effective Actions to Correct an Adverse Trend**
The inspectors identified a finding after Pacific Gas and Electric was ineffective in addressing an adverse trend in missed quality control inspection hold points. Licensee Procedure OM7, "Corrective Action Program," required that the licensee evaluate problems commensurate with their significance, determine the cause, and conduct a proper evaluation and resolution of repeat occurrences. The procedure further required that corrective actions are completed in a timely manner consistent with the problem significance. On May 19, 2007, Pacific Gas and Electric identified an adverse trend of missing quality control inspection hold points and requested that an apparent cause evaluation be performed. On July 11, 2007, this adverse trend was also evaluated by the Quality Verification Department as part of an assessment of Refueling Outage 14 maintenance. In March 2008, the licensee completed the evaluations and corrective actions. During the subsequent Unit 2 refueling outage, the Quality Verification Department identified over 11 additional missing quality inspection hold points. The inspectors identified that the licensee's corrective actions were ineffective to correct the adverse trend in missing quality control inspection hold points. Pacific Gas and Electric Company entered this finding into the corrective action program as Notification 50135175.
The finding was more than minor because, if left uncorrected, the failure to perform inspections has the potential to lead to a more significant safety concern. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded that this finding was of very low safety significance because the uncorrected adverse trend did not represent a loss of system safety function, the loss of safety function of a single train for greater than its Technical Specification allowed outage time, actual loss of safety function of one or more non-Technical Specification trains greater than 24 hours, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating. The finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to thoroughly evaluate the adverse trend and take corrective actions that addressed the cause and extent of condition [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to provide adequate design control measures for verifying the emergency diesel generators meet the design basis. The inspectors requested to review the design control measures that Pacific Gas and Electric maintained to demonstrate compliance with General Design Criteria 17, "Electric Power Systems," design basis. The licensee was not able to retrieve the requested design control measures for the onsite electrical power systems. The licensee provided unit specific diesel loading calculations. The inspectors identified that the licensee failed to include all design basis accidents, a single limiting failure, consider bus frequency and voltage fluctuations, motor starting currents, or manually initiated loads in the calculation. In response to the inspectors' observations, the licensee performed an operability evaluation. The inspectors reviewed the evaluation and concluded that the emergency diesel generators remained operable and capable of performing their intended safety function. The licensee has entered this issue into the corrective action program as Notification 50163396.
This finding is greater than minor because the design control attribute of the Mitigating Systems Cornerstone and the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences are affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the significance of this finding. The inspectors concluded the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality. The inspectors did not assign a crosscutting aspect because the finding represented a latent design issue. Pacific Gas and Electric revised the calculations in September 2006 and did not have a recent opportunity to identify this issue.

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct Violations of the Seismically-Induced Systems Interaction Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly implement housekeeping procedures to prevent seismically-induced system interactions. Specifically, the team identified two instances during a plant walk down where transient equipment was staged in the vicinity of safety-related equipment identified as seismically-induced system interaction targets. This transient equipment had not been analyzed to assess the risk to these safety-related components. Following identification by the team, licensee staff secured and analyzed the transient equipment. Licensee staff entered this finding into the corrective action program as Notifications 50084856 and 50084761.

The failure of plant personnel to follow the requirements to properly secure or analyze equipment in close proximity to sensitive equipment was a performance deficiency. The finding was more than minor because it was similar to Inspection Manual Chapter 0612, "Power Reactor Inspection Reports'" Appendix E, Example 3.j., in that it was indicative of a significant programmatic deficiency in the licensee's Seismically-Induced System Interactions Program that could lead to worse errors if uncorrected. Specifically, a change in program ownership in 2006 resulted in a degradation of the sensitivity of plant personnel to the risk of seismically-induced system interactions due to transient materials, insufficient training of plant personnel on the program, and an absence of quality records over an approximately two-year period. Using Inspection Manual Chapter 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," the finding was determined to have very low safety significance because it did not result in an actual loss of a system safety function, did not result in a loss of a single train of safety equipment for greater than its technical specification allowed outage time, did not involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a cross-cutting aspect in the area of human performance associated with the work practices area component because the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel failed to follow procedures [H.4(b)].

Inspection Report# : 2008008 *(pdf)*

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## Barrier Integrity

**Significance:** G Sep 30, 2008
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Procedure Resulting in Inoperable Auxiliary Building Ventilation System**
The inspectors reviewed a self-revealing noncited violation of Technical Specification 5.4.1, "Procedures," after Pacific Gas and Electric personnel failed to provide adequate work instructions for removal of equipment from service, resulting in the inoperability of both Unit 2 auxiliary building ventilation system trains, a condition prohibited by plant technical specifications. The work instruction did not provide a step for properly realigning the system to maintain operability of one train. The licensee entered this condition into the corrective actions system as Notification 50070612.

This finding was more than minor because the loss of both ventilation trains affected the Barrier Integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events and the inadequate procedure affected the attribute of procedure quality. The finding was of very low safety significance because it only represented a degradation of the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance with a Work Practices component because Pacific Gas & Electric staff failed to perform an adequate prejob brief to address questions regarding the sequence of steps and operators proceeded with the clearance in the face of uncertainty.

Inspection Report# : 2008004 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

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# Public Radiation Safety

**Significance: SL-IV** Dec 11, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report**
The team identified a non-cited violation of 10 CFR 50.71(e) for the failure of the licensee to periodically (every 24 months) update its Final Safety Analysis Report Update with all changes made in the facility or procedures. Specifically, in July 2005, the licensee stopped using the boric acid evaporator system as described in the Final Safety Analysis Report Update, Section 11.2.6, and did not submit an update to the NRC regarding this operational change. This issue was entered into the licensee's corrective action program as Notification 50116337 and licensee representatives stated an update would be submitted.

The team determined that the failure to update the Final Safety Analysis Report Update to reflect changes made to the facility was a performance deficiency. This issue is subject to traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. The finding is characterized as a Severity Level IV, non-cited violation in accordance with NRC Enforcement Policy, Supplement I, Example D.6, in that, the erroneous information in the Final Safety Analysis Report Update was not used to make an unacceptable change to the facility or procedures.
Inspection Report# : 2008009 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Take Appropriate Actions to Correct an Identified Adverse Trend**
The team identified a finding for failure to take adequate corrective actions to correct adverse trends in control of radioactive and potentially contaminated material as required by the corrective action program. Specifically, between May 2005 and June 2008, the licensee on two occasions identified and failed to correct adverse trends in the control of radioactive and potentially contaminated material. Licensee staff entered this finding into the corrective action program as Notification 50085121.

The finding was more than minor because it affected the Public Radiation Safety cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Using Inspection Manual Chapter 0609 Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to have very low safety significance because the dose impact to a member of the public was less than or equal to 0.005 rem total effective dose equivalent . The finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action area component; because the licensee failed to thoroughly evaluate problems such that the resolution addressed the cause [P.1(c)].

Inspection Report# : 2008008 *(pdf)*

## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

## Miscellaneous

Last modified : August 31, 2009

# Diablo Canyon 2
# 3Q/2009 Plant Inspection Findings

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## Initiating Events

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: FIN Finding
**Failure to Perform Corrective Actions Resulted in an Unplanned Trip**
A self-revealing finding was identified after Pacific Gas and Electric failed to implement planned corrective actions resulting in the loss of cooling to a main transformer, a rapid shutdown and a manual reactor trip of Unit 2. On June 30, 2009, cooling to a main transformer was lost because a fuse opened in the 480 volt power circuit due to loose terminal connections in the cooling control panel. Plant operators rapidly shut-down the unit from full power after transformer cooling was lost. A previous failure of transformer cooling due to loose terminal connections occurred on Unit 1, also resulting in a reactor trip. Corrective actions to prevent recurrence following the previous event included replacement of the main transformer terminations in the cooling control panels. Review of the work orders revealed that these corrective actions were not completed and the work documents were closed. While the failure to complete the corrective actions was a latent issue, the inspectors concluded that the licensee had a recent opportunity to identify the issue. Plant technicians implemented thermograph monitoring of main transformer cooling circuits and identified hot 480 volt power terminations in the Unit 2 main transformer cooling disconnect box in April 2009. These hot terminations should have prompted Pacific Gas and Electric to review internal operating experience related to main transformer cooling issues. The licensee entered this finding into corrective program as Notification 50260721.
The inspectors concluded that the finding is greater than minor because it is associated with the equipment performance attribute of the initiating events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that interrupt plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined the finding to have very low safety significance because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigation equipment or functions. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the operating experience component because Pacific Gas and Electric failed to perform an adequate internal operating experience review following the discovery of hot terminations on Unit 2 main transformer in April 2009.

Inspection Report# : 2009004 *(pdf)*

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## Mitigating Systems

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct a Degraded Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain Fire Door 155 in the rated condition. On September 1, 2009, the inspectors identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. Fire Door 155 was required to provide a 1½ hour rated barrier between Fire Areas 4B and S 2. The licensee re-engaged the latching mechanism and entered the condition into the corrective action program as Notification 50265691. On September 16, 2009, the inspectors again identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. The licensee subsequently determined that the latching mechanism had been defective. The inspectors concluded the most significant contributor to the violation was the less than adequate corrective action taken by the licensee following identification of the problem on September 1, 2009.

This finding is more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door is a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated barrier function. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water-based fire suppression system was in place in the exposing fire area. The licensee entered this violation into the corrective action program as Notification 50268494. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the degraded fire door such that the resolution address causes and extent of condition.

Inspection Report# : 2009004 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Emergency Operating Procedures Following a Reactor Trip**
The inspectors identified a noncited violation of Technical Specification 5.4.1.b, “Emergency Operating Procedures,” after plant operators failed to enter Emergency Operating Procedures E 0, “Reactor Trip or Safety Injection,” and E 0.1, “Reactor Trip Response,” following a Unit 2 reactor trip on June 30, 2009. Plant operators initiated a rapid load reduction from full power following loss of cooling to a main transformer bank. Plant operators manually tripped the reactor at about two percent power and proceeded to the procedure for placing the unit in cold shutdown. Plant operators did not perform the required steps in Emergency Operating Procedures E 0 and E 0.1 following the reactor trip. The inspectors concluded that the most significant contributor to the violation was less than adequate direction in the procedure used for rapid load reduction. The licensee entered this violation into the corrective action program as Notification 50262363.
The finding is greater than minor because the failure of operations personnel to implement emergency operator procedures was associated with the mitigating systems cornerstone human performance attribute to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded the significance of this finding is of very low safety significance because the finding was not a design or qualification deficiency, did not result in loss of equipment operability or functionality, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had a crosscutting aspect in the area of human performance associated with the resource component because Pacific Gas and Electric did not have a complete rapid load reduction procedure.
Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with Current Accident Analysis**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with a critical operator action assumed in the plant steam generator tube rupture accident analysis. The steam generator tube rupture accident analysis assumed that the ruptured steam generator will not overfill with water during the accident. To ensure a margin to overfill, the accident analysis included a critical assumption that plant operators would manually trip the turbine-driven auxiliary feedwater pump within 5.54 minutes following the reactor trip. Final Safety Analysis Report Update Section 15.4.3.1, “Identification of Causes and Accident Description,” and Final Safety Analysis Report Update Table 15.4 12, “Operator Action Times for Design Basis SGTR Analysis,” provided a detailed description of the time dependant operator actions assumed in the accident analysis. The inspectors identified that neither section included the critical assumed operator action to trip the turbine-driven auxiliary feedwater pump. The inspectors concluded that the licensee had a reasonable opportunity to identify and correct the problem when the results of the revised steam generator tube rupture accident, supporting steam generator replacement, was updated in the Final Safety Analysis Report Update in October 2008. The licensee entered this violation into the corrective action program as Notification 50269753.
The inspectors evaluated this finding with the traditional enforcement process because the issue affected the NRC’s ability to perform its regulatory function. The inspectors concluded that the finding is greater than minor because the failure to update the required critical operator action assumed in the accident analysis could have a material impact on safety or licensed activities. The inspectors concluded that the violation is Severity Level IV because the erroneous

information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement a corrective action program with a low threshold for identifying issues and failed to identify the inaccuracies in the accident analysis as described in the Final Safety Analysis Report Update.
Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition Prohibited by the Plant's Technical Specifications**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a condition prohibited by technical specifications. The licensee failed to correctly evaluate the March 18, 2009, failure of the Unit 2 control rod demand position indicators for reportability. The inspectors concluded that the failure of control rod position indicators was a condition prohibited by Technical Specification 3.17, “Rod Position Indication.”

This finding is greater than minor because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the regulations in order to perform its regulatory function. This finding affected the mitigating systems cornerstone. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV, noncited violation. The licensee entered this issue into the corrective action program as Notification 50242153. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the Unit 2 control rod demand position indicators for reportability [P.1(c)]

Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the FSARU with Current Plant Design Criteria**
The inspectors identified a noncited violation of 10 CFR Part 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with current plant design criteria. The Final Safety Analysis Report Update stated that Diablo Canyon was designed to comply with the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, published in July 1967. The inspectors identified that the Diablo Canyon Safety Evaluation Report stated that the NRC used General Design Criteria published in July 1971 to review the plant design. In addition, during the initial licensing process, the licensee stated that the plant was evaluated against the 1971 design criteria during the licensing process.

The inspectors evaluated this finding using the traditional enforcement process because the failure to update the Final Safety Analysis Report affected the NRC’s ability to perform its regulatory function. The inspectors concluded that the failure to update the Final Safety Analysis Report was a Severity Level IV violation based on the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I – Reactor Operations, dated January 14, 2005, because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner [P.1(d)]

Inspection Report# : 2009003 *(pdf)*

**Significance:** G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following the Loss of Design Control for the 500 kV Offsite Power Source**

The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, “Corrective Action,” after Pacific Gas and Electric failed to adequately correct a nonconforming condition related to the adequacy of design documentation to demonstrate the acceptability of design control for the 500 kV delayed offsite power system. The licensee stated the design control documentation demonstrated that the offsite power system met the design basis was not retrievable. The licensee entered this nonconforming condition into the corrective action system. On October 28, 2008, plant engineers completed an evaluation of the nonconforming condition and concluded the delayed offsite power system design basis was demonstrated by a “road map” of pre-existing analyses created to support other plant functions. The inspectors concluded that the “road map” was less than adequate because the licensee failed to consider the affect of the loss of reactor coolant pump seal cooling and injection anticipated during the time needed to align the offsite power supply to the engineering safety feature buses. The inspectors concluded that the failure of the licensee to promptly correct the nonconforming condition and ensure that the “road map” implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance.

This finding is more than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors concluded this finding is of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition to ensure that the offsite power system design basis was met [P.1(c)](

Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility Associated with the 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a thermal hydraulic analysis to determine if prior NRC approval was required for a 30-minute delay time to align offsite power. This analysis, Calculation STA 274, “RETRAN Evaluation of GDC 17 Loss of AC Scenario,” Revision 0, demonstrated that the 30-minute delayed offsite power source was acceptable. On December 31, 2008, a Pacific Gas and Electric 10 CFR 50.59 screen concluded that Calculation STA 274 was not required to be evaluated to determine if prior NRC approval was required for the delay time. On March 31, 2009, the inspectors concluded that the licensee was required to evaluate Calculation STA 274 to determine if prior NRC approval was needed. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required for the 30-minute delay time to align offsite power.

The inspectors concluded that the finding is more than minor because the changes made to the facility required prior NRC review and approval. The finding affected the Mitigating Systems Cornerstone because the change described how the delayed offsite power source met the design basis. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency that did not result in the loss of operability or functionality. Because the issue affected the NRC’s ability to perform its regulatory function, the inspectors evaluated this finding using the traditional enforcement process. This issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1 (c)]

Inspection Report# : 2009003 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Operation of 230 kV Offsite Power System Outside the Design Basis**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria III, Design Control, after

Pacific Gas and Electric failed to adequately translate the design basis for the 230 kV preferred offsite power system into specifications and procedures. Between November 3 and 7, 2008, the licensee operated with both units aligned to a single startup transformer. This created a situation where a dual unit trip or trip on one unit and accident on the other unit could result in loss of the preferred immediate offsite power source offsite power to both units.
The finding is greater than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the significance of this finding. The inspectors concluded that the finding is a design deficiency that did result in loss of operability. However, the inspectors concluded the finding is of very low safety significance because the actual loss of safety function of the 230 kV offsite power system was less than the Technical Specification allowed outage time. The inspectors also concluded that the finding did not represent a loss of safety function for greater than 24 hours or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors determined that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the operability of the offsite power circuit prior to removing the Unit 2 startup transformer from service [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Implement Effective Actions to Correct an Adverse Trend**
The inspectors identified a finding after Pacific Gas and Electric was ineffective in addressing an adverse trend in missed quality control inspection hold points. Licensee Procedure OM7, “Corrective Action Program,” required that the licensee evaluate problems commensurate with their significance, determine the cause, and conduct a proper evaluation and resolution of repeat occurrences. The procedure further required that corrective actions are completed in a timely manner consistent with the problem significance. On May 19, 2007, Pacific Gas and Electric identified an adverse trend of missing quality control inspection hold points and requested that an apparent cause evaluation be performed. On July 11, 2007, this adverse trend was also evaluated by the Quality Verification Department as part of an assessment of Refueling Outage 14 maintenance. In March 2008, the licensee completed the evaluations and corrective actions. During the subsequent Unit 2 refueling outage, the Quality Verification Department identified over 11 additional missing quality inspection hold points. The inspectors identified that the licensee’s corrective actions were ineffective to correct the adverse trend in missing quality control inspection hold points. Pacific Gas and Electric Company entered this finding into the corrective action program as Notification 50135175.
The finding was more than minor because, if left uncorrected, the failure to perform inspections has the potential to lead to a more significant safety concern. The inspectors used Inspection Manual Chapter 0609, Appendix A, “Determining the Significance of reactor Inspection Findings for At-Power Situations,” to analyze the significance of this finding. The inspectors concluded that this finding was of very low safety significance because the uncorrected adverse trend did not represent a loss of system safety function, the loss of safety function of a single train for greater than its Technical Specification allowed outage time, actual loss of safety function of one or more non-Technical Specification trains greater than 24 hours, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating. The finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to thoroughly evaluate the adverse trend and take corrective actions that addressed the cause and extent of condition [P.1(c)].

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Dec 31, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to provide adequate design control measures for verifying the emergency diesel generators meet the design basis. The inspectors requested to review the design control measures that Pacific Gas and

Electric maintained to demonstrate compliance with General Design Criteria 17, “Electric Power Systems,” design basis. The licensee was not able to retrieve the requested design control measures for the onsite electrical power systems. The licensee provided unit specific diesel loading calculations. The inspectors identified that the licensee failed to include all design basis accidents, a single limiting failure, consider bus frequency and voltage fluctuations, motor starting currents, or manually initiated loads in the calculation. In response to the inspectors' observations, the licensee performed an operability evaluation. The inspectors reviewed the evaluation and concluded that the emergency diesel generators remained operable and capable of performing their intended safety function. The licensee has entered this issue into the corrective action program as Notification 50163396.
This finding is greater than minor because the design control attribute of the Mitigating Systems Cornerstone and the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences are affected. The inspectors used Inspection Manual Chapter 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the significance of this finding. The inspectors concluded the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality. The inspectors did not assign a crosscutting aspect because the finding represented a latent design issue. Pacific Gas and Electric revised the calculations in September 2006 and did not have a recent opportunity to identify this issue.

Inspection Report# : 2008005 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct Violations of the Seismically-Induced Systems Interaction Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the failure to properly implement housekeeping procedures to prevent seismically-induced system interactions. Specifically, the team identified two instances during a plant walk down where transient equipment was staged in the vicinity of safety-related equipment identified as seismically-induced system interaction targets. This transient equipment had not been analyzed to assess the risk to these safety-related components. Following identification by the team, licensee staff secured and analyzed the transient equipment. Licensee staff entered this finding into the corrective action program as Notifications 50084856 and 50084761.

The failure of plant personnel to follow the requirements to properly secure or analyze equipment in close proximity to sensitive equipment was a performance deficiency. The finding was more than minor because it was similar to Inspection Manual Chapter 0612, “Power Reactor Inspection Reports'” Appendix E, Example 3.j., in that it was indicative of a significant programmatic deficiency in the licensee's Seismically-Induced System Interactions Program that could lead to worse errors if uncorrected. Specifically, a change in program ownership in 2006 resulted in a degradation of the sensitivity of plant personnel to the risk of seismically-induced system interactions due to transient materials, insufficient training of plant personnel on the program, and an absence of quality records over an approximately two-year period. Using Inspection Manual Chapter 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” the finding was determined to have very low safety significance because it did not result in an actual loss of a system safety function, did not result in a loss of a single train of safety equipment for greater than its technical specification allowed outage time, did not involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a cross-cutting aspect in the area of human performance associated with the work practices area component because the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel failed to follow procedures [H.4(b)].

Inspection Report# : 2008008 *(pdf)*

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## Barrier Integrity

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# Emergency Preparedness

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# Occupational Radiation Safety

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# Public Radiation Safety

**Significance: SL-IV** Dec 11, 2008
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report**
The team identified a non-cited violation of 10 CFR 50.71(e) for the failure of the licensee to periodically (every 24 months) update its Final Safety Analysis Report Update with all changes made in the facility or procedures. Specifically, in July 2005, the licensee stopped using the boric acid evaporator system as described in the Final Safety Analysis Report Update, Section 11.2.6, and did not submit an update to the NRC regarding this operational change. This issue was entered into the licensee's corrective action program as Notification 50116337 and licensee representatives stated an update would be submitted.

The team determined that the failure to update the Final Safety Analysis Report Update to reflect changes made to the facility was a performance deficiency. This issue is subject to traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. The finding is characterized as a Severity Level IV, non-cited violation in accordance with NRC Enforcement Policy, Supplement I, Example D.6, in that, the erroneous information in the Final Safety Analysis Report Update was not used to make an unacceptable change to the facility or procedures.
Inspection Report# : 2008009 *(pdf)*

**Significance:** G Nov 20, 2008
Identified By: NRC
Item Type: FIN Finding
**Failure to Take Appropriate Actions to Correct an Identified Adverse Trend**
The team identified a finding for failure to take adequate corrective actions to correct adverse trends in control of radioactive and potentially contaminated material as required by the corrective action program. Specifically, between May 2005 and June 2008, the licensee on two occasions identified and failed to correct adverse trends in the control of radioactive and potentially contaminated material. Licensee staff entered this finding into the corrective action program as Notification 50085121.

The finding was more than minor because it affected the Public Radiation Safety cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive materials released into the public domain as a result of routine civilian nuclear reactor operation. Using Inspection Manual Chapter 0609 Appendix D, "Public Radiation Safety Significance Determination Process," the finding was determined to have very low safety significance because the dose impact to a member of the public was less than or equal to 0.005 rem total effective dose equivalent . The finding has a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action area component; because the licensee failed to thoroughly evaluate problems such that the resolution addressed the cause [P.1(c)].

Inspection Report# : 2008008 *(pdf)*

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# Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : December 10, 2009

# Diablo Canyon 2
# 4Q/2009 Plant Inspection Findings

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## Initiating Events

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: FIN Finding
**Failure to Perform Corrective Actions Resulted in an Unplanned Trip**
A self-revealing finding was identified after Pacific Gas and Electric failed to implement planned corrective actions resulting in the loss of cooling to a main transformer, a rapid shutdown and a manual reactor trip of Unit 2. On June 30, 2009, cooling to a main transformer was lost because a fuse opened in the 480 volt power circuit due to loose terminal connections in the cooling control panel. Plant operators rapidly shut-down the unit from full power after transformer cooling was lost. A previous failure of transformer cooling due to loose terminal connections occurred on Unit 1, also resulting in a reactor trip. Corrective actions to prevent recurrence following the previous event included replacement of the main transformer terminations in the cooling control panels. Review of the work orders revealed that these corrective actions were not completed and the work documents were closed. While the failure to complete the corrective actions was a latent issue, the inspectors concluded that the licensee had a recent opportunity to identify the issue. Plant technicians implemented thermograph monitoring of main transformer cooling circuits and identified hot 480 volt power terminations in the Unit 2 main transformer cooling disconnect box in April 2009. These hot terminations should have prompted Pacific Gas and Electric to review internal operating experience related to main transformer cooling issues. The licensee entered this finding into corrective program as Notification 50260721.
The inspectors concluded that the finding is greater than minor because it is associated with the equipment performance attribute of the initiating events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that interrupt plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined the finding to have very low safety significance because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigation equipment or functions. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the operating experience component because Pacific Gas and Electric failed to perform an adequate internal operating experience review following the discovery of hot terminations on Unit 2 main transformer in April 2009.

Inspection Report# : 2009004 *(pdf)*

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## Mitigating Systems

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct a Degraded Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition (5), "Fire Protection," after Pacific Gas and Electric failed to maintain Fire Door 155 in the rated condition. On September 1, 2009, the inspectors identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. Fire Door 155 was required to provide a 1½ hour rated barrier between Fire Areas 4B and S 2. The licensee re-engaged the latching mechanism and entered the condition into the corrective action program as Notification 50265691. On September 16, 2009, the inspectors again identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. The licensee subsequently determined that the latching mechanism had been defective. The inspectors concluded the most significant contributor to the violation was the less than adequate corrective action taken by the licensee following identification of the problem on September 1, 2009.

This finding is more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door is a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated barrier function. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water-based fire suppression system was in place in the exposing fire area. The licensee entered this violation into the corrective action program as Notification 50268494. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the degraded fire door such that the resolution address causes and extent of condition.

Inspection Report# : 2009004 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Emergency Operating Procedures Following a Reactor Trip**
The inspectors identified a noncited violation of Technical Specification 5.4.1.b, "Emergency Operating Procedures," after plant operators failed to enter Emergency Operating Procedures E 0, "Reactor Trip or Safety Injection," and E 0.1, "Reactor Trip Response," following a Unit 2 reactor trip on June 30, 2009. Plant operators initiated a rapid load reduction from full power following loss of cooling to a main transformer bank. Plant operators manually tripped the reactor at about two percent power and proceeded to the procedure for placing the unit in cold shutdown. Plant operators did not perform the required steps in Emergency Operating Procedures E 0 and E 0.1 following the reactor trip. The inspectors concluded that the most significant contributor to the violation was less than adequate direction in the procedure used for rapid load reduction. The licensee entered this violation into the corrective action program as Notification 50262363.
The finding is greater than minor because the failure of operations personnel to implement emergency operator procedures was associated with the mitigating systems cornerstone human performance attribute to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded the significance of this finding is of very low safety significance because the finding was not a design or qualification deficiency, did not result in loss of equipment operability or functionality, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had a crosscutting aspect in the area of human performance associated with the resource component because Pacific Gas and Electric did not have a complete rapid load reduction procedure.
Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with Current Accident Analysis**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with a critical operator action assumed in the plant steam generator tube rupture accident analysis. The steam generator tube rupture accident analysis assumed that the ruptured steam generator will not overfill with water during the accident. To ensure a margin to overfill, the accident analysis included a critical assumption that plant operators would manually trip the turbine-driven auxiliary feedwater pump within 5.54 minutes following the reactor trip. Final Safety Analysis Report Update Section 15.4.3.1, "Identification of Causes and Accident Description," and Final Safety Analysis Report Update Table 15.4 12, "Operator Action Times for Design Basis SGTR Analysis," provided a detailed description of the time dependant operator actions assumed in the accident analysis. The inspectors identified that neither section included the critical assumed operator action to trip the turbine-driven auxiliary feedwater pump. The inspectors concluded that the licensee had a reasonable opportunity to identify and correct the problem when the results of the revised steam generator tube rupture accident, supporting steam generator replacement, was updated in the Final Safety Analysis Report Update in October 2008. The licensee entered this violation into the corrective action program as Notification 50269753.
The inspectors evaluated this finding with the traditional enforcement process because the issue affected the NRC's ability to perform its regulatory function. The inspectors concluded that the finding is greater than minor because the failure to update the required critical operator action assumed in the accident analysis could have a material impact on safety or licensed activities. The inspectors concluded that the violation is Severity Level IV because the erroneous

information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement a corrective action program with a low threshold for identifying issues and failed to identify the inaccuracies in the accident analysis as described in the Final Safety Analysis Report Update.
Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit a Licensee Event Report for a Condition Prohibited by the Plant's Technical Specifications**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a condition prohibited by technical specifications. The licensee failed to correctly evaluate the March 18, 2009, failure of the Unit 2 control rod demand position indicators for reportability. The inspectors concluded that the failure of control rod position indicators was a condition prohibited by Technical Specification 3.17, “Rod Position Indication.”

This finding is greater than minor because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the regulations in order to perform its regulatory function. This finding affected the mitigating systems cornerstone. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV, noncited violation. The licensee entered this issue into the corrective action program as Notification 50242153. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the Unit 2 control rod demand position indicators for reportability.
Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the FSAR update with Current Plant Design Criteria**
The inspectors identified a noncited violation of 10 CFR Part 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with current plant design criteria. The Final Safety Analysis Report Update stated that Diablo Canyon was designed to comply with the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, published in July 1967. The inspectors identified that the Diablo Canyon Safety Evaluation Report stated that the NRC used General Design Criteria published in July 1971 to review the plant design. In addition, during the initial licensing process, the licensee stated that the plant was evaluated against the 1971 design criteria during the licensing process.

The inspectors evaluated this finding using the traditional enforcement process because the failure to update the Final Safety Analysis Report affected the NRC’s ability to perform its regulatory function. The inspectors concluded that the failure to update the Final Safety Analysis Report was a Severity Level IV violation based on the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I – Reactor Operations, dated January 14, 2005, because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner.
Inspection Report# : 2009003 *(pdf)*

**Significance:** G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following the Loss of Design Control for the 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, “Corrective Action,” after Pacific Gas and Electric failed to adequately correct a nonconforming condition related to the adequacy of design

documentation to demonstrate the acceptability of design control for the 500 kV delayed offsite power system. The licensee stated the design control documentation demonstrated that the offsite power system met the design basis was not retrievable. The licensee entered this nonconforming condition into the corrective action system. On October 28, 2008, plant engineers completed an evaluation of the nonconforming condition and concluded the delayed offsite power system design basis was demonstrated by a "road map" of pre-existing analyses created to support other plant functions. The inspectors concluded that the "road map" was less than adequate because the licensee failed to consider the affect of the loss of reactor coolant pump seal cooling and injection anticipated during the time needed to align the offsite power supply to the engineering safety feature buses. The inspectors concluded that the failure of the licensee to promptly correct the nonconforming condition and ensure that the "road map" implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance.

This finding is more than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors concluded this finding is of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition to ensure that the offsite power system design basis was met.
Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility as Described in the Final Safety Analysis Report Update Associated with 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a thermal hydraulic analysis to determine if prior NRC approval was required for a 30-minute delay time to align offsite power. This analysis, Calculation STA 274, "RETRAN Evaluation of GDC 17 Loss of AC Scenario," Revision 0, demonstrated that the 30-minute delayed offsite power source was acceptable. On December 31, 2008, a Pacific Gas and Electric 10 CFR 50.59 screen concluded that Calculation STA 274 was not required to be evaluated to determine if prior NRC approval was required for the delay time. On March 31, 2009, the inspectors concluded that the licensee was required to evaluate Calculation STA 274 to determine if prior NRC approval was needed. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required for the 30-minute delay time to align offsite power.

The inspectors concluded that the finding is more than minor because the changes made to the facility required prior NRC review and approval. The finding affected the Mitigating Systems Cornerstone because the change described how the delayed offsite power source met the design basis. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency that did not result in the loss of operability or functionality. Because the issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated this finding using the traditional enforcement process. This issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required.
Inspection Report# : 2009003 *(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : March 01, 2010

# Diablo Canyon 2
# 1Q/2010 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Replacement Reactor Head Modification Design Control**
The inspectors identified a noncited violation of Title 10 CFR, Part 50, Appendix B, Criteria III, "Design Control," after the design contractor failed to perform adequately calculations demonstrating that the replacement reactor head met ASME Code acceptance criteria. The contractor failed to use the critical seismic damping values specified in the plant design basis for the design of the integrated head assembly and the control rod drive mechanism housing assembly and when calculating component stress during a postulated design basis earthquakes. The licensee entered this condition into the corrective action program as Notifications 50276107 and 50276288.

The inspectors concluded that the failure to properly implement the plant design basis in the replacement head design was a performance deficiency. The finding is more than minor because the performance deficiency is associated with the Initiating Events Cornerstone design control attribute and adversely affected the cornerstone objective to limit the likelihood of loss of a coolant accident during a seismic event. The inspectors determined the finding is of very low safety significance because assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for reactor coolant system leakage nor have likely affected other mitigation systems resulting in a total loss of their safety function. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not identify the use of improper damping values with a low threshold for identifying issues during oversight of contractor activities and design reviews [P.1(a)].

Inspection Report# : 2009005 *(pdf)*

**Significance: SL-IV** Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Lees Than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after the licensee failed to perform an adequate evaluation to demonstrate that prior NRC approval was not required before making changes to the facility as described in the Final Safety Analysis Report Update. In October 2009, the inspectors identified that the replacement reactor head contractor used incorrect damping values in the replacement head design. The contractor was unable to demonstrate that the design met ASME Code using the damping values specified in the plant design basis. On November 5, 2009, the licensee incorporated the new damping values and revised the method for determining the seismic response spectra. Using NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, the inspectors concluded that these changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases. The licensee's 50.59 evaluation, Licensing Basis Impact Evaluation LEBE 2009-021, "Integrated Head Assembly,“ was less than adequate to conclude that prior NRC approval was not required for the changes. The licensee entered this issue into their corrective action program as 50276288.

The failure of Pacific Gas and Electric to perform an adequate 10 CFR 50.59 evaluation prior to changing the facility as described in the Final Safety Analysis Report Update is a performance deficiency. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. The inspectors concluded that the issue was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated this issue using the Significance Determination Process. The inspectors

concluded that the violation affected the Initiating Events Cornerstone because the change potentially decreased the structural integrity of the control rod drive mechanism reactor coolant pressure barrier and screened Green because assuming worst case degradation, the finding would not result in exceeding the technical specification limit for reactor coolant system leakage nor have a likely effect on other mitigation systems resulting in a total loss of their safety function. The inspectors concluded that the violation was a Severity Level IV because the issue screened Green under the Significance Determination Process. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the original problem associated with the replacement reactor head design such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2009005 *(pdf)*

**Significance:** G Dec 04, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow a Level 1 Quality Assurance Program Affecting Human Performance Procedure**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure of training personnel to ensure activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Specifically, between September 27, 2009 and November 17, 2009, training personnel failed to follow Level 1 Quality Assurance Program Affecting Procedure SO123 XXI-1.11.23, "Human Performance Training Program Description," Revision 0, to ensure workers received human performance training before hands-on work was performed in the plant, which resulted in over 80 employees not receiving human performance training and contributed to at least two human performance events. This finding was entered into the licensee’s corrective action program as Nuclear Notification 200670169.
The finding is greater than minor because, if left uncorrected, the failure to follow procedures to provide human performance training, could lead to more significant safety concerns as is evidenced by the two human performance events that occurred by untrained individuals. This finding is associated with the Initiating Events Cornerstone. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding is determined to have very low safety significance because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee failed to take appropriate corrective actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity [P.1(d)].

Inspection Report# : 2009009 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: FIN Finding
**Failure to Perform Corrective Actions Resulted in an Unplanned Trip**
A self-revealing finding was identified after Pacific Gas and Electric failed to implement planned corrective actions resulting in the loss of cooling to a main transformer, a rapid shutdown and a manual reactor trip of Unit 2. On June 30, 2009, cooling to a main transformer was lost because a fuse opened in the 480 volt power circuit due to loose terminal connections in the cooling control panel. Plant operators rapidly shut-down the unit from full power after transformer cooling was lost. A previous failure of transformer cooling due to loose terminal connections occurred on Unit 1, also resulting in a reactor trip. Corrective actions to prevent recurrence following the previous event included replacement of the main transformer terminations in the cooling control panels. Review of the work orders revealed that these corrective actions were not completed and the work documents were closed. While the failure to complete the corrective actions was a latent issue, the inspectors concluded that the licensee had a recent opportunity to identify the issue. Plant technicians implemented thermograph monitoring of main transformer cooling circuits and identified hot 480 volt power terminations in the Unit 2 main transformer cooling disconnect box in April 2009. These hot terminations should have prompted Pacific Gas and Electric to review internal operating experience related to main transformer cooling issues. The licensee entered this finding into corrective program as Notification 50260721.

The inspectors concluded that the finding is greater than minor because it is associated with the equipment performance attribute of the initiating events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that interrupt plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined the finding to have very low safety significance because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigation equipment or functions. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the operating experience component because Pacific Gas and Electric failed to perform an adequate internal operating experience review following the discovery of hot terminations on Unit 2 main transformer in April 2009 [P.2(a)].

Inspection Report# : 2009004 *(pdf)*

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# Mitigating Systems

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Effectively Implement the Seismically-induced Systems Interaction Program**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric personnel failed to effectively implement the Seismically Induced System Interaction Program. The Seismic Interaction Program is part of the design basis mitigation strategy for a potential 7.5 magnitude Hosgri earthquake and is required by Procedure AD4.ID3, “SISIP Housekeeping Activities.” The inspectors identified three examples of transient equipment and materials improperly staged in seismically induced system interaction target areas. Pacific Gas and Electric had not analyzed the transient equipment to assess the risk to safety related components as required by plant procedures. Pacific Gas and Electric entered this finding into the corrective action program as Notification 50299740.

The finding is more than minor because the failure to follow the Seismically Induced System Interaction Program is associated with the Mitigating Systems Cornerstone external events protection attribute and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding had very low safety significance because none of the examples of improperly staged equipment resulted in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. The inspectors concluded this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee’s past actions to address Seismically Induced System Interaction Program deficiencies were not effective.
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report with the Current Plant Design Bases**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with the current design basis. The inspectors identified that the current Final Safety Analysis Report Update, Revision 18, Sections 3.1, 6.4, 6.5, and 9.4 did not capture the current design basis for the control room, component cooling water, and auxiliary feedwater systems. The failure of the licensee to provide current design basis information in the Final Safety Analysis Report Update had an adverse impact on the plant modification process, the licensee’s ability to assess operability for degraded plant systems, and the NRC’s ability to ensure that regulatory requirements were met. The licensee entered this violation into the corrective action program as Notifications 50308588, 50306131, 5030799, and 50307476.

The inspectors evaluated this violation using the traditional enforcement process because the issue affected the NRC’s ability to perform its regulatory function. The inspectors concluded that the violation is more than minor because the

incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. The inspectors concluded the violation is Severity Level IV because the erroneous information was not used to make an unacceptable change to the facility or procedures that would have resulted in greater than very low safety significance under the Significance Determination Process. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of condition of previous similar violation and take appropriate corrective actions.
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovering a condition that could have prevented the fulfillment of a safety function. On November 22, 2005, the licensee determined that plant operators may not have had the capability to align either residual heat removal train to the cold leg recirculation mode of emergency core cooling following certain small break loss of coolant accidents. Plant engineers determined that the residual heat removal containment sump suction valve operators were inadequately sized to open against the differential pressure generated by the pumps operating in recirculation for an extended period. Plant engineers identified this condition during a follow up of industry operating experience. The licensee initially concluded that the condition was not reportable because the operating experience was not applicable to Diablo Canyon. The licensee failed to re-screen the issue for reportability after determining that the plant was susceptible to the condition. The licensee entered this issue into the corrective action program as Notifications 50301839 and 50295784.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded the violation was a Severity Level IV because the licensee failed to submit a required licensee event report. The inspectors did not assign a crosscutting aspect because the performance deficiency represented a latent issue.
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation Following the Failure of Both Motor-Driven Auxiliary Feedwater Trains**
The inspectors identified a noncited violation of 10 CFR, Part 50, Appendix B, Criteria XVI, "Corrective Actions," after Pacific Gas and Electric failed to implement adequate corrective actions following a protection system failure. On June 29, 2009, a protection system card failure resulted in the inoperability of both motor-driven auxiliary feedwater trains. The licensee concluded that the failure of the auxiliary feedwater trains were expected as part of the protection system design and limited corrective actions to replacing the failed card. The inspectors concluded that the protection system design did not meet the design basis, which required that no single active failure would prevent the auxiliary feedwater system from meeting the safety function. The licensee entered this issue into the corrective action program as Notifications 50251823, 50298491 and 50254412.

The inspectors concluded that the finding is greater than minor because the vulnerability of auxiliary feedwater to a single failure is associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding to have very low safety significance because the condition did not represent a loss of system safety function. While the single failure of the protection system card resulted in the inoperability of both motor-driven auxiliary feedwater trains, the turbine-driven auxiliary feedwater train was available to perform the safety function. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the auxiliary feedwater failure such that the resolutions address causes and extent of conditions, as necessary.

Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit a Licensee Event Report following the Common-Cause Failure of Independent Trains or Channels**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a common-cause failure of three control room radiation monitors. The inspectors concluded that monitors failed on October 13, 2009 as a result of water intrusion due to heavy rains. The inspectors concluded that common cause failure of the radiation monitors was reportable under 10 CFR 50.73(a)(2)(vii). Pacific Gas and Electric subsequently reported the event on February 17, 2010, as Licensee Event Report 2010-001-00, Control Room Ventilation Pressurization Due to Radiation Detector Failures. The licensee entered this issue into the corrective action program as Notification 50301839.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded that this was a Severity Level IV noncited violation because the licensee failed to submit a required licensee event report. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the radiation monitor failures to ensure NRC reportability requirements were met.
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate 50.59 Evaluation for Steam Generator Tube Rupture Analysis**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a change to the facility as described in the Final Safety Analysis Report Update. In 1992, the licensee identified that auxiliary feedwater and steam generator power-operated relief valve flow rates assumed in the steam generator tube rupture accident analysis were non-conservative. To address the non-conforming condition, Pacific Gas and Electric changed the accident analysis to include a new time critical operator action to terminate turbine-driven auxiliary feedwater flow 5.54 minutes after the reactor trip and credit motor driven auxiliary feedwater automatic level control to the ruptured steam generator. The licensee did not perform a 10 CFR 50.59 safety evaluation of these changes. The NRC basis of approval of the accident analysis include four time critical operator actions, each assumed to occur after the first 10 minutes following the accident. The inspectors concluded that NRC approval was required before the licensee added the new time critical manual action under the 10 CFR 50.59 Rule in effect at the time because the change reduced the margin to safety to the basis of Technical Specification 3.7.4, "10% Atmospheric Dump Valves." The inspectors also concluded that prior NRC approval was required under the current 50.59 Rule because the change result in a departure from a method of evaluation described in the Final Safety Analysis Report Update. The performance deficiency, a less than adequate 50.59 evaluation, was the result of a latent issue. However, the inspectors concluded that the licensee had reasonable recent opportunities to identify the problem. The inspectors also concluded that plant programs, processes or organizations have not changed such that the problem would not reasonably occur today and that the most significant contributor to the performance deficiency was reflective of current plant performance. The licensee entered this issue into their corrective action program as Notification 50270786.

The failure of Pacific Gas and Electric to perform a 10 CFR 50.59 evaluation of the changes to the steam generator tube rupture accident analysis was a performance deficiency. The inspectors evaluated this issue using traditional enforcement because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. The issue was more than minor because of reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609.04, "Phase 1 Initial Screening and Characterization of Findings." The finding affected the Mitigating Systems Cornerstone because the change described the operator actions required to mitigate steam generator tube rupture accident. The inspectors concluded the finding screened Green because the finding was a design deficiency that did not result in the loss of operability or functionality. The inspectors concluded that the violation was a Severity Level IV because the issue screened Green under the Significance Determination Process. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the steam generator tube rupture analysis such that the resolutions addressed causes and extent of condition [P.1(c).

Inspection Report# : 2009005 *(pdf)*

**Significance:** G Dec 04, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Written Procedures Covered in Regulatory Guide 1.33**
The inspectors identified a non-cited violation of Technical Specification 5.5.1, "Procedures," for the failure of procedure writer personnel to maintain written procedures covered in Regulatory Guide 1.33. Specifically, from initial plant startup of Units 2 and 3 to November 2009, no process requirement or procedure existed to identify procedures that required technical changes so that those procedures could be suspended or put an administrative hold until the required changes were made. This resulted in an uncontrolled procedure requiring technical changes available to use on a safety-related system without flagging the required changes. This finding was entered into the licensee's corrective action program as Nuclear Notification 200671179.
The finding is greater than minor because, if left uncorrected, the failure to maintain and control procedures could lead to a more significant safety concern by having technically inaccurate procedures being used on safety-related systems. This finding is associated with the Mitigating Systems Cornerstone. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding was determined to have a very low safety significance because the finding did not result in a loss of a system safety function, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program because problems were not thoroughly evaluated such that the resolutions addressed the causes and extent of conditions. This includes properly classifying and prioritizing conditions adverse to quality [P.1(c)] (Section 4OA2).

Inspection Report# : 2009009 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct a Degraded Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition (5), "Fire Protection," after Pacific Gas and Electric failed to maintain Fire Door 155 in the rated condition. On September 1, 2009, the inspectors identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. Fire Door 155 was required to provide a 1½ hour rated barrier between Fire Areas 4B and S-2. The licensee re-engaged the latching mechanism and entered the condition into the corrective action program as Notification 50265691. On September 16, 2009, the inspectors again identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. The licensee subsequently determined that the latching mechanism had been defective. The inspectors concluded the most significant contributor to the violation was the less than adequate corrective action taken by the licensee following identification of the problem on September 1, 2009.

This finding is more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door is a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated barrier function. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water-based fire suppression system was in place in the exposing fire area. The licensee entered this violation into the corrective action program as Notification 50268494. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the degraded fire door such that the resolution address causes and extent of condition [P.1(c)].

Inspection Report# : 2009004 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Emergency Operating Procedures Following a Reactor Trip**
The inspectors identified a noncited violation of Technical Specification 5.4.1.b, "Emergency Operating Procedures," after plant operators failed to enter Emergency Operating Procedures E-0, "Reactor Trip or Safety Injection," and E-0.1, "Reactor Trip Response," following a Unit 2 reactor trip on June 30, 2009. Plant operators initiated a rapid load reduction from full power following loss of cooling to a main transformer bank. Plant operators manually tripped the reactor at about two percent power and proceeded to the procedure for placing the unit in cold shutdown. Plant operators did not perform the required steps in Emergency Operating Procedures E-0 and E-0.1 following the reactor trip. The inspectors concluded that the most significant contributor to the violation was less than adequate direction in the procedure used for rapid load reduction. The licensee entered this violation into the corrective action program as Notification 50262363.

The finding is greater than minor because the failure of operations personnel to implement emergency operator procedures was associated with the mitigating systems cornerstone human performance attribute to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded the significance of this finding is of very low safety significance because the finding was not a design or qualification deficiency, did not result in loss of equipment operability or functionality, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had a crosscutting aspect in the area of human performance associated with the resource component because Pacific Gas and Electric did not have a complete rapid load reduction procedure [H.2(c)].
Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with Current Accident Analysis**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with a critical operator action assumed in the plant steam generator tube rupture accident analysis. The steam generator tube rupture accident analysis assumed that the ruptured steam generator will not overfill with water during the accident. To ensure a margin to overfill, the accident analysis included a critical assumption that plant operators would manually trip the turbine-driven auxiliary feedwater pump within 5.54 minutes following the reactor trip. Final Safety Analysis Report Update Section 15.4.3.1, "Identification of Causes and Accident Description," and Final Safety Analysis Report Update Table 15.4-12, "Operator Action Times for Design Basis SGTR Analysis," provided a detailed description of the time dependant operator actions assumed in the accident analysis. The inspectors identified that neither section included the critical assumed operator action to trip the turbine-driven auxiliary feedwater pump. The inspectors concluded that the licensee had a reasonable opportunity to identify and correct the problem when the results of the revised steam generator tube rupture accident, supporting steam generator replacement, was updated in the Final Safety Analysis Report Update in October 2008. The licensee entered this violation into the corrective action program as Notification 50269753.

The inspectors evaluated this finding with the traditional enforcement process because the issue affected the NRC's ability to perform its regulatory function. The inspectors concluded that the finding is greater than minor because the failure to update the required critical operator action assumed in the accident analysis could have a material impact on safety or licensed activities. The inspectors concluded that the violation is Severity Level IV because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement a corrective action program with a low threshold for identifying issues and failed to identify the inaccuracies in the accident analysis as described in the Final Safety Analysis Report Update [P.1(a)].

Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit a Licensee Event Report for a Condition Prohibited by the Plant's Technical Specifications**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a condition prohibited by technical specifications. The licensee failed to correctly evaluate the March 18, 2009, failure of the Unit 2 control rod demand position indicators for reportability. The inspectors concluded that the failure of control rod position indicators was a condition prohibited by Technical Specification 3.17, "Rod Position Indication."

This finding is greater than minor because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the regulations in order to perform its regulatory function. This finding affected the mitigating systems cornerstone. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV, noncited violation. The licensee entered this issue into the corrective action program as Notification 50242153. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the Unit 2 control rod demand position indicators for reportability.
Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the FSAR update with Current Plant Design Criteria**
The inspectors identified a noncited violation of 10 CFR Part 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with current plant design criteria. The Final Safety Analysis Report Update stated that Diablo Canyon was designed to comply with the Atomic Energy Commission General Design Criteria for Nuclear Power Plant Construction Permits, published in July 1967. The inspectors identified that the Diablo Canyon Safety Evaluation Report stated that the NRC used General Design Criteria published in July 1971 to review the plant design. In addition, during the initial licensing process, the licensee stated that the plant was evaluated against the 1971 design criteria during the licensing process.

The inspectors evaluated this finding using the traditional enforcement process because the failure to update the Final Safety Analysis Report affected the NRC's ability to perform its regulatory function. The inspectors concluded that the failure to update the Final Safety Analysis Report was a Severity Level IV violation based on the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I – Reactor Operations, dated January 14, 2005, because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take appropriate corrective actions to address safety issues and adverse trends in a timely manner.
Inspection Report# : 2009003 *(pdf)*

**Significance:** G Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following the Loss of Design Control for the 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, "Corrective Action," after Pacific Gas and Electric failed to adequately correct a nonconforming condition related to the adequacy of design documentation to demonstrate the acceptability of design control for the 500 kV delayed offsite power system. The licensee stated the design control documentation demonstrated that the offsite power system met the design basis was not retrievable. The licensee entered this nonconforming condition into the corrective action system. On October 28, 2008, plant engineers completed an evaluation of the nonconforming condition and concluded the delayed offsite power system design basis was demonstrated by a "road map" of pre-existing analyses created to support other plant

functions. The inspectors concluded that the “road map” was less than adequate because the licensee failed to consider the affect of the loss of reactor coolant pump seal cooling and injection anticipated during the time needed to align the offsite power supply to the engineering safety feature buses. The inspectors concluded that the failure of the licensee to promptly correct the nonconforming condition and ensure that the “road map” implemented measures for verifying or checking the adequacy of design assumptions was reflective of current performance.

This finding is more than minor because the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. The inspectors concluded this finding is of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition to ensure that the offsite power system design basis was met.
Inspection Report# : 2009003 *(pdf)*

**Significance: SL-IV** Jun 30, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility as Described in the Final Safety Analysis Report Update Associated with 500 kV Offsite Power Source**
The inspectors identified a noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a thermal hydraulic analysis to determine if prior NRC approval was required for a 30-minute delay time to align offsite power. This analysis, Calculation STA 274, “RETRAN Evaluation of GDC 17 Loss of AC Scenario,” Revision 0, demonstrated that the 30-minute delayed offsite power source was acceptable. On December 31, 2008, a Pacific Gas and Electric 10 CFR 50.59 screen concluded that Calculation STA 274 was not required to be evaluated to determine if prior NRC approval was required for the delay time. On March 31, 2009, the inspectors concluded that the licensee was required to evaluate Calculation STA 274 to determine if prior NRC approval was needed. On May 27, 2009, Pacific Gas and Electric completed the 50.59 evaluation and concluded that prior NRC approval was required for the 30-minute delay time to align offsite power.

The inspectors concluded that the finding is more than minor because the changes made to the facility required prior NRC review and approval. The finding affected the Mitigating Systems Cornerstone because the change described how the delayed offsite power source met the design basis. The inspectors concluded the finding is of very low safety significance because the finding was a design deficiency that did not result in the loss of operability or functionality. Because the issue affected the NRC’s ability to perform its regulatory function, the inspectors evaluated this finding using the traditional enforcement process. This issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required.
Inspection Report# : 2009003 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Work Planning Resulted in the Release of Two Gas Decay Tanks**
The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1, “Procedures,” after Pacific Gas and Electric inadvertently released the contents of two gas decay tanks into the auxiliary building. Gas Decay Tank 2-2 was in “purge mode.” On October 11, 2009, plant operators were implementing an equipment control clearance to drain the emergency core cooling systems. A second group of operators were implementing a core offload master clearance. The parallel implementation of both equipment clearances resulted in Gas Decay Tank 2-2 to be vented into the auxiliary building. The auxiliary building operator received a low gas header pressure alarm after

the pressure dropped to 15 psig. Per procedure, the operator aligned Gas Decay Tank 2-3 to “purge” mode. As a result, the second gas decay tank was released into the auxiliary building through the open vent path. The inspectors concluded that the radiological consequence of the event did not result in a potential for overexposure because the reactor had been shutdown since October 3, 2009.

The inspectors concluded that the failure to properly implement the core offload master equipment control clearance was a performance deficiency. The finding is more than minor because the performance deficiency could be reasonably viewed as a precursor to a significant event. The inspectors determined the finding to have very low safety significance because the performance deficiency only represented a degradation of the auxiliary building radiological barrier function. This finding has a crosscutting aspect in the area of human performance associated with the work control component because the licensee did not adequately plan and coordinate the two clearance activities or fully consider the impact the work had on different job activities and the need for the two work groups to maintain interfaces [H.3(b)].

Inspection Report# : 2009005 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Properly Plan a Maintenance Activity**
The inspectors reviewed a self-revealing, noncited violation of Technical Specification 5.4.1(a) for failure to properly plan numerous outage maintenance activities including the disassembly of the Unit 2 reactor head. Specifically, Work Orders 68004363 (disassembly of the old head) and 68003988 (scaffolding activities) were not properly planned, thereby requiring those maintenance activities to be changed and/or repeated, which resulted in increased radiation exposure. Radiation Work Permits 09-2233 and 09-2237 for the disassembly of the Unit 2 old reactor vessel closure head and supporting activities during Refueling Outage 15 had an initial combined dose estimate of 5.869 rem and 1102 man-hours. However, the job ended with an actual combined dose of 17.378 rem and 1882 man-hours, which exceeded the initial dose estimate by 296 percent. The overarching reason for exceeding the original dose estimate was improper planning and control for the maintenance, which increased the man-hours to complete the task by 170 percent. The licensee entered this deficiency in the corrective action program as Notification 50275107 and plan to perform an apparent cause evaluation.

The failure to properly plan maintenance activities is a performance deficiency. This finding is greater than minor because it affected the Occupational Radiation Safety cornerstone attribute of Program and Process in that the inadequate ALARA planning caused increased collective radiation dose for the job activity to exceed 5 person-rem and the planned dose by more than 50 percent. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined this finding to be of very low safety significance because although it involved ALARA planning and controls, the licensee’s latest rolling three-year average does not exceed 135 person-rem per unit. Furthermore, the finding had an associated human performance cross-cutting aspect in the work control component because the licensee did not fully incorporate job site conditions, plant structures, systems, and components, as well as human-system interface and the need for planned contingencies to maintain doses ALARA [H.3(a)].

Inspection Report# : 2009005 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : May 26, 2010

# Diablo Canyon 2
# 2Q/2010 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Replacement Reactor Head Modification Design Control**
The inspectors identified a noncited violation of Title 10 CFR, Part 50, Appendix B, Criterion III, "Design Control," after the design contractor failed to perform adequately calculations demonstrating that the replacement reactor head met ASME Code acceptance criteria. The contractor failed to use the critical seismic damping values specified in the plant design basis for the design of the integrated head assembly and the control rod drive mechanism housing assembly and when calculating component stress during a postulated design basis earthquake. The licensee entered this condition into the corrective action program as Notifications 50276107 and 50276288.

The inspectors concluded that the failure to properly implement the plant design basis in the replacement head design was a performance deficiency. The finding is more than minor because the performance deficiency is associated with the Initiating Events Cornerstone design control attribute and adversely affected the cornerstone objective to limit the likelihood of loss of a coolant accident during a seismic event. The inspectors determined the finding is of very low safety significance because assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for reactor coolant system leakage nor have likely affected other mitigation systems resulting in a total loss of their safety function. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not identify the use of improper damping values with a low threshold for identifying issues during oversight of contractor activities and design reviews [P.1(a)].

Inspection Report# : 2009005 *(pdf)*

**Significance: SL-IV** Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Lees Than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after the licensee failed to perform an adequate evaluation to demonstrate that prior NRC approval was not required before making changes to the facility as described in the Final Safety Analysis Report Update. In October 2009, the inspectors identified that the replacement reactor head contractor used incorrect damping values in the replacement head design. The contractor was unable to demonstrate that the design met ASME Code using the damping values specified in the plant design basis. On November 5, 2009, the licensee incorporated the new damping values and revised the method for determining the seismic response spectra. Using NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, the inspectors concluded that these changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases. The licensee's 50.59 evaluation, Licensing Basis Impact Evaluation LEBE 2009-021, "Integrated Head Assembly," was less than adequate to conclude that prior NRC approval was not required for the changes. The licensee entered this issue into their corrective action program as 50276288.

The failure of Pacific Gas and Electric to perform an adequate 10 CFR 50.59 evaluation prior to changing the facility as described in the Final Safety Analysis Report Update is a performance deficiency. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. The inspectors concluded that the issue was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated this issue using the Significance Determination Process. The inspectors

concluded that the violation affected the Initiating Events Cornerstone because the change potentially decreased the structural integrity of the control rod drive mechanism reactor coolant pressure barrier and screened Green because assuming worst case degradation, the finding would not result in exceeding the technical specification limit for reactor coolant system leakage nor have a likely effect on other mitigation systems resulting in a total loss of their safety function. The inspectors concluded that the violation was a Severity Level IV because the issue screened Green under the Significance Determination Process. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the original problem associated with the replacement reactor head design such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2009005 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: FIN Finding
**Failure to Perform Corrective Actions Resulted in an Unplanned Trip**
A self-revealing finding was identified after Pacific Gas and Electric failed to implement planned corrective actions resulting in the loss of cooling to a main transformer, a rapid shutdown and a manual reactor trip of Unit 2. On June 30, 2009, cooling to a main transformer was lost because a fuse opened in the 480 volt power circuit due to loose terminal connections in the cooling control panel. Plant operators rapidly shut-down the unit from full power after transformer cooling was lost. A previous failure of transformer cooling due to loose terminal connections occurred on Unit 1, also resulting in a reactor trip. Corrective actions to prevent recurrence following the previous event included replacement of the main transformer terminations in the cooling control panels. Review of the work orders revealed that these corrective actions were not completed and the work documents were closed. While the failure to complete the corrective actions was a latent issue, the inspectors concluded that the licensee had a recent opportunity to identify the issue. Plant technicians implemented thermograph monitoring of main transformer cooling circuits and identified hot 480 volt power terminations in the Unit 2 main transformer cooling disconnect box in April 2009. These hot terminations should have prompted Pacific Gas and Electric to review internal operating experience related to main transformer cooling issues. The licensee entered this finding into corrective program as Notification 50260721.

The inspectors concluded that the finding is greater than minor because it is associated with the equipment performance attribute of the initiating events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that interrupt plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined the finding to have very low safety significance because the condition did not contribute to both the likelihood of a reactor trip and the unavailability of mitigation equipment or functions. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the operating experience component because Pacific Gas and Electric failed to perform an adequate internal operating experience review following the discovery of hot terminations on Unit 2 main transformer in April 2009 [P.2(a)].

Inspection Report# : 2009004 *(pdf)*

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# Mitigating Systems

**Significance:** G Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following Identification of a Non-conservative Technical Specification**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, “Corrective Action,” after Pacific Gas and Electric failed to implement prompt corrective actions after identifying a nonconservative technical specification. In December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design basis were met. On January 9, 2009, the licensee entered this condition into the corrective action program. On April 9, 2009, Pacific Gas and Electric concluded that Technical Specification Surveillance Requirement 3.8.1, “AC Sources – Operating,” was not adequate to preserve plant safety and applied the

provisions of Technical Specification Surveillance Requirement 3.0.3, and Administrative Letter 98 10, “Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety.” The licensee did not complete the necessary actions to correct the deficient technical specification by submitting an adequate license amendment request. The inspectors concluded the most significant contributor to the finding was a less than adequate engineering evaluation to support the new emergency diesel generator loading profiles following the previous violation. The licensee entered the performance deficiency into the corrective action program as Notification 50232181.

The inspectors determined that the performance deficiency is more than minor because if left uncorrected, the failure to implement prompt corrective actions has the potential to lead to a more significant safety concern. The inspectors concluded the finding was of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. The finding is associated with the Mitigating Systems Cornerstone. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative technical specification such that the resolutions address causes and extent of conditions, as necessary.

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(B) and after Pacific Gas and Electric failed to submit a required licensee event report within 60 days following discovery of a condition prohibited by the plant technical specifications and a condition that could have prevented the fulfillment of a safety function. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme, required by Technical Specification 3.3.5, “Loss of Power Diesel Generator Start Instrumentation,” was inadequate to protect operating engineering safety feature pump motors. The licensee concluded that sustained degraded voltage could result in an overcurrent condition affecting equipment powered from the preferred offsite power supply. This condition was required to be reported to the NRC because the degraded voltage protection scheme rendered engineered safety feature pumps inoperable for a period in excess of the allowable technical specification out of service time and the condition resulted in the loss of the degraded voltage protection scheme safety function on all three vital 4 kV power buses.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC’s ability to perform its regulatory function. The inspectors concluded the violation was a Severity Level IV because the licensee failed to submit an adequate licensee event report. The inspectors determined that the violation was also a finding under the reactor oversight process because licensee personnel failed to adequately evaluate a condition adverse to quality for operability and reportability, as required by station procedures. The inspectors concluded that the finding is more than minor because the failure to properly evaluate degraded plant equipment for past operability and reportability could reasonably be seen to lead to a more significant condition. The inspectors concluded that the finding had very low safety significance because the failure to adequately evaluate the condition did not result in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the degraded voltage protection scheme such that the resolutions address causes and extent of conditions, as necessary.

Inspection Report# : 2010003 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation

**Failure to Effectively Implement the Seismically-induced Systems Interaction Program**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric personnel failed to effectively implement the Seismically Induced System Interaction Program. The Seismic Interaction Program is part of the design basis mitigation strategy for a potential 7.5 magnitude Hosgri earthquake and is required by Procedure AD4.ID3, "SISIP Housekeeping Activities." The inspectors identified three examples of transient equipment and materials improperly staged in seismically induced system interaction target areas. Pacific Gas and Electric had not analyzed the transient equipment to assess the risk to safety related components as required by plant procedures. Pacific Gas and Electric entered this finding into the corrective action program as Notification 50299740.

The finding is more than minor because the failure to follow the Seismically Induced System Interaction Program is associated with the Mitigating Systems Cornerstone external events protection attribute and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding had very low safety significance because none of the examples of improperly staged equipment resulted in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. The inspectors concluded this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee's past actions to address Seismically Induced System Interaction Program deficiencies were not effective [P.1(d)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report with the Current Plant Design Bases**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with the current design basis. The inspectors identified that the current Final Safety Analysis Report Update, Revision 18, Sections 3.1, 6.4, 6.5, and 9.4 did not capture the current design basis for the control room, component cooling water, and auxiliary feedwater systems. The failure of the licensee to provide current design basis information in the Final Safety Analysis Report Update had an adverse impact on the plant modification process, the licensee's ability to assess operability for degraded plant systems, and the NRC's ability to ensure that regulatory requirements were met. The licensee entered this violation into the corrective action program as Notifications 50308588, 50306131, 5030799, and 50307476.

The inspectors evaluated this violation using the traditional enforcement process because the issue affected the NRC's ability to perform its regulatory function. The inspectors concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. The inspectors concluded the violation is Severity Level IV because the erroneous information was not used to make an unacceptable change to the facility or procedures that would have resulted in greater than very low safety significance under the Significance Determination Process. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of condition of previous similar violation and take appropriate corrective actions [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovering a condition that could have prevented the fulfillment of a safety function. On November 22, 2005, the licensee determined that plant operators may not have had the capability to align either residual heat removal train to the cold leg recirculation mode of emergency core cooling following certain small break loss of coolant accidents. Plant engineers determined that the residual heat removal

containment sump suction valve operators were inadequately sized to open against the differential pressure generated by the pumps operating in recirculation for an extended period. Plant engineers identified this condition during a follow up of industry operating experience. The licensee initially concluded that the condition was not reportable because the operating experience was not applicable to Diablo Canyon. The licensee failed to re-screen the issue for reportability after determining that the plant was susceptible to the condition. The licensee entered this issue into the corrective action program as Notifications 50301839 and 50295784.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded the violation was a Severity Level IV because the licensee failed to submit a required licensee event report. The inspectors did not assign a crosscutting aspect because the performance deficiency represented a latent issue.
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation Following the Failure of Both Motor-Driven Auxiliary Feedwater Trains**
The inspectors identified a noncited violation of 10 CFR, Part 50, Appendix B, Criteria XVI, "Corrective Actions," after Pacific Gas and Electric failed to implement adequate corrective actions following a protection system failure. On June 29, 2009, a protection system card failure resulted in the inoperability of both motor-driven auxiliary feedwater trains. The licensee concluded that the failure of the auxiliary feedwater trains were expected as part of the protection system design and limited corrective actions to replacing the failed card. The inspectors concluded that the protection system design did not meet the design basis, which required that no single active failure would prevent the auxiliary feedwater system from meeting the safety function. The licensee entered this issue into the corrective action program as Notifications 50251823, 50298491 and 50254412.

The inspectors concluded that the finding is greater than minor because the vulnerability of auxiliary feedwater to a single failure is associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding to have very low safety significance because the condition did not represent a loss of system safety function. While the single failure of the protection system card resulted in the inoperability of both motor-driven auxiliary feedwater trains, the turbine-driven auxiliary feedwater train was available to perform the safety function. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the auxiliary feedwater failure such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit a Licensee Event Report following the Common-Cause Failure of Independent Trains or Channels**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a common-cause failure of three control room radiation monitors. The inspectors concluded that monitors failed on October 13, 2009 as a result of water intrusion due to heavy rains. The inspectors concluded that common cause failure of the radiation monitors was reportable under 10 CFR 50.73(a)(2)(vii). Pacific Gas and Electric subsequently reported the event on February 17, 2010, as Licensee Event Report 2010-001-00, Control Room Ventilation Pressurization Due to Radiation Detector Failures. The licensee entered this issue into the corrective action program as Notification 50301839.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded that this was a Severity Level IV noncited violation because the licensee failed to submit a required licensee event report.

Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the radiation monitor failures to ensure NRC reportability requirements were met [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Design and Configuration Control Requirements**
The inspection team identified a noncited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, which requires licensees to implement measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These design control measures include verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. Specifically, on February 16, 2008, plant engineering personnel failed to implement the design control process for a modification to the Unit 2 residual heat removal containment sump valves when they inappropriately used maintenance procedures to reduce the valve stroke lengths from 15.5 to 13.8 inches. The invalid design change resulted in the inoperability of both emergency core cooling trains between April 8, 2008, (when the plant entered Mode 4) and October 22, 2009. The reduced sump valve stroke length also caused a portion of the sump valve disc to remain in the residual heat removal suction flow path, reducing the available residual heat removal pump net positive suction head. The licensee entered this condition into their corrective action program as Notification 50277252.

The inspection team concluded that the failure of plant engineering to use the design control process was a performance deficiency within the licensee's ability to foresee and correct. The finding is more than minor because it affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding required a Phase 2 analysis because the finding represented the loss of a safety system function. The Phase 2 analysis determined that this finding was potentially greater than Green; therefore, a Phase 3 analysis was completed by a regional senior reactor analyst. The Phase 3 analysis determined that this issue was of very low safety significance (Green), owing principally to the fact that operators could have opened the affected valves locally with a very high probability of success. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the failure of the valves to meet the specified stroke time to ensure that the resolution fully addressed the causes and extent of condition, as necessary [P.1(c)].

Inspection Report# : 2009009 *(pdf)*

**Significance:** G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Conduct an Adequate Post-Modification Test**
The inspection team identified a noncited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, which requires that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service. Specifically, the licensee failed to perform testing to assure that the interlock circuitry associated with the residual heat removal containment sump valves (SI-2-8982A and B) would perform satisfactorily in service following a modification on February 16, 2008, that changed the stroke lengths. As a consequence, remote operation of the valves needed to initiate high pressure recirculation was lost for an entire operating cycle. The licensee entered this issue into their corrective action program as Notification 50277252.

The failure to establish adequate post-modification testing requirements was a performance deficiency within the licensee's ability to foresee and correct. The finding is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding required a Phase 2 analysis because the finding represented the loss of a safety system function. The Phase 2 analysis determined that this finding was potentially greater than Green; therefore, a Phase 3 analysis was completed by a regional senior reactor analyst. The Phase 3 analysis determined that this issue was of very low safety significance (Green), owing principally to the fact that operators could have opened the affected valves locally with a very high probability of success. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee failed to implement a corrective action program with a threshold sufficient to identify issues associated with the failure to meet sump valve post-modification test acceptance criteria [P.1(a)].

Inspection Report# : 2009009 *(pdf)*

**Significance: SL-IV** Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility as Described in the Final Safety Report Update Associated with the Addition of Manual Actions in the Safety Analysis**
The inspection team identified a noncited violation of 10 CFR 50.59, which states that a licensee may make changes to the facility as described in the final safety analysis report without obtaining a license amendment if the change does not result in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases or in the safety analyses. This regulation further requires the licensee to include a written evaluation providing the basis for concluding that a license amendment is not required. On November 21, 2005, the licensee failed to provide a written evaluation concluding that a license amendment was not required for a change to the facility as described in the final safety analysis report. Specifically, the licensee identified a condition where large differential pressure across the residual heat removal suction containment sump valves could cause them to fail to open during certain small break loss of coolant accidents. On October 5, 2005, the licensee revised Emergency Operating Procedure E-1, “Loss of Reactor or Secondary Coolant,” to add an operator action to align component cooling water to the residual heat removal heat exchanger. On June 16, 2009, the licensee again revised Emergency Operating Procedure E-1 to specify that operator action to align component cooling water within 30 minutes was a time critical operator action. The licensee did not evaluate either change to determine if prior NRC approval was required for the new manual actions. The licensee entered this issue into their corrective action program as Notification 50276288.

The failure of the licensee to perform a 10 CFR 50.59 evaluation of a new manual action supporting the plant’s design basis was a performance deficiency within the licensee’s ability to foresee and correct. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC’s ability to perform its regulatory function. The inspectors concluded that the issue was more than minor because of a reasonable likelihood that the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609.04, “Phase 1 Initial Screening and Characterization of Findings.” The inspectors concluded that the issue affected the Mitigating Systems Cornerstone and screened Green because the finding was a design or qualification deficiency confirmed not to result in loss of operability. The issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1(c)].

Inspection Report# : 2009009 *(pdf)*

**Significance: SL-IV** Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate 50.59 Evaluation for Steam Generator Tube Rupture Analysis**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a change to the facility as described in the Final Safety Analysis Report Update. In 1992, the licensee identified that auxiliary feedwater and steam generator power-operated relief valve flow rates assumed in the steam generator tube rupture accident analysis were non-conservative. To address the non-conforming condition, Pacific Gas and Electric changed the accident analysis to include a new time critical operator action to

terminate turbine-driven auxiliary feedwater flow 5.54 minutes after the reactor trip and credit motor driven auxiliary feedwater automatic level control to the ruptured steam generator. The licensee did not perform a 10 CFR 50.59 safety evaluation of these changes. The NRC basis of approval of the accident analysis include four time critical operator actions, each assumed to occur after the first 10 minutes following the accident. The inspectors concluded that NRC approval was required before the licensee added the new time critical manual action under the 10 CFR 50.59 Rule in effect at the time because the change reduced the margin to safety to the basis of Technical Specification 3.7.4, "10% Atmospheric Dump Valves." The inspectors also concluded that prior NRC approval was required under the current 50.59 Rule because the change result in a departure from a method of evaluation described in the Final Safety Analysis Report Update. The performance deficiency, a less than adequate 50.59 evaluation, was the result of a latent issue. However, the inspectors concluded that the licensee had reasonable recent opportunities to identify the problem. The inspectors also concluded that plant programs, processes or organizations have not changed such that the problem would not reasonably occur today and that the most significant contributor to the performance deficiency was reflective of current plant performance. The licensee entered this issue into their corrective action program as Notification 50270786.

The failure of Pacific Gas and Electric to perform a 10 CFR 50.59 evaluation of the changes to the steam generator tube rupture accident analysis was a performance deficiency. The inspectors evaluated this issue using traditional enforcement because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. The issue was more than minor because of reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609.04, "Phase 1 Initial Screening and Characterization of Findings." The finding affected the Mitigating Systems Cornerstone because the change described the operator actions required to mitigate steam generator tube rupture accident. The inspectors concluded the finding screened Green because the finding was a design deficiency that did not result in the loss of operability or functionality. The inspectors concluded that the violation was a Severity Level IV because the issue screened Green under the Significance Determination Process. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the steam generator tube rupture analysis such that the resolutions addressed causes and extent of condition [P.1(c).

Inspection Report# : 2009005 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify and Correct a Degraded Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition (5), "Fire Protection," after Pacific Gas and Electric failed to maintain Fire Door 155 in the rated condition. On September 1, 2009, the inspectors identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. Fire Door 155 was required to provide a 1½ hour rated barrier between Fire Areas 4B and S-2. The licensee re-engaged the latching mechanism and entered the condition into the corrective action program as Notification 50265691. On September 16, 2009, the inspectors again identified that Fire Door 155 was inoperable because the external latching mechanism device was not engaged. The licensee subsequently determined that the latching mechanism had been defective. The inspectors concluded the most significant contributor to the violation was the less than adequate corrective action taken by the licensee following identification of the problem on September 1, 2009.

This finding is more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door is a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated barrier function. The inspectors concluded that this finding is of very low safety significance because a non-degraded automatic full area water-based fire suppression system was in place in the exposing fire area. The licensee entered this violation into the corrective action program as Notification 50268494. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the degraded fire door such that the resolution address causes and extent of condition [P.1(c)].

Inspection Report# : 2009004 *(pdf)*

**Significance:** G Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Emergency Operating Procedures Following a Reactor Trip**
The inspectors identified a noncited violation of Technical Specification 5.4.1.b, “Emergency Operating Procedures,” after plant operators failed to enter Emergency Operating Procedures E-0, “Reactor Trip or Safety Injection,” and E-0.1, “Reactor Trip Response,” following a Unit 2 reactor trip on June 30, 2009. Plant operators initiated a rapid load reduction from full power following loss of cooling to a main transformer bank. Plant operators manually tripped the reactor at about two percent power and proceeded to the procedure for placing the unit in cold shutdown. Plant operators did not perform the required steps in Emergency Operating Procedures E-0 and E-0.1 following the reactor trip. The inspectors concluded that the most significant contributor to the violation was less than adequate direction in the procedure used for rapid load reduction. The licensee entered this violation into the corrective action program as Notification 50262363.

The finding is greater than minor because the failure of operations personnel to implement emergency operator procedures was associated with the mitigating systems cornerstone human performance attribute to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded the significance of this finding is of very low safety significance because the finding was not a design or qualification deficiency, did not result in loss of equipment operability or functionality, or screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. This finding had a crosscutting aspect in the area of human performance associated with the resource component because Pacific Gas and Electric did not have a complete rapid load reduction procedure [H.2(c)].
Inspection Report# : 2009004 *(pdf)*

**Significance: SL-IV** Sep 25, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with Current Accident Analysis**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with a critical operator action assumed in the plant steam generator tube rupture accident analysis. The steam generator tube rupture accident analysis assumed that the ruptured steam generator will not overfill with water during the accident. To ensure a margin to overfill, the accident analysis included a critical assumption that plant operators would manually trip the turbine-driven auxiliary feedwater pump within 5.54 minutes following the reactor trip. Final Safety Analysis Report Update Section 15.4.3.1, “Identification of Causes and Accident Description,” and Final Safety Analysis Report Update Table 15.4-12, “Operator Action Times for Design Basis SGTR Analysis,” provided a detailed description of the time dependant operator actions assumed in the accident analysis. The inspectors identified that neither section included the critical assumed operator action to trip the turbine-driven auxiliary feedwater pump. The inspectors concluded that the licensee had a reasonable opportunity to identify and correct the problem when the results of the revised steam generator tube rupture accident, supporting steam generator replacement, was updated in the Final Safety Analysis Report Update in October 2008. The licensee entered this violation into the corrective action program as Notification 50269753.

The inspectors evaluated this finding with the traditional enforcement process because the issue affected the NRC’s ability to perform its regulatory function. The inspectors concluded that the finding is greater than minor because the failure to update the required critical operator action assumed in the accident analysis could have a material impact on safety or licensed activities. The inspectors concluded that the violation is Severity Level IV because the erroneous information was not used to make an unacceptable change to the facility or procedures. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement a corrective action program with a low threshold for identifying issues and failed to identify the inaccuracies in the accident analysis as described in the Final Safety Analysis Report Update [P.1(a)].
Inspection Report# : 2009004 *(pdf)*

# Barrier Integrity

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Work Planning Resulted in the Release of Two Gas Decay Tanks**
The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1, “Procedures,” after Pacific Gas and Electric inadvertently released the contents of two gas decay tanks into the auxiliary building. Gas Decay Tank 2-2 was in “purge mode.” On October 11, 2009, plant operators were implementing an equipment control clearance to drain the emergency core cooling systems. A second group of operators were implementing a core offload master clearance. The parallel implementation of both equipment clearances resulted in Gas Decay Tank 2-2 to be vented into the auxiliary building. The auxiliary building operator received a low gas header pressure alarm after the pressure dropped to 15 psig. Per procedure, the operator aligned Gas Decay Tank 2-3 to “purge” mode. As a result, the second gas decay tank was released into the auxiliary building through the open vent path. The inspectors concluded that the radiological consequence of the event did not result in a potential for overexposure because the reactor had been shutdown since October 3, 2009.

The inspectors concluded that the failure to properly implement the core offload master equipment control clearance was a performance deficiency. The finding is more than minor because the performance deficiency could be reasonably viewed as a precursor to a significant event. The inspectors determined the finding to have very low safety significance because the performance deficiency only represented a degradation of the auxiliary building radiological barrier function. This finding has a crosscutting aspect in the area of human performance associated with the work control component because the licensee did not adequately plan and coordinate the two clearance activities or fully consider the impact the work had on different job activities and the need for the two work groups to maintain interfaces [H.3(b)].

Inspection Report# : 2009005 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Properly Plan a Maintenance Activity**
The inspectors reviewed a self-revealing, noncited violation of Technical Specification 5.4.1(a) for failure to properly plan numerous outage maintenance activities including the disassembly of the Unit 2 reactor head. Specifically, Work Orders 68004363 (disassembly of the old head) and 68003988 (scaffolding activities) were not properly planned, thereby requiring those maintenance activities to be changed and/or repeated, which resulted in increased radiation exposure. Radiation Work Permits 09-2233 and 09-2237 for the disassembly of the Unit 2 old reactor vessel closure head and supporting activities during Refueling Outage 15 had an initial combined dose estimate of 5.869 rem and 1102 man-hours. However, the job ended with an actual combined dose of 17.378 rem and 1882 man-hours, which exceeded the initial dose estimate by 296 percent. The overarching reason for exceeding the original dose estimate was improper planning and control for the maintenance, which increased the man-hours to complete the task by 170 percent. The licensee entered this deficiency in the corrective action program as Notification 50275107 and plan to perform an apparent cause evaluation.

The failure to properly plan maintenance activities is a performance deficiency. This finding is greater than minor because it affected the Occupational Radiation Safety cornerstone attribute of Program and Process in that the inadequate ALARA planning caused increased collective radiation dose for the job activity to exceed 5 person-rem and the planned dose by more than 50 percent. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined this finding to be of very low safety significance because although it involved ALARA planning and controls, the licensee's latest rolling three-year average does not exceed 135 person-rem per unit. Furthermore, the finding had an associated human performance cross-cutting aspect in the work control component because the licensee did not fully incorporate job site conditions, plant structures, systems, and components, as well as human-system interface and the need for planned contingencies to maintain doses ALARA [H.3(a)].

Inspection Report# : 2009005 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : September 02, 2010

# Diablo Canyon 2
# 3Q/2010 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Replacement Reactor Head Modification Design Control**
The inspectors identified a noncited violation of Title 10 CFR, Part 50, Appendix B, Criterion III, “Design Control,” after the design contractor failed to perform adequately calculations demonstrating that the replacement reactor head met ASME Code acceptance criteria. The contractor failed to use the critical seismic damping values specified in the plant design basis for the design of the integrated head assembly and the control rod drive mechanism housing assembly and when calculating component stress during a postulated design basis earthquake. The licensee entered this condition into the corrective action program as Notifications 50276107 and 50276288.

The inspectors concluded that the failure to properly implement the plant design basis in the replacement head design was a performance deficiency. The finding is more than minor because the performance deficiency is associated with the Initiating Events Cornerstone design control attribute and adversely affected the cornerstone objective to limit the likelihood of loss of a coolant accident during a seismic event. The inspectors determined the finding is of very low safety significance because assuming worst case degradation, the finding would not result in exceeding the Technical Specification limit for reactor coolant system leakage nor have likely affected other mitigation systems resulting in a total loss of their safety function. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not identify the use of improper damping values with a low threshold for identifying issues during oversight of contractor activities and design reviews [P.1(a)].

Inspection Report# : 2009005 *(pdf)*

**Significance: SL-IV** Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Lees Than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after the licensee failed to perform an adequate evaluation to demonstrate that prior NRC approval was not required before making changes to the facility as described in the Final Safety Analysis Report Update. In October 2009, the inspectors identified that the replacement reactor head contractor used incorrect damping values in the replacement head design. The contractor was unable to demonstrate that the design met ASME Code using the damping values specified in the plant design basis. On November 5, 2009, the licensee incorporated the new damping values and revised the method for determining the seismic response spectra. Using NEI 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” Revision 1, the inspectors concluded that these changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases. The licensee’s 50.59 evaluation, Licensing Basis Impact Evaluation LEBE 2009-021, “Integrated Head Assembly,“ was less than adequate to conclude that prior NRC approval was not required for the changes. The licensee entered this issue into their corrective action program as 50276288.

The failure of Pacific Gas and Electric to perform an adequate 10 CFR 50.59 evaluation prior to changing the facility as described in the Final Safety Analysis Report Update is a performance deficiency. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC’s ability to perform its regulatory function. The inspectors concluded that the issue was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated this issue using the Significance Determination Process. The inspectors

concluded that the violation affected the Initiating Events Cornerstone because the change potentially decreased the structural integrity of the control rod drive mechanism reactor coolant pressure barrier and screened Green because assuming worst case degradation, the finding would not result in exceeding the technical specification limit for reactor coolant system leakage nor have a likely effect on other mitigation systems resulting in a total loss of their safety function. The inspectors concluded that the violation was a Severity Level IV because the issue screened Green under the Significance Determination Process. The finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the original problem associated with the replacement reactor head design such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2009005 *(pdf)*

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# Mitigating Systems

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify a Degraded Fire Barrier**
The inspectors identified a noncited violation of the Diablo Canyon Facility Operating License Condition (5), "Fire Protection," after Pacific Gas and Electric failed to maintain the integrity of a fire door in the rated configuration. On August 19, 2010, the inspectors identified that Fire Door 223 was inoperable. Fire Door 223 was required to provide a 3-hour rated barrier between Fire Areas 5-A-4 and 5-B-4. A fire in either of these areas could have prevented operation of the auxiliary feedwater, auxiliary saltwater, or component cooling water pumps or steam generator level control from the remote shutdown panel. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to either maintain Fire Door 223 operable or implement compensatory actions within one hour. The inspectors concluded the most significant contributor to the finding was that licensee personnel did not identify and enter the degraded fire door into the Corrective Action Program. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notification 50336901 and completed repairs to the door on August 23, 2010.

The inspectors concluded that the performance deficiency was more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door was a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated fire barrier function. The inspectors concluded the finding was of very low safety significance because the degraded barrier would have provided a minimum of 20 minutes fire endurance protection and ignition sources and combustible materials were positioned that had a fire spread to secondary combustibles, the degraded barrier would not have been subject to direct flame impingement. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a low threshold for identifying and entering issues into the Corrective Action Program [P.1(a)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: FIN Finding
**Inadequate Risk Management During a Planned Auxiliary Saltwater Pump Outage**
The inspectors identified a finding after Pacific Gas and Electric failed to adequately manage risk during planned maintenance activity as required by Procedure AD7.DC6, "On-line Maintenance Risk Management." On April 5, 2010, work control personnel requested that plant operators simultaneously remove Auxiliary Saltwater Pump 2-2 and Component Cooling Water Heat Exchanger 2-2 from service for two scheduled maintenance activities. Plant operators identified that the combination of the auxiliary saltwater pump and component cooling water heat exchanger out of service at the same time would result in an elevated maintenance risk (Yellow). Procedure AD7.DC6, "On-line Maintenance Risk Management", Section 2.1, required that the licensee manage plant risk during on-line maintenance

by minimizing the number of risk significant equipment simultaneously removed from service. The inspectors concluded that these two maintenance activities could have been performed in series rather than in parallel without affecting the duration either component was unavailable for maintenance. The licensee entered the performance deficiency into the corrective action program as Notification 50309451.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” because the work scope unnecessarily placed the plant into a higher licensee-established risk category and required additional risk management actions. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement adequate corrective actions to prevent unnecessarily entering elevated plant risk for the planned maintenance [P.1(d)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Risk Assessment during Planned Maintenance Activities**
The inspectors identified a noncited violation of 10 CFR 50.65 after Pacific Gas and Electric failed to perform a risk assessment after plant conditions had changed. On July 13, 2010, Pacific Gas and Electric identified that station personnel failed to complete Technical Specification Surveillance Requirement 3.3.4.2, “Remote Shutdown System,” within the specified frequency for both Units. As provided by Surveillance Requirement 3.0.3, the licensee performed a risk evaluation to extend the required surveillance completion time beyond twenty-four hours. The licensee initiated the missed surveillance tests and identified results were outside acceptance criteria. On July 26, 2010, Operations personnel declared several remote shutdown system functions inoperable because reasonable expectation no longer existed that remote shutdown system could perform its safety function. Pacific Gas and Electric failed to reassess the effect on plant risk resulting from inoperable remote shutdown system functions before continuing with scheduled maintenance. A subsequent risk assessment concluded that plant risk was in a higher risk category due to planned maintenance activities conducted during this time frame. The licensee entered the performance deficiency into the corrective action program as Notification 50331841.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” because the overall elevated plant risk would put the plant into a higher licensee-established risk category. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee failed to follow its maintenance risk procedure and reassess plant risk due to changing plant conditions [H.4(b)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination**
The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to promptly evaluate two nonconforming conditions for operability as

required by Procedure OM7.ID12, “Operability Determination.” The first example involved the failure of engineering personnel to promptly notify plant operations of the failure of the emergency diesel generators to meet licensing and design frequency and voltage recovery requirements. This issue was identified by the NRC on May 11, 2010, but not evaluated for the effect on diesel operability until September 9, 2010. The second example also involved the failure of engineering personnel to promptly notify plant operations to evaluate a nonconforming condition associated with a common cross-tie line that connected both auxiliary saltwater trains. This issue was identified by the NRC on July 22, 2010, but not evaluated for the effect on auxiliary saltwater operability until August 4, 2010. In both examples, engineering personnel failed to follow Procedure OM7.ID12, “Operability Determination,” Section 5.1, which required any individual identifying a degraded or nonconforming condition that potentially impacts operability of a system, structure or component to ensure that operations shift management is informed. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notifications 50340417 and 50335847.

The inspectors concluded that the performance deficiency is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. The finding was of very low safety significance (Green) because neither of the two examples was subsequently determined to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming conditions for operability [P.1(c)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the AuxiliarySaltwater System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to maintain adequate design control measures associated with the auxiliary saltwater system. The inspectors identified that the auxiliary saltwater system design did not comply with the plant design bases as described the Final Safety Analysis Report Update. Specifically, an auxiliary saltwater vent line did not meet the requirements established of General Design Criteria 1, “Quality Standards and Records,” and Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants.” The licensee entered the performance deficiency into the corrective action program as Notification 50328942.

This performance deficiency is greater than minor because the design control attribute of the mitigating systems cornerstone and the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Mitigating Systems Cornerstone, the inspectors concluded the finding was of very low significance (Green) because it was a design deficiency confirmed not to result in the loss of operability or functionality. The inspectors concluded that the finding does not have a crosscutting aspect since the performance deficiency is not reflective of current plant performance.
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to maintain adequate design control measures associated with the emergency diesel generating air system. Specifically, failure of non-seismically qualified air compressor unloader sensing lines during a seismic event could impact the safety function of the emergency diesel generators. Subsequent analysis of the nonconforming condition performed by the licensee determined the piping would not fail during a postulated seismic event. The licensee entered this issue into the corrective action program as Notifications 50307496, 50307497, 50307504, 50307670, 50308204, and 50308824.

The finding was more than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones the finding was potentially risk significant for a seismic initiating event requiring a Phase 3 analysis. The analyst estimated the nonrecovery probabilities for operators failing to isolate air between the receiver and the compressor prior to air pressure depletion, and operators failing to manually open fuel transfer valves to makeup to the diesel day tank. The final quantitative result was calculated to be 1.06 x 10-6. However, using a qualitative evaluation of the bounding assumptions, the analyst determined that the best available information indicated that the finding was of very low risk significance (Green). The team determined that the finding was reflective of current plant performance because it had been recently identified during the license renewal inspection and had a human performance crosscutting aspect related to decision making because the licensee did not use conservative assumptions when evaluating this nonconforming condition in previous evaluations.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Proficiency of Operators to Meet the Time Critical Operator Actions**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to ensure that operators are able to implement specified actions in response to operational events and accidents. Specifically, operators could not achieve actions within the analysis time estimates for the cold leg recirculation phase of a loss of coolant accident response and the steam generator tube rupture response as described in the licensee’s safety analysis report.

The finding is more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding represented a potential loss of a safety function requiring a Phase 2 analysis. Because the probability of human error is not effectively addressed by a Phase 2 analysis, a Phase 3 analysis was performed. The senior reactor analyst reviewed the actual timing of the walkdowns associated with the steam generator tube rupture time critical actions. The analyst determined that, while the licensee failed to meet the specific cooldown timing documented in the Final Safety Analysis Report, the total time to start cooling the reactor was well within the total critical timing of the event. The analyst found no impact on safety in delaying the cooldown of the reactor for one minute given that the other time critical actions were performed more quickly than required. Therefore, the analyst determined that this portion of the finding was of very low safety significance because it does not represent an actual loss of safety function (Green). The senior reactor analyst reviewed the issue related to the assumed action times associated with switching over to containment sump recirculation lineup for their emergency core cooling system pumps during a large break loss of coolant accident. The analyst noted that this time critical action was only required if a large-break loss of coolant accident occurred simultaneously with the failure of an residual heat removal pump to stop automatically, requiring local isolation of the pump. Given that the frequency of the initial conditions for the time critical action are below the Green/White threshold, the change in core damage frequency associated with this finding must be of very low safety significance (Green). The team determined that the finding was reflective of current plant performance because the licensee participated in a recent industry-wide study on time critical operator actions, but did not implement any of the group’s recommendations. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process related to verifying the validity of the underlying assumptions used to evaluate the feasibility of operators implementing time critical operator actions.

Inspection Report# : 2010006 *(pdf)*

**Significance: SL-IV** Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Complete and Accurate Information for a Requested License Amendment**
The team identified a noncited violation of 10 CFR 50.9(a), “Completeness and Accuracy of Information” with

multiple examples. Specifically, information supplied to the NRC in License Amendment Request 01-10, dated February 24, 2010, related to the revision of Technical Specification 3.8.1, "AC Sources - Operating," were not complete and accurate in all material respects. Following NRC questioning of the discrepancies the licensee withdrew the amendment request.

The finding is more than minor because the inaccurate information was material to the NRC. Specifically, this information was under review by the NRC to evaluate specific changes to the surveillance requirements associated with the emergency diesel generators. Following management review, this violation was determined to be of very low safety-significance because the amendment request was withdrawn before the NRC amended the facility technical specifications. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement VII, paragraph D.1, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV noncited violation. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not adequately evaluate the extent of condition and take appropriate corrective actions after the NRC identified a similar violation.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Untimely and Inadequate Corrective Actions for the Emergency Diesel Generators**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” with two examples for the failure of the licensee to promptly identify and correct nonconforming conditions related to the emergency diesel generators meeting the design basis. The first example resulted from the failure to identify that instrument inaccuracies were not accounted for in the bounding calculations. The second example involved the failure to identify that the worst case loading calculations exceeded the emergency diesel generator operating load limit.

The failure to promptly identify and correct the design deficiencies associated with the emergency diesel generators was a performance deficiency. This finding is greater than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, “Significant Determination Process,” the team performed a Phase 1 analysis to analyze the significance of this finding and determined the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent an actual loss of safety function of the system or train, did not result in the loss of one or more trains of nontechnical specification equipment, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process or conduct an adequate effectiveness review to verify the validity of the underlying assumptions for a safety-significant decision.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Appropriately Evaluate Failed Residual Heat Removal Surveillance Test**
The team identified a noncited violation of Technical Specification 5.4.1.a for failure to appropriately evaluate and correct a condition adverse to quality, as instructed by Surveillance Test Procedure P-RHR-A22,” Comprehensive Testing of Residual Heat Removal Pump.” Specifically, the licensee failed to recognize a deviation in differential pressure towards the alert range, following the February 9, 2008, comprehensive surveillance test of the 2-2 residual heat removal pump. Continued degradation of the 2-2 residual heat removal pump resulted in failure of the October 9, 2009, comprehensive surveillance test due to the differential pressure exceeding the action limit. The licensee entered this issue into the corrective action program as Notification 50308225.

The finding is more than minor because it was associated with the equipment reliability attribute of the Mitigating Systems Cornerstone and it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating Systems Cornerstone. The finding was determined to be of very low safety significance (Green) because: (1) it was a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The team determined that this finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program, because the licensee failed to appropriately evaluate the 2009 residual heat removal surveillance test failure such that the resolution identified and corrected the cause of the failure.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following Identification of a Non-conservative Technical Specification**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, "Corrective Action," after Pacific Gas and Electric failed to implement prompt corrective actions after identifying a nonconservative technical specification. In December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design basis were met. On January 9, 2009, the licensee entered this condition into the corrective action program. On April 9, 2009, Pacific Gas and Electric concluded that Technical Specification Surveillance Requirement 3.8.1, "AC Sources – Operating," was not adequate to preserve plant safety and applied the provisions of Technical Specification Surveillance Requirement 3.0.3, and Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety." The licensee did not complete the necessary actions to correct the deficient technical specification by submitting an adequate license amendment request. The inspectors concluded the most significant contributor to the finding was a less than adequate engineering evaluation to support the new emergency diesel generator loading profiles following the previous violation. The licensee entered the performance deficiency into the corrective action program as Notification 50232181.

The inspectors determined that the performance deficiency is more than minor because if left uncorrected, the failure to implement prompt corrective actions has the potential to lead to a more significant safety concern. The inspectors concluded the finding was of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. The finding is associated with the Mitigating Systems Cornerstone. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative technical specification such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(B) and after Pacific Gas and Electric failed to submit a required licensee event report within 60 days following discovery of a condition prohibited by the plant technical specifications and a condition that could have prevented the fulfillment of a safety function. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme, required by Technical Specification 3.3.5, "Loss of Power Diesel Generator Start Instrumentation," was inadequate to protect operating engineering safety feature pump motors. The licensee concluded that sustained degraded voltage could result in an overcurrent condition affecting equipment powered from the preferred offsite power supply. This

condition was required to be reported to the NRC because the degraded voltage protection scheme rendered engineered safety feature pumps inoperable for a period in excess of the allowable technical specification out of service time and the condition resulted in the loss of the degraded voltage protection scheme safety function on all three vital 4 kV power buses.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. The inspectors concluded the violation was a Severity Level IV because the licensee failed to submit an adequate licensee event report. The inspectors determined that the violation was also a finding under the reactor oversight process because licensee personnel failed to adequately evaluate a condition adverse to quality for operability and reportability, as required by station procedures. The inspectors concluded that the finding is more than minor because the failure to properly evaluate degraded plant equipment for past operability and reportability could reasonably be seen to lead to a more significant condition. The inspectors concluded that the finding had very low safety significance because the failure to adequately evaluate the condition did not result in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the degraded voltage protection scheme such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 for the licensee's failure to demonstrate that prior NRC approval was not required prior to making changes to the facility degraded voltage protection scheme as described in the Final Safety Analysis Report Update. In response to this violation, the licensee re-performed the corresponding safety analysis to demonstrate that the subject change to the facility degraded voltage protection scheme was consistent with General Design Criteria 17, "Electric Power Systems." The violation is in the licensee's corrective action program as Notification 50306053.

The failure of Pacific Gas and Electric to perform a 10 CFR 50.59 evaluation of modifications to the offsite power protection scheme, in accordance with NEI 96-07, was a performance deficiency. The violation was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The violation screened as very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee did not adopt the requirement to demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that the proposed action was unsafe in order to disapprove the action, in that the Plant Safety Review Committee did not require that a 50.59 evaluation be performed to demonstrate that the proposed action was safe in order to proceed [H.1(b)].
Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Evaluate Changes to the Diesel Testing as Described in the Final Safety Analysis Report Update**
The team identified two examples of a Severity Level IV noncited violation of 10 CFR 50.59 after the licensee failed to perform an adequate evaluation to demonstrate that prior NRC approval was not required before making changes to the frequency and voltage recovery criteria and to the diesel testing commitments as described in the Final Safety Analysis Report Update. Specifically, the 1998 Final Safety Analysis Report Update identified a change from Safety

Guide 9 to Regulatory Guide 1.9, Revision 2. The scope involved the removal of the KWS delay and included new requirements for voltage and frequency response. This resulted in a reduction in acceptance criteria. The team also identified a second example where the licensee failed to evaluate the 2005 Final Safety Analysis Report Update change from Regulatory Guide 1.9, Revision 2 to Revision 3 for diesel testing and interval frequency. Using NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, the team concluded that these changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases. In addition, the licensee's 50.59 evaluation, for DCP E-049425, Revision 0, "EDG Starting, and Loading Capability" was less than adequate to conclude that prior NRC approval was not required for the changes. The licensee has entered these issues into their corrective action program as Notification 50302481.

The failure of Pacific Gas and Electric to perform an adequate 10 CFR 50.59 evaluation prior to changing the facility as described in the Final Safety Analysis Report Update is a performance deficiency. The violation was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The violation screened as very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The finding has a crossuctting aspect in the area of problem identification and resolution associated with the corrective action program component. In License Amendment Request 10-01, dated February 24, 2010, the licencee did not thoroughly evaluate the original problem of using the 10 CFR 50.59 evaluation process to justify using Regulatory Guide 1.9, Revision 2, Section C, Position 4, as an exception to meeting the frequency and voltage criteria identified in Safety Guide 9 [P.1(c)].
Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-Conservative Decision Making resulted in a Violation of Technical Specification**
On March 10, 2010, the inspectors identified a noncited violation of Technical Specification 3.7.7, "Vital Component Cooling Water System," after both Unit 2 component cooling water loops were inoperable longer than permitted during power operations. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme was inadequate to ensure minimum required voltage would be available to operating engineered safety feature pumps during a degraded offsite power grid. The licensee concluded that operating pumps could trip and lock out on over current before the protection scheme would automatically transfer power to the emergency diesel generators. The licensee declared the 230kV offsite power systems inoperable and took compensatory actions to enable the automatic transfer of busses with operating engineered safety feature pumps directly to the diesel generators following a unit trip. On March 10, 2010, the inspectors identified that operating component cooling water pump 2-3 was still aligned to automatically transfer to 230kV offsite power source following a unit trip. The licensee had previously removed component cooling water pump 2-2 from service for maintenance on March 7, 2010. Technical Specification 3.7.7, "Vital Component Cooling Water System," required a minimum of two operable component cooling water pumps to establish operability of a vital component cooling water loop. Contrary to Technical Specification 3.7.7, on March 10, 2010, the licensee operated Unit 2 without an operable vital component cooling water loop for greater than 14 hours. The licensee has entered this issue into their corrective action program as Notification 50304802.

Either the failure of Pacific Gas and Electric to restore at least two operable component cooling water pumps or to have placed Unit 2 in Mode 3 within six hours, as required by plant Technical Specification 3.7.7, was a performance deficiency. The performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance, of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences (i.e., core damage), and it was within the licencee's ability to correct this problem. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the finding because the violation represents the actual loss of safety function for greater than the technical specification allowed outage time. The finding was of very low safety significance (Green) based on a bounding qualitative evaluation using Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," The inspectors based this conclusion on the low probability of an actual degraded grid condition coincidental with an accident or anticipated operational occurrence during the 14-hour exposure that the vital component cooling water loops were unavailable due to the performance deficiency. The inspectors concluded that this finding had a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the degraded voltage protection scheme [H.1(b).

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with the Current Plant Design Bases**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to include the current plant design basis for the 230kV degraded voltage protection scheme in the Final Safety Analysis Report Update. Title 10 CFR 50.71 (e) states in part, "Each person licensed to operate a nuclear power reactor shall update periodically, as provided in paragraphs (e) (3) and (4) of this ection, the Final Safety Analysis Report Update originally submitted as part of the application for the operating license, to assure that the information included in the report contains the latest information developed. Contrary to the above, on March 14, 2010, the inspectors identified that Pacific Gas and Electric failed to update the Final Safety Analysis Report Update to include complete design basis information for the offsite degraded voltage protection scheme. The inspectors identified that Final Safety Analysis Report Update did not include the design basis for the allowance time delay or the limiting voltage setpoints. The licensee has entered this issue into their corrective action process as Notification 50313763.

Failure to include the current plant design basis for the 230kV degraded voltage protection scheme in the Final Safety Analysis Report Update is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this issue was to be ealuated using the traditional enforcement process because the performance deficiency was a failure to meet a requirement or standard, had the potential for impacting the NRC's ability to perform its regulatory function, and the concern was within the licensee's ability to foresee and correct and should have been prevented. The team used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I "Reactor Operations," dated January 14,2 005, to evaluate the significance of this violation. The team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation because the erroneous information was not used to make any unacceptable change to the facility or procedures. Using Inspection Manual Chapter 0609.04, “Phase 1 Initial Screening and Characterization of Findings,” the team concluded that the issue screened as having low safety significance (Green) under the Significance Determination. Because this violation is of very low safety significance and was entered into the licensee’s corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process and had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of the condition and take appropriate corrective actions after the NRC identified a similar violation [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination Associated With the Offsite Degraded Voltage Protection Scheme**
On February 27, 2010, the inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V,

“Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to complete an adequate operability evaluation, as required by Procedure OM7.ID12, “Operability Determination,” Revision 14. The inspectors identified that the offsite power degraded voltage protection scheme time delay was inconsistent with key assumptions in the accident analysis. The licensee entered this nonconforming condition into the corrective action program as Notification 50301167 on February 24, 2010. Plant operators subsequently requested plant engineering to perform an operability determination of the nonconforming condition per Operability Determination Procedure OM7.ID12. On February 27, 2010, the plant operating authority concluded that the protection scheme was operable based on the information provided in the operability determination. Contrary to the above, on March 2, 2010, the inspectors concluded that the licensee’s operability determination was inadequate to demonstrate protection scheme operability and was not performed as required by Operability Determination Procedure OM7.ID12. Plant engineering only addressed the capability of the protection scheme at normal grid voltage following a mechanical failure of the 230 kV load tap changer. Operability Determination Procedure OM7.ID12, Section 5.3, “Write the Prompt Operability Assessment (POA),” required that the licensee address the potential effect of the nonconforming condition to perform the specified safety function. The licensee has entered this finding into the corrective action program as Notification 50319258.

Failure to complete an adequate operability evaluation, as required by Procedure OM7.ID12, “Operability Determination,” Revision 14, is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality, and the failure to perform an adequate operability evaluation affects the ability to ensure operability of the protection scheme at normal grid voltage following a mechanical failure of the 230 kV load tap changer. The inspectors used Inspection Manual Chapter 609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the finding because the violation represent the actual loss of safety function for greater than the Technical Specification 3.3.5 allowed outage time. Using Appendix M, of the “Significance Determination Process Using Qualitative Criteria,” the inspectors concluded that the finding was of very low safety significance (Green) based on a bounding qualitative evaluation. The inspectors based this conclusion on the low probability of an actual degraded grid condition coincidental with an accident or anticipated operational occurrence during the exposure time that protection scheme was available due to the performance deficiency. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition for operability and reportability [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Second Level Undervoltage Relay Time Delay to Initiate Load Shed and Sequencing Upon the Diesel Generator is Adequate to Assure Plant Safety**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure that plant conditions were consistent with design calculation inputs and assumptions. The licensee failed to assure and verify that Technical Specification 3.3.5 (SR3.3.5.3) pertaining to the second level undervoltage relay time delay to initiate load shed and sequencing upon the diesel generator was adequate to assure plant safety. Supplemental Safety Evaluation Report 09, Section 8.1, requires that a second level of under voltage protection for the onsite power system be provided. Subsection (1)(c)(i), reads: “The allowable second level undervoltage relay time delay, including margin, shall not exceed the maximum time delay that is assumed in the Final Safety Analysis Report Update accident analyses.” Contrary to the above, as of March 4, 2010, the licensee failed to adequately implement the requirements of Supplemental Safety Evaluation Report 09. The second level undervoltage relay time delay setpoint for the emergency diesel generator of less than or equal to 20 seconds, assuming a safety injection signal concurrent with a degraded off site power source, exceeded the Final Safety Analysis Report Update accident analysis. This item is identified in the licensee’s corrective action document Notification 50301167.

Failure to ensure that plant conditions were consistent with design calculation inputs and assumptions is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix E, Section 3 Example j, the violation was determined to be more than minor because the engineering calculation error results in a condition where there is now a reasonable doubt on the operability of a system or component. These deficiencies represented reasonable doubt regarding the mitigation of an accident by being in an unanalyzed condition. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 2, the finding was determined to have very low safety significance (Green), did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. The team reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Drawings and Procedures to Align Emergency Makeup Water Supply from Diablo Canyon Creek to Support the Auxiliary Feedwater System**
The team identified a noncited violation of Diablo Canyon Technical Specification 5.4.1. "Procedures," for failure to have a procedure. The Diablo Canyon Final Safety Analysis Report Update, Revision 18, Section 6.5.2.1.1 documents the design of the auxiliary feedwater system and credits eight sources of water that can provide backup means of supply in the event that its primary source of water, the condensate storage tank, becomes exhausted. One of the sources included is the Diablo Canyon Creek. Diablo Canyon Technical Specification 5.4.1 states: "Written procedures shall be established, implemented, and maintained covering the following activities: [a.] The applicable procedures recommended in NRC Regulatory Guide 1.33, Revision 2, Appendix A, February 1978". NRC Regulatory Guide 1.33, Revision 2, Appendix A, describes procedures under Items 3l (instructions for shutdown, startup, and operation, including system filling, of the auxiliary feedwater system) and 6j (loss of feedwater system or feedwater system failure). Contrary to Technical Specification 5.4.1, on March 4, 2010, the team identified that the licensee did not have an established procedure for accomplish the task identified in the Final Safety Analysis Report Update, Section 6.5.2.1.1 for taking water from the Diablo Canyon Creek to be a supply for the auxiliary feedwater system. This item is identified in the licensee's corrective action document Notification 50298563.

Failure to provide a procedure or instructions and acceptance criteria to perform an emergency makeup water alignment to the auxiliary feedwater system is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality, and the lack of having this procedure affects the ability to ensure the availability, reliability, and capability of the auxiliary feedwater system to respond to initiating events to prevent undesirable consequences, (i.e., core damage, and it was within the licensee's ability to correct this problem.) Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding was determined to have very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. The team concluded that this finding had a crosscutting aspect in the area of problem identification and resolution, in that the licensees' corrective action program thoroughly evaluates problems such that the resolutions address causes and the extent of conditions, as necessary. Per licensee Notification 50298563, changes were made to pumping systems associated with the Diablo Canyon Creek in 2007, which affected the ability to pump water through the discussed credited lineup supporting the auxiliary feedwater system. This effect was not identified as part of the changes, so no review of procedures related to the emergency auxiliary feedwater system alignment in question was performed. Since these actions occurred within the last three years, this performance characteristic reflects current performance [P.1 (c)].

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Feedwater Rupture Accident Analysis in the Final Safety Analysis Report Update**
The team identified a Severity Level IV noncited violation of 10 CFR 50.71, "Maintenance of records, making of reports." Paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." In the Diablo Canyon Final Safety Analysis Report Update section addressing the feedwater line break accident, it states that operator actions are credited with precluding

the operation of pressurizer safety valves based on determinations in Westinghouse study WCAP-11667 (1998) (Final Safety Analysis Report Update, Section 15.4.2.2.2). Review of this study, and associated correspondence on the topic during 2006 indicated that the Westinghouse study did not state that operator actions could be credited for this event, but analysis of the worst case pressurizer overfill accidents by the licensee may show that this is the bounding case for such accidents, and that it did not need to be addressed in the feedwater line break analysis. In 2006, the licensee indicated that they would revise the Final Safety Analysis Report Update text to remove this reference to the Westinghouse study, which had been in the Final Safety Analysis Report Update since Revision 16. Contrary to the above, since 2006 (Final Safety Analysis Report Update Revision 16), the licensee failed to update Final Safety Analysis Report Update, Section 15.4.2.2.2. The licensee has entered this issue into their corrective action process as Notification 50301747.

Failure to periodically update the Final Safety Analysis Report Update with a known error is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this issue was to be evaluated using the traditional enforcement process because the performance deficiency was a failure to meet a requirement or standard, had the potential for impacting the NRC's ability to perform its regulatory function, and the concern was within the licensee's ability to foresee and correct and should have been prevented. The team used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, "Reactor Operations," dated January 14, 2005, to evaluate the significances of this violation. The team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The inspectors reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Text to Reflect Credited Design Class I Makeup Flowpath to Component Cooling Water Expansion Tank in the Final Safety Analysis Report Update**
The team identified a Severity Level IV noncited violation of 10 CFR 50.71, "Maintenance of records, making of reports." Title 10 CFR 50.71, paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." In the Final Safety Analysis Report Update Table 9.2-7, Component 5, it states, "This 250 gpm, Design Class I, makeup water flowpath, described under Makeup Provisions in Subsection 2.3.3 (Section 9.2.2.3.3), can be started within 10 minutes." Final Safety Analysis Report Update, Section 9.2.2.3.3 states, "All piping and valves in the makeup path from the condensate storage tank (including their cross-connections) and the firewater tank, through the makeup water transfer pumps up to and including the makeup valves on the component cooling water system lines, are Design Class I." Text later in the section implied that the flow path from the firewater tank was not Design Class I. Review by the licensee staff revealed that the only Design Class I flow path to provide makeup to the component cooling water expansion tank is via the condensate storage tank. This revealed that the text provided in Final Safety Analysis Report Update, Section 9.2.2.3.3 stating that both the condensate storage tank and firewater tank makeup paths are credited is incorrect. Contrary to above, since 1984 (Final Safety Analysis Report Update, Revision 0), the licensee did not update Final Safety Analysis Report Update, Section 9.2.2.3.3 to correct the error of including firewater as a possible makeup path to the component cooling water expansion tank. The licensee has entered this issue into their corrective action process as Notification 50301884.

Failure to periodically update the Final Safety Analysis Report Update with a known error is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this performance deficiency was to be evaluated using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. Using General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, Reactor Operations, dated January 14, 2005, to evaluate the significances of this violation, the team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement

I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The team reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Effectively Implement the Seismically-induced Systems Interaction Program**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric personnel failed to effectively implement the Seismically Induced System Interaction Program. The Seismic Interaction Program is part of the design basis mitigation strategy for a potential 7.5 magnitude Hosgri earthquake and is required by Procedure AD4.ID3, "SISIP Housekeeping Activities." The inspectors identified three examples of transient equipment and materials improperly staged in seismically induced system interaction target areas. Pacific Gas and Electric had not analyzed the transient equipment to assess the risk to safety related components as required by plant procedures. Pacific Gas and Electric entered this finding into the corrective action program as Notification 50299740.

The finding is more than minor because the failure to follow the Seismically Induced System Interaction Program is associated with the Mitigating Systems Cornerstone external events protection attribute and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding had very low safety significance because none of the examples of improperly staged equipment resulted in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. The inspectors concluded this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee's past actions to address Seismically Induced System Interaction Program deficiencies were not effective [P.1(d)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report with the Current Plant Design Bases**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with the current design basis. The inspectors identified that the current Final Safety Analysis Report Update, Revision 18, Sections 3.1, 6.4, 6.5, and 9.4 did not capture the current design basis for the control room, component cooling water, and auxiliary feedwater systems. The failure of the licensee to provide current design basis information in the Final Safety Analysis Report Update had an adverse impact on the plant modification process, the licensee's ability to assess operability for degraded plant systems, and the NRC's ability to ensure that regulatory requirements were met. The licensee entered this violation into the corrective action program as Notifications 50308588, 50306131, 5030799, and 50307476.

The inspectors evaluated this violation using the traditional enforcement process because the issue affected the NRC's ability to perform its regulatory function. The inspectors concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. The inspectors concluded the violation is Severity Level IV because the erroneous information was not used to make an unacceptable change to the facility or procedures that would have resulted in greater than very low safety significance under the Significance Determination Process. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of condition of previous similar violations and take appropriate corrective actions [P.1(c)].

Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovering a condition that could have prevented the fulfillment of a safety function. On November 22, 2005, the licensee determined that plant operators may not have had the capability to align either residual heat removal train to the cold leg recirculation mode of emergency core cooling following certain small break loss of coolant accidents. Plant engineers determined that the residual heat removal containment sump suction valve operators were inadequately sized to open against the differential pressure generated by the pumps operating in recirculation for an extended period. Plant engineers identified this condition during a follow up of industry operating experience. The licensee initially concluded that the condition was not reportable because the operating experience was not applicable to Diablo Canyon. The licensee failed to re-screen the issue for reportability after determining that the plant was susceptible to the condition. The licensee entered this issue into the corrective action program as Notifications 50301839 and 50295784.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded the violation was a Severity Level IV because the licensee failed to submit a required licensee event report. The inspectors did not assign a crosscutting aspect because the performance deficiency represented a latent issue.
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation Following the Failure of Both Motor-Driven Auxiliary Feedwater Trains**
The inspectors identified a noncited violation of 10 CFR, Part 50, Appendix B, Criteria XVI, "Corrective Actions," after Pacific Gas and Electric failed to implement adequate corrective actions following a protection system failure. On June 29, 2009, a protection system card failure resulted in the inoperability of both motor-driven auxiliary feedwater trains. The licensee concluded that the failure of the auxiliary feedwater trains were expected as part of the protection system design and limited corrective actions to replacing the failed card. The inspectors concluded that the protection system design did not meet the design basis, which required that no single active failure would prevent the auxiliary feedwater system from meeting the safety function. The licensee entered this issue into the corrective action program as Notifications 50251823, 50298491 and 50254412.

The inspectors concluded that the finding is greater than minor because the vulnerability of auxiliary feedwater to a single failure is associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding to have very low safety significance because the condition did not represent a loss of system safety function. While the single failure of the protection system card resulted in the inoperability of both motor-driven auxiliary feedwater trains, the turbine-driven auxiliary feedwater train was available to perform the safety function. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the auxiliary feedwater failure such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit a Licensee Event Report following the Common-Cause Failure of Independent Trains or Channels**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a

required licensee event report within 60 days after discovery of a common-cause failure of three control room radiation monitors. The inspectors concluded that monitors failed on October 13, 2009, as a result of water intrusion due to heavy rains. The inspectors concluded that common cause failure of the radiation monitors was reportable under 10 CFR 50.73(a)(2)(vii). Pacific Gas and Electric subsequently reported the event on February 17, 2010, as Licensee Event Report 2010-001-00, Control Room Ventilation Pressurization Due to Radiation Detector Failures. The licensee entered this issue into the corrective action program as Notification 50301839.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded that this was a Severity Level IV noncited violation because the licensee failed to submit a required licensee event report. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the radiation monitor failures to ensure NRC reportability requirements were met [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Design and Configuration Control Requirements**
The inspection team identified a noncited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, which requires licensees to implement measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These design control measures include verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. Specifically, on February 16, 2008, plant engineering personnel failed to implement the design control process for a modification to the Unit 2 residual heat removal containment sump valves when they inappropriately used maintenance procedures to reduce the valve stroke lengths from 15.5 to 13.8 inches. The invalid design change resulted in the inoperability of both emergency core cooling trains between April 8, 2008, (when the plant entered Mode 4) and October 22, 2009. The reduced sump valve stroke length also caused a portion of the sump valve disc to remain in the residual heat removal suction flow path, reducing the available residual heat removal pump net positive suction head. The licensee entered this condition into their corrective action program as Notification 50277252.

The inspection team concluded that the failure of plant engineering to use the design control process was a performance deficiency within the licensee's ability to foresee and correct. The finding is more than minor because it affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding required a Phase 2 analysis because the finding represented the loss of a safety system function. The Phase 2 analysis determined that this finding was potentially greater than Green; therefore, a Phase 3 analysis was completed by a regional senior reactor analyst. The Phase 3 analysis determined that this issue was of very low safety significance (Green), owing principally to the fact that operators could have opened the affected valves locally with a very high probability of success. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the failure of the valves to meet the specified stroke time to ensure that the resolution fully addressed the causes and extent of condition, as necessary [P.1(c)].

Inspection Report# : 2009009 *(pdf)*

**Significance:** G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Conduct an Adequate Post-Modification Test**
The inspection team identified a noncited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, which requires that a test program be established to assure that all testing required to demonstrate that structures, systems,

and components will perform satisfactorily in service. Specifically, the licensee failed to perform testing to assure that the interlock circuitry associated with the residual heat removal containment sump valves (SI-2-8982A and B) would perform satisfactorily in service following a modification on February 16, 2008, that changed the stroke lengths. As a consequence, remote operation of the valves needed to initiate high pressure recirculation was lost for an entire operating cycle. The licensee entered this issue into their corrective action program as Notification 50277252.

The failure to establish adequate post-modification testing requirements was a performance deficiency within the licensee's ability to foresee and correct. The finding is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding required a Phase 2 analysis because the finding represented the loss of a safety system function. The Phase 2 analysis determined that this finding was potentially greater than Green; therefore, a Phase 3 analysis was completed by a regional senior reactor analyst. The Phase 3 analysis determined that this issue was of very low safety significance (Green), owing principally to the fact that operators could have opened the affected valves locally with a very high probability of success. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee failed to implement a corrective action program with a threshold sufficient to identify issues associated with the failure to meet sump valve post-modification test acceptance criteria [P.1(a)].

Inspection Report# : 2009009 *(pdf)*

**Significance: SL-IV** Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility as Described in the Final Safety Report Update Associated with the Addition of Manual Actions in the Safety Analysis**
The inspection team identified a noncited violation of 10 CFR 50.59, which states that a licensee may make changes to the facility as described in the final safety analysis report without obtaining a license amendment if the change does not result in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases or in the safety analyses. This regulation further requires the licensee to include a written evaluation providing the basis for concluding that a license amendment is not required. On November 21, 2005, the licensee failed to provide a written evaluation concluding that a license amendment was not required for a change to the facility as described in the final safety analysis report. Specifically, the licensee identified a condition where large differential pressure across the residual heat removal suction containment sump valves could cause them to fail to open during certain small break loss of coolant accidents. On October 5, 2005, the licensee revised Emergency Operating Procedure E-1, "Loss of Reactor or Secondary Coolant," to add an operator action to align component cooling water to the residual heat removal heat exchanger. On June 16, 2009, the licensee again revised Emergency Operating Procedure E-1 to specify that operator action to align component cooling water within 30 minutes was a time critical operator action. The licensee did not evaluate either change to determine if prior NRC approval was required for the new manual actions. The licensee entered this issue into their corrective action program as Notification 50276288.

The failure of the licensee to perform a 10 CFR 50.59 evaluation of a new manual action supporting the plant's design basis was a performance deficiency within the licensee's ability to foresee and correct. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. The inspectors concluded that the issue was more than minor because of a reasonable likelihood that the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609.04, "Phase 1 Initial Screening and Characterization of Findings." The inspectors concluded that the issue affected the Mitigating Systems Cornerstone and screened Green because the finding was a design or qualification deficiency confirmed not to result in loss of operability. The issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1(c)].

Inspection Report# : 2009009 *(pdf)*

**Significance: SL-IV** Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate 50.59 Evaluation for Steam Generator Tube Rupture Analysis**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 after Pacific Gas and Electric failed to perform an adequate evaluation of a change to the facility as described in the Final Safety Analysis Report Update. In 1992, the licensee identified that auxiliary feedwater and steam generator power-operated relief valve flow rates assumed in the steam generator tube rupture accident analysis were non-conservative. To address the non-conforming condition, Pacific Gas and Electric changed the accident analysis to include a new time critical operator action to terminate turbine-driven auxiliary feedwater flow 5.54 minutes after the reactor trip and credit motor driven auxiliary feedwater automatic level control to the ruptured steam generator. The licensee did not perform a 10 CFR 50.59 safety evaluation of these changes. The NRC basis of approval of the accident analysis include four time critical operator actions, each assumed to occur after the first 10 minutes following the accident. The inspectors concluded that NRC approval was required before the licensee added the new time critical manual action under the 10 CFR 50.59 Rule in effect at the time because the change reduced the margin to safety to the basis of Technical Specification 3.7.4, “10% Atmospheric Dump Valves.” The inspectors also concluded that prior NRC approval was required under the current 50.59 Rule because the change result in a departure from a method of evaluation described in the Final Safety Analysis Report Update. The performance deficiency, a less than adequate 50.59 evaluation, was the result of a latent issue. However, the inspectors concluded that the licensee had reasonable recent opportunities to identify the problem. The inspectors also concluded that plant programs, processes or organizations have not changed such that the problem would not reasonably occur today and that the most significant contributor to the performance deficiency was reflective of current plant performance. The licensee entered this issue into their corrective action program as Notification 50270786.

The failure of Pacific Gas and Electric to perform a 10 CFR 50.59 evaluation of the changes to the steam generator tube rupture accident analysis was a performance deficiency. The inspectors evaluated this issue using traditional enforcement because the performance deficiency had the potential for impacting the NRC’s ability to perform its regulatory function. The issue was more than minor because of reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609.04, “Phase 1 Initial Screening and Characterization of Findings.” The finding affected the Mitigating Systems Cornerstone because the change described the operator actions required to mitigate steam generator tube rupture accident. The inspectors concluded the finding screened Green because the finding was a design deficiency that did not result in the loss of operability or functionality. The inspectors concluded that the violation was a Severity Level IV because the issue screened Green under the Significance Determination Process. The inspectors concluded that this finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the steam generator tube rupture analysis such that the resolutions addressed causes and extent of condition [P.1(c).

Inspection Report# : 2009005 *(pdf)*

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## Barrier Integrity

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Work Planning Resulted in the Release of Two Gas Decay Tanks**
The inspectors identified a self-revealing noncited violation of Technical Specification 5.4.1, “Procedures,” after Pacific Gas and Electric inadvertently released the contents of two gas decay tanks into the auxiliary building. Gas Decay Tank 2-2 was in “purge mode.” On October 11, 2009, plant operators were implementing an equipment control clearance to drain the emergency core cooling systems. A second group of operators were implementing a core offload master clearance. The parallel implementation of both equipment clearances resulted in Gas Decay Tank 2-2

to be vented into the auxiliary building. The auxiliary building operator received a low gas header pressure alarm after the pressure dropped to 15 psig. Per procedure, the operator aligned Gas Decay Tank 2-3 to "purge" mode. As a result, the second gas decay tank was released into the auxiliary building through the open vent path. The inspectors concluded that the radiological consequence of the event did not result in a potential for overexposure because the reactor had been shutdown since October 3, 2009.

The inspectors concluded that the failure to properly implement the core offload master equipment control clearance was a performance deficiency. The finding is more than minor because the performance deficiency could be reasonably viewed as a precursor to a significant event. The inspectors determined the finding to have very low safety significance because the performance deficiency only represented a degradation of the auxiliary building radiological barrier function. This finding has a crosscutting aspect in the area of human performance associated with the work control component because the licensee did not adequately plan and coordinate the two clearance activities or fully consider the impact the work had on different job activities and the need for the two work groups to maintain interfaces [H.3(b)].

Inspection Report# : 2009005 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2009
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Properly Plan a Maintenance Activity**
The inspectors reviewed a self-revealing, noncited violation of Technical Specification 5.4.1(a) for failure to properly plan numerous outage maintenance activities including the disassembly of the Unit 2 reactor head. Specifically, Work Orders 68004363 (disassembly of the old head) and 68003988 (scaffolding activities) were not properly planned, thereby requiring those maintenance activities to be changed and/or repeated, which resulted in increased radiation exposure. Radiation Work Permits 09-2233 and 09-2237 for the disassembly of the Unit 2 old reactor vessel closure head and supporting activities during Refueling Outage 15 had an initial combined dose estimate of 5.869 rem and 1102 man-hours. However, the job ended with an actual combined dose of 17.378 rem and 1882 man-hours, which exceeded the initial dose estimate by 296 percent. The overarching reason for exceeding the original dose estimate was improper planning and control for the maintenance, which increased the man-hours to complete the task by 170 percent. The licensee entered this deficiency in the corrective action program as Notification 50275107 and plan to perform an apparent cause evaluation.

The failure to properly plan maintenance activities is a performance deficiency. This finding is greater than minor because it affected the Occupational Radiation Safety cornerstone attribute of Program and Process in that the inadequate ALARA planning caused increased collective radiation dose for the job activity to exceed 5 person-rem and the planned dose by more than 50 percent. Using the Occupational Radiation Safety Significance Determination Process, the inspector determined this finding to be of very low safety significance because although it involved ALARA planning and controls, the licensee's latest rolling three-year average does not exceed 135 person-rem per unit. Furthermore, the finding had an associated human performance cross-cutting aspect in the work control component because the licensee did not fully incorporate job site conditions, plant structures, systems, and components, as well as human-system interface and the need for planned contingencies to maintain doses ALARA [H.3(a)].

Inspection Report# : 2009005 *(pdf)*

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# Public Radiation Safety

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# Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

Last modified : January 06, 2011

# Diablo Canyon 2
# 4Q/2010 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain the integrity of Door 155 in the rated condition. On December 9, 2010, the inspectors identified that the fire door was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain Door 155 in a configuration that would provide at least a 1½-hour rated fire barrier. The inspectors previously identified that Door 155 was degraded as a fire barrier in 2009. The licensee entered the violation into the corrective action program as Notification 50367381 and took immediate corrective actions to restore the fire barrier to the rated condition and to implement weekly plant fire door walkdowns.

The inspectors concluded that the finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute and objective to prevent undesirable consequences due to fire. The inspectors determined that the finding was within the fire confinement category and that the fire barrier was moderately degraded. The inspectors concluded that the finding was of very low safety significance (Green) because there was a non-degraded automatic full area water-based suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take effective corrective actions to following the previous occurrence of the violation [P.1(d)].

Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Transient Combustibles Procedure**
The inspectors identified a noncited violation of Diablo Canyon Unit 2 Facility Operating License Condition 2.C.(5), “Fire Protection,” after Pacific Gas and Electric failed to ensure procedures for controlling flammable and combustible materials adequately incorporated requirements of the fire hazard analysis. On October 18, 2010, the inspectors identified that transient combustible materials staged in the Unit 1 12 kilovolt switchgear room did not have an approved transient combustibles permit. The licensee stated that the combustibles permit procedure did not require a permit for the room while Unit 1 was shutdown. However, the plant fire hazards design basis described safe shutdown equipment in the room that would be needed to support a safe shutdown of the operating unit, specifically the Unit 2 startup bus located in the room. The inspectors determined that the licensee’s transient combustibles permit procedure was inadequate because the procedure did not require a permit for the Unit1 12 kilovolt switchgear room when Unit 2 was operating. The licensee entered the issue into the corrective action program as Notification 50366302 and performed an evaluation of the transient combustibles stored in the area.

The inspectors concluded that this finding was more than minor because it affected the Mitigating Systems Cornerstone external factors attribute objective to limit the likelihood of those events that upset plant stability and

challenge critical safety functions. The inspectors determined that the finding was within the fire prevention and administrative controls category and represented a low degradation level due to the minimal impact on the effectiveness and reliability of the affected systems. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening, the low degradation rating, and only equipment needed to reach and maintain cold shutdown conditions was affected. This finding had a crosscutting aspect in the area of human performance associated with the resources component because the licensee failed to ensure that the design documentation adequately identified the Unit 2 startup bus as equipment required for safe shutdown for Unit 2 [H.2 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determinations**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to adequately evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, "Operability Determination." On October 15, 2010, the inspectors identified a less than adequate technical evaluation supporting Prompt Operability Assessment 50350918, "Unit 2 - Insulation in Bio-Wall Penetration." Engineering personnel failed to adequately evaluate the extent of condition after technicians identified about 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous insulation in the Unit 1 reactor coolant loop biological shield wall penetrations. This fibrous material could have potentially been transported and plugged the emergency core cooling containment sump screen. The licensee performed the prompt operability assessment for Unit 2, which was operating at the time. The inspectors concluded that the engineering personnel inappropriately applied the leak-before-break methodology to exclude about 87 percent of this material from the extent of condition review in the prompt operability assessment.
The second example involved Prompt Operability Assessment Notification 50355265, "RHR Sump Margin," which was completed by the licensee on October 23, 2010. In this example, engineering personnel failed to identify and demonstrate that the specified safety function of the refueling water storage tank could be maintained as required by the plant operability procedure. The inspectors identified that the post accident flow path from the reactor cavity to the containment sump was blocked by a large shield plug. This blockage
reduced the amount of post accident inventory available at the containment sump at the time of transition from injection to recirculation mode of emergency core cooling operation. Engineering personnel failed to demonstrate that the safety function to ensure full sump submergence was maintained with the blocked flow path. Full submergence of the sump was used by the NRC as the basis for approval of Technical Specification 3.5.4, "Refueling Water Storage Tank," inventory requirements. The licensee entered the violation into the corrective action program as Notification 50369117 and revised the prompt operability assessments using assumptions consistent with the current licensing bases.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because Pacific Gas and Electric did not use conservative assumptions in decisions to demonstrate component operability in either example [H.1(b)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Containment Recirculation Sump Design Control**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to ensure Calculation STA-255, "Minimum Required Refueling Water Storage Tank Level for GE Sumps," Revision 2, demonstrated adequate available refueling water storage tank inventory. On October 19, 2010, the inspectors identified that emergency core cooling post accident flow path from the reactor

cavity to the containment sumps was blocked by a large steel plug on Unit 1. The accident analysis assumed this 35 square foot path was open to allow coolant from a pipe break inside the biological shield to communicate with containment sumps during the recirculation mode of emergency core cooling. The licensee credited the inventory from the reactor cavity when determining the minimum required refueling water storage tank volume in Calculation STA-255. Pacific Gas and Electric used Calculation STA-255 as the basis for determining the minimum required refueling water storage tank volume specified by Technical Specification 3.5.4, "Refueling Water Storage Tank." The inspectors identified that the recirculation flow path was also blocked on Unit 2. The inspectors concluded that the most significant contributor to the violation was inaccurate plant drawings used by plant engineers during the performance of Calculation STA-255. The licensee's corrective actions included completion of a prompt operability assessment justifying continued operation of Unit 2 and replacement of the shield plug with a movable platform on Unit 1 prior to plant restart.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone plant modification design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the performance deficiency involved a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the resources component because Pacific Gas and Electric failed to use complete, accurate and up-to-date drawing for Calculation STA 255 [H.2(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Emergency Diesel Generator Surveillance Testing**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," after Pacific Gas and Electric failed to develop and implement an adequate testing program for the emergency diesel generators that met design requirements and recommendations. Specifically, in December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design bases were met. Pacific Gas and Electric updated the load calculations, but failed to make the necessary revisions to Surveillance Test Procedure STP M-9D1, "Diesel Generator Full Load Rejection Test." As a result, Pacific Gas and Electric failed to test several of the emergency diesel generators at the complete load as required by Regulatory Guide 1.108, Revision 1, which is part of the current licensing bases. The licensee entered this into the corrective action program as Notification 50368801, determined there was no loss of safety function for the affected components, and applied the provisions of Surveillance Requirement 3.0.3 for a missed surveillance test. The inspectors concluded the most significant contributor to the finding was less than adequate diesel generator loading evaluations to support corrective action from previous violations associated with the emergency diesel generator testing.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the equipment control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative surveillance test such that the resolution addressed the fundamental basis for the surveillance [P.1 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Quality Verification Audits**
The inspectors identified a noncited violation of 10CFR Appendix B, Criterion XVIII, "Audits", which required that a comprehensive system of planned and periodic audits be carried out to verify compliance with all aspects of the

quality assurance program and to determine the effectiveness of the program as well as follow up action, including re-audit of deficient areas, where indicated. Contrary to this requirement, Pacific Gas and Electric failed to ensure that a comprehensive system of planned and periodic audits were carried out to verify compliance with all aspects of the quality assurance program, determine the effectiveness of the program, and perform necessary follow up actions. Specifically, the 2008 Quality Verification audit of the corrective action program failed to adequately address an adverse trend in the problem evaluation process documented in NRC Inspection report 2008005, which identified eleven examples of an adverse trend in problem evaluation. The licensee entered this into their corrective action program as Notification 50365083 and determined
there was no loss of safety function for the affected components. The inspectors concluded the most significant contributor to the finding was a less than adequate evaluation of the corrective action trending program.

This finding was more than minor because it was associated with the equipment control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the performance deficiency was of very low safety significance (Green) it was a deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to coordinate and communicate the results from assessments to affected personnel, and track the corrective actions to address issues commensurate with their significance [P.3(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify a Degraded Fire Barrier**
The inspectors identified a noncited violation of the Diablo Canyon Facility Operating License Condition (5), "Fire Protection," after Pacific Gas and Electric failed to maintain the integrity of a fire door in the rated configuration. On August 19, 2010, the inspectors identified that Fire Door 223 was inoperable. Fire Door 223 was required to provide a 3-hour rated barrier between Fire Areas 5-A-4 and 5-B-4. A fire in either of these areas could have prevented operation of the auxiliary feedwater, auxiliary saltwater, or component cooling water pumps or steam generator level control from the remote shutdown panel. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to either maintain Fire Door 223 operable or implement compensatory actions within one hour. The inspectors concluded the most significant contributor to the finding was that licensee personnel did not identify and enter the degraded fire door into the Corrective Action Program. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notification 50336901 and completed repairs to the door on August 23, 2010.

The inspectors concluded that the performance deficiency was more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door was a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated fire barrier function. The inspectors concluded the finding was of very low safety significance because the degraded barrier would have provided a minimum of 20 minutes fire endurance protection and ignition sources and combustible materials were positioned that had a fire spread to secondary combustibles, the degraded barrier would not have been subject to direct flame impingement. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a low threshold for identifying and entering issues into the Corrective Action Program [P.1(a)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: FIN Finding
**Inadequate Risk Management During a Planned Auxiliary Saltwater Pump Outage**
The inspectors identified a finding after Pacific Gas and Electric failed to adequately manage risk during planned maintenance activity as required by Procedure AD7.DC6, "On-line Maintenance Risk Management." On April 5,

2010, work control personnel requested that plant operators simultaneously remove Auxiliary Saltwater Pump 2-2 and Component Cooling Water Heat Exchanger 2-2 from service for two scheduled maintenance activities. Plant operators identified that the combination of the auxiliary saltwater pump and component cooling water heat exchanger out of service at the same time would result in an elevated maintenance risk (Yellow). Procedure AD7.DC6, “On-line Maintenance Risk Management”, Section 2.1, required that the licensee manage plant risk during on-line maintenance by minimizing the number of risk significant equipment simultaneously removed from service. The inspectors concluded that these two maintenance activities could have been performed in series rather than in parallel without affecting the duration either component was unavailable for maintenance. The licensee entered the performance deficiency into the corrective action program as Notification 50309451.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” because the work scope unnecessarily placed the plant into a higher licensee-established risk category and required additional risk management actions. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement adequate corrective actions to prevent unnecessarily entering elevated plant risk for the planned maintenance [P.1(d)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Risk Assessment during Planned Maintenance Activities**
The inspectors identified a noncited violation of 10 CFR 50.65 after Pacific Gas and Electric failed to perform a risk assessment after plant conditions had changed. On July 13, 2010, Pacific Gas and Electric identified that station personnel failed to complete Technical Specification Surveillance Requirement 3.3.4.2, “Remote Shutdown System,” within the specified frequency for both Units. As provided by Surveillance Requirement 3.0.3, the licensee performed a risk evaluation to extend the required surveillance completion time beyond twenty-four hours. The licensee initiated the missed surveillance tests and identified results were outside acceptance criteria. On July 26, 2010, Operations personnel declared several remote shutdown system functions inoperable because reasonable expectation no longer existed that remote shutdown system could perform its safety function. Pacific Gas and Electric failed to reassess the effect on plant risk resulting from inoperable remote shutdown system functions before continuing with scheduled maintenance. A subsequent risk assessment concluded that plant risk was in a higher risk category due to planned maintenance activities conducted during this time frame. The licensee entered the performance deficiency into the corrective action program as Notification 50331841.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” because the overall elevated plant risk would put the plant into a higher licensee-established risk category. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee failed to follow its maintenance risk procedure and reassess plant risk due to changing plant conditions [H.4(b)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010

Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination**
The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to promptly evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, "Operability Determination." The first example involved the failure of engineering personnel to promptly notify plant operations of the failure of the emergency diesel generators to meet licensing and design frequency and voltage recovery requirements. This issue was identified by the NRC on May 11, 2010, but not evaluated for the effect on diesel operability until September 9, 2010. The second example also involved the failure of engineering personnel to promptly notify plant operations to evaluate a nonconforming condition associated with a common cross-tie line that connected both auxiliary saltwater trains. This issue was identified by the NRC on July 22, 2010, but not evaluated for the effect on auxiliary saltwater operability until August 4, 2010. In both examples, engineering personnel failed to follow Procedure OM7.ID12, "Operability Determination," Section 5.1, which required any individual identifying a degraded or nonconforming condition that potentially impacts operability of a system, structure or component to ensure that operations shift management is informed. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notifications 50340417 and 50335847.

The inspectors concluded that the performance deficiency is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. The finding was of very low safety significance (Green) because neither of the two examples was subsequently determined to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming conditions for operability [P.1(c)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the AuxiliarySaltwater System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the failure to maintain adequate design control measures associated with the auxiliary saltwater system. The inspectors identified that the auxiliary saltwater system design did not comply with the plant design bases as described the Final Safety Analysis Report Update. Specifically, an auxiliary saltwater vent line did not meet the requirements established of General Design Criteria 1, "Quality Standards and Records," and Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants." The licensee entered the performance deficiency into the corrective action program as Notification 50328942.

This performance deficiency is greater than minor because the design control attribute of the mitigating systems cornerstone and the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Mitigating Systems Cornerstone, the inspectors concluded the finding was of very low significance (Green) because it was a design deficiency confirmed not to result in the loss of operability or functionality. The inspectors concluded that the finding does not have a crosscutting aspect since the performance deficiency is not reflective of current plant performance.
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the failure to maintain adequate design control measures associated with the emergency diesel generating air system.

Specifically, failure of non-seismically qualified air compressor unloader sensing lines during a seismic event could impact the safety function of the emergency diesel generators. Subsequent analysis of the nonconforming condition performed by the licensee determined the piping would not fail during a postulated seismic event. The licensee entered this issue into the corrective action program as Notifications 50307496, 50307497, 50307504, 50307670, 50308204, and 50308824.

The finding was more than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones the finding was potentially risk significant for a seismic initiating event requiring a Phase 3 analysis. The analyst estimated the nonrecovery probabilities for operators failing to isolate air between the receiver and the compressor prior to air pressure depletion, and operators failing to manually open fuel transfer valves to makeup to the diesel day tank. The final quantitative result was calculated to be 1.06 x 10-6. However, using a qualitative evaluation of the bounding assumptions, the analyst determined that the best available information indicated that the finding was of very low risk significance (Green). The team determined that the finding was reflective of current plant performance because it had been recently identified during the license renewal inspection and had a human performance crosscutting aspect related to decision making because the licensee did not use conservative assumptions when evaluating this nonconforming condition in previous evaluations.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Proficiency of Operators to Meet the Time Critical Operator Actions**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to ensure that operators are able to implement specified actions in response to operational events and accidents. Specifically, operators could not achieve actions within the analysis time estimates for the cold leg recirculation phase of a loss of coolant accident response and the steam generator tube rupture response as described in the licensee’s safety analysis report.

The finding is more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding represented a potential loss of a safety function requiring a Phase 2 analysis. Because the probability of human error is not effectively addressed by a Phase 2 analysis, a Phase 3 analysis was performed. The senior reactor analyst reviewed the actual timing of the walkdowns associated with the steam generator tube rupture time critical actions. The analyst determined that, while the licensee failed to meet the specific cooldown timing documented in the Final Safety Analysis Report, the total time to start cooling the reactor was well within the total critical timing of the event. The analyst found no impact on safety in delaying the cooldown of the reactor for one minute given that the other time critical actions were performed more quickly than required. Therefore, the analyst determined that this portion of the finding was of very low safety significance because it does not represent an actual loss of safety function (Green). The senior reactor analyst reviewed the issue related to the assumed action times associated with switching over to containment sump recirculation lineup for their emergency core cooling system pumps during a large break loss of coolant accident. The analyst noted that this time critical action was only required if a large-break loss of coolant accident occurred simultaneously with the failure of an residual heat removal pump to stop automatically, requiring local isolation of the pump. Given that the frequency of the initial conditions for the time critical action are below the Green/White threshold, the change in core damage frequency associated with this finding must be of very low safety significance (Green). The team determined that the finding was reflective of current plant performance because the licensee participated in a recent industry-wide study on time critical operator actions, but did not implement any of the group’s recommendations. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process related to verifying the validity of the underlying assumptions used to evaluate the feasibility of operators implementing time critical operator actions.

Inspection Report# : 2010006 *(pdf)*

**Significance: SL-IV** Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Complete and Accurate Information for a Requested License Amendment**
The team identified a noncited violation of 10 CFR 50.9(a), “Completeness and Accuracy of Information” with multiple examples. Specifically, information supplied to the NRC in License Amendment Request 01-10, dated February 24, 2010, related to the revision of Technical Specification 3.8.1, "AC Sources - Operating," were not complete and accurate in all material respects. Following NRC questioning of the discrepancies the licensee withdrew the amendment request.

The finding is more than minor because the inaccurate information was material to the NRC. Specifically, this information was under review by the NRC to evaluate specific changes to the surveillance requirements associated with the emergency diesel generators. Following management review, this violation was determined to be of very low safety-significance because the amendment request was withdrawn before the NRC amended the facility technical specifications. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement VII, paragraph D.1, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV noncited violation. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not adequately evaluate the extent of condition and take appropriate corrective actions after the NRC identified a similar violation.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Untimely and Inadequate Corrective Actions for the Emergency Diesel Generators**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” with two examples for the failure of the licensee to promptly identify and correct nonconforming conditions related to the emergency diesel generators meeting the design basis. The first example resulted from the failure to identify that instrument inaccuracies were not accounted for in the bounding calculations. The second example involved the failure to identify that the worst case loading calculations exceeded the emergency diesel generator operating load limit.

The failure to promptly identify and correct the design deficiencies associated with the emergency diesel generators was a performance deficiency. This finding is greater than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, “Significant Determination Process,” the team performed a Phase 1 analysis to analyze the significance of this finding and determined the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent an actual loss of safety function of the system or train, did not result in the loss of one or more trains of nontechnical specification equipment, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process or conduct an adequate effectiveness review to verify the validity of the underlying assumptions for a safety-significant decision.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Appropriately Evaluate Failed Residual Heat Removal Surveillance Test**
The team identified a noncited violation of Technical Specification 5.4.1.a for failure to appropriately evaluate and correct a condition adverse to quality, as instructed by Surveillance Test Procedure P-RHR-A22,” Comprehensive

Testing of Residual Heat Removal Pump." Specifically, the licensee failed to recognize a deviation in differential pressure towards the alert range, following the February 9, 2008, comprehensive surveillance test of the 2-2 residual heat removal pump. Continued degradation of the 2-2 residual heat removal pump resulted in failure of the October 9, 2009, comprehensive surveillance test due to the differential pressure exceeding the action limit. The licensee entered this issue into the corrective action program as Notification 50308225.

The finding is more than minor because it was associated with the equipment reliability attribute of the Mitigating Systems Cornerstone and it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating Systems Cornerstone. The finding was determined to be of very low safety significance (Green) because: (1) it was a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The team determined that this finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program, because the licensee failed to appropriately evaluate the 2009 residual heat removal surveillance test failure such that the resolution identified and corrected the cause of the failure.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following Identification of a Non-conservative Technical Specification**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, "Corrective Action," after Pacific Gas and Electric failed to implement prompt corrective actions after identifying a nonconservative technical specification. In December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design basis were met. On January 9, 2009, the licensee entered this condition into the corrective action program. On April 9, 2009, Pacific Gas and Electric concluded that Technical Specification Surveillance Requirement 3.8.1, "AC Sources – Operating," was not adequate to preserve plant safety and applied the provisions of Technical Specification Surveillance Requirement 3.0.3, and Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety." The licensee did not complete the necessary actions to correct the deficient technical specification by submitting an adequate license amendment request. The inspectors concluded the most significant contributor to the finding was a less than adequate engineering evaluation to support the new emergency diesel generator loading profiles following the previous violation. The licensee entered the performance deficiency into the corrective action program as Notification 50232181.

The inspectors determined that the performance deficiency is more than minor because if left uncorrected, the failure to implement prompt corrective actions has the potential to lead to a more significant safety concern. The inspectors concluded the finding was of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. The finding is associated with the Mitigating Systems Cornerstone. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative technical specification such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(B) and after Pacific Gas and Electric failed to submit a required licensee event report within 60 days following discovery of a

condition prohibited by the plant technical specifications and a condition that could have prevented the fulfillment of a safety function. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme, required by Technical Specification 3.3.5, “Loss of Power Diesel Generator Start Instrumentation,” was inadequate to protect operating engineering safety feature pump motors. The licensee concluded that sustained degraded voltage could result in an overcurrent condition affecting equipment powered from the preferred offsite power supply. This condition was required to be reported to the NRC because the degraded voltage protection scheme rendered engineered safety feature pumps inoperable for a period in excess of the allowable technical specification out of service time and the condition resulted in the loss of the degraded voltage protection scheme safety function on all three vital 4 kV power buses.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC’s ability to perform its regulatory function. The inspectors concluded the violation was a Severity Level IV because the licensee failed to submit an adequate licensee event report. The inspectors determined that the violation was also a finding under the reactor oversight process because licensee personnel failed to adequately evaluate a condition adverse to quality for operability and reportability, as required by station procedures. The inspectors concluded that the finding is more than minor because the failure to properly evaluate degraded plant equipment for past operability and reportability could reasonably be seen to lead to a more significant condition. The inspectors concluded that the finding had very low safety significance because the failure to adequately evaluate the condition did not result in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the degraded voltage protection scheme such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 for the licensee's failure to demonstrate that prior NRC approval was not required prior to making changes to the facility degraded voltage protection scheme as described in the Final Safety Analysis Report Update. In response to this violation, the licensee re-performed the corresponding safety analysis to demonstrate that the subject change to the facility degraded voltage protection scheme was consistent with General Design Criteria 17, "Electric Power Systems." The violation is in the licensee's corrective action program as Notification 50306053.

The failure of Pacific Gas and Electric to perform a 10 CFR 50.59 evaluation of modifications to the offsite power protection scheme, in accordance with NEI 96-07, was a performance deficiency. The violation was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The violation screened as very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee did not adopt the requirement to demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that the proposed action was unsafe in order to disapprove the action, in that the Plant Safety Review Committee did not require that a 50.59 evaluation be performed to demonstrate that the proposed action was safe in order to proceed [H.1(b)].
Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Evaluate Changes to the Diesel Testing as Described in the Final Safety Analysis Report**

**Update**
The team identified two examples of a Severity Level IV noncited violation of 10 CFR 50.59 after the licensee failed to perform an adequate evaluation to demonstrate that prior NRC approval was not required before making changes to the frequency and voltage recovery criteria and to the diesel testing commitments as described in the Final Safety Analysis Report Update. Specifically, the 1998 Final Safety Analysis Report Update identified a change from Safety Guide 9 to Regulatory Guide 1.9, Revision 2. The scope involved the removal of the KWS delay and included new requirements for voltage and frequency response. This resulted in a reduction in acceptance criteria. The team also identified a second example where the licensee failed to evaluate the 2005 Final Safety Analysis Report Update change from Regulatory Guide 1.9, Revision 2 to Revision 3 for diesel testing and interval frequency. Using NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, the team concluded that these changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases. In addition, the licensee's 50.59 evaluation, for DCP E-049425, Revision 0, "EDG Starting, and Loading Capability" was less than adequate to conclude that prior NRC approval was not required for the changes. The licensee has entered these issues into their corrective action program as Notification 50302481.

The failure of Pacific Gas and Electric to perform an adequate 10 CFR 50.59 evaluation prior to changing the facility as described in the Final Safety Analysis Report Update is a performance deficiency. The violation was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The violation screened as very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The finding has a crossutting aspect in the area of problem identification and resolution associated with the corrective action program component. In License Amendment Request 10-01, dated February 24, 2010, the licencee did not thoroughly evaluate the original problem of using the 10 CFR 50.59 evaluation process to justify using Regulatory Guide 1.9, Revision 2, Section C, Position 4, as an exception to meeting the frequency and voltage criteria identified in Safety Guide 9 [P.1(c)].
Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-Conservative Decision Making resulted in a Violation of Technical Specification**
On March 10, 2010, the inspectors identified a noncited violation of Technical Specification 3.7.7, "Vital Component Cooling Water System," after both Unit 2 component cooling water loops were inoperable longer than permitted during power operations. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme was inadequate to ensure minimum required voltage would be available to operating engineered safety feature pumps during a degraded offsite power grid. The licensee concluded that operating pumps could trip and lock out on over current before the protection scheme would automatically transfer power to the emergency diesel generators. The licensee declared the 230kV offsite power systems inoperable and took compensatory actions to enable the automatic transfer of busses with operating engineered safety feature pumps directly to the diesel generators following a unit trip. On March 10, 2010, the inspectors identified that operating component cooling water pump 2-3 was still aligned to automatically transfer to 230kV offsite power source following a unit trip. The licensee had previously removed component cooling water pump 2-2 from service for maintenance on March 7, 2010. Technical Specification 3.7.7, "Vital Component Cooling Water System," required a minimum of two operable component cooling water pumps to establish operability of a vital component cooling water loop. Contrary to Technical Specification 3.7.7, on March 10, 2010, the licensee operated Unit 2 without an operable vital component cooling water loop for greater than 14 hours. The licensee has entered this issue into their corrective action program as Notification 50304802.

Either the failure of Pacific Gas and Electric to restore at least two operable component cooling water pumps or to have placed Unit 2 in Mode 3 within six hours, as required by plant Technical Specification 3.7.7, was a performance deficiency. The performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance, of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences (i.e., core damage), and it was within the licencee's ability to correct this problem. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the finding because the violation represents the actual loss of safety function for greater than the technical specification allowed outage time. The finding was of very low safety significance (Green) based on a bounding qualitative evaluation using Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," The inspectors based this conclusion on the low probability of an actual degraded grid condition coincidental with an accident or anticipated operational occurrence during the 14-hour exposure that the vital component cooling water loops were unavailable due to the performance deficiency. The inspectors concluded that this finding had a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the degraded voltage protection scheme [H.1(b).

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with the Current Plant Design Bases**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to include the current plant design basis for the 230kV degraded voltage protection scheme in the Final Safety Analysis Report Update. Title 10 CFR 50.71 (e) states in part, "Each person licensed to operate a nuclear power reactor shall update periodically, as provided in paragraphs (e) (3) and (4) of this ection, the Final Safety Analysis Report Update originally submitted as part of the application for the operating license, to assure that the information included in the report contains the latest information developed. Contrary to the above, on March 14, 2010, the inspectors identified that Pacific Gas and Electric failed to update the Final Safety Analysis Report Update to include complete design basis information for the offsite degraded voltage protection scheme. The inspectors identified that Final Safety Analysis Report Update did not include the design basis for the allowance time delay or the limiting voltage setpoints. The licensee has entered this issue into their corrective action process as Notification 50313763.

Failure to include the current plant design basis for the 230kV degraded voltage protection scheme in the Final Safety Analysis Report Update is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this issue was to be ealuated using the traditional enforcement process because the performance deficiency was a failure to meet a requirement or standard, had the potential for impacting the NRC's ability to perform its regulatory function, and the concern was within the licensee's ability to foresee and correct and should have been prevented. The team used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I "Reactor Operations," dated January 14,2 005, to evaluate the significance of this violation. The team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation because the erroneous information was not used to make any unacceptable change to the facility or procedures. Using Inspection Manual Chapter 0609.04, “Phase 1 Initial Screening and Characterization of Findings,” the team concluded that the issue screened as having low safety significance (Green) under the Significance Determination. Because this violation is of very low safety significance and was entered into the licensee’s corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process and had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of the condition and take appropriate corrective actions after the NRC identified a similar violation [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination Associated With the Offsite Degraded Voltage Protection Scheme**
On February 27, 2010, the inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to complete an adequate operability evaluation, as required by Procedure OM7.ID12, “Operability Determination,” Revision 14. The inspectors identified that the offsite power degraded voltage protection scheme time delay was inconsistent with key assumptions in the accident analysis. The licensee entered this nonconforming condition into the corrective action program as Notification 50301167 on February 24, 2010. Plant operators subsequently requested plant engineering to perform an operability determination of the nonconforming condition per Operability Determination Procedure OM7.ID12. On February 27, 2010, the plant operating authority concluded that the protection scheme was operable based on the information provided in the operability determination. Contrary to the above, on March 2, 2010, the inspectors concluded that the licensee’s operability determination was inadequate to demonstrate protection scheme operability and was not performed as required by Operability Determination Procedure OM7.ID12. Plant engineering only addressed the capability of the protection scheme at normal grid voltage following a mechanical failure of the 230 kV load tap changer. Operability Determination Procedure OM7.ID12, Section 5.3, “Write the Prompt Operability Assessment (POA),” required that the licensee address the potential effect of the nonconforming condition to perform the specified safety function. The licensee has entered this finding into the corrective action program as Notification 50319258.

Failure to complete an adequate operability evaluation, as required by Procedure OM7.ID12, “Operability Determination,” Revision 14, is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality, and the failure to perform an adequate operability evaluation affects the ability to ensure operability of the protection scheme at normal grid voltage following a mechanical failure of the 230 kV load tap changer. The inspectors used Inspection Manual Chapter 609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the finding because the violation represent the actual loss of safety function for greater than the Technical Specification 3.3.5 allowed outage time. Using Appendix M, of the “Significance Determination Process Using Qualitative Criteria,” the inspectors concluded that the finding was of very low safety significance (Green) based on a bounding qualitative evaluation. The inspectors based this conclusion on the low probability of an actual degraded grid condition coincidental with an accident or anticipated operational occurrence during the exposure time that protection scheme was available due to the performance deficiency. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition for operability and reportability [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Second Level Undervoltage Relay Time Delay to Initiate Load Shed and Sequencing Upon the Diesel Generator is Adequate to Assure Plant Safety**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure that plant conditions were consistent with design calculation inputs and assumptions. The licensee failed to assure and verify that Technical Specification 3.3.5 (SR3.3.5.3) pertaining to the second level undervoltage relay time delay to initiate load shed and sequencing upon the diesel generator was adequate to assure plant safety. Supplemental Safety Evaluation Report 09, Section 8.1, requires that a second level of under voltage protection for the onsite power system be provided. Subsection (1)(c)(i), reads: “The allowable second level undervoltage relay time delay, including margin, shall not exceed the maximum time delay that is assumed in the Final Safety Analysis Report Update accident analyses.” Contrary to the above, as of March 4, 2010, the licensee failed to adequately implement the requirements of Supplemental Safety Evaluation Report 09. The second level undervoltage relay time delay setpoint for the emergency diesel generator of less than or equal to 20 seconds, assuming a safety injection signal concurrent with a degraded off site power source, exceeded the Final Safety Analysis Report Update accident analysis. This item is identified in the

licensee's corrective action document Notification 50301167.

Failure to ensure that plant conditions were consistent with design calculation inputs and assumptions is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix E, Section 3 Example j, the violation was determined to be more than minor because the engineering calculation error results in a condition where there is now a reasonable doubt on the operability of a system or component. These deficiencies represented reasonable doubt regarding the mitigation of an accident by being in an unanalyzed condition. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 2, the finding was determined to have very low safety significance (Green), did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. The team reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Drawings and Procedures to Align Emergency Makeup Water Supply from Diablo Canyon Creek to Support the Auxiliary Feedwater System**
The team identified a noncited violation of Diablo Canyon Technical Specification 5.4.1. "Procedures," for failure to have a procedure. The Diablo Canyon Final Safety Analysis Report Update, Revision 18, Section 6.5.2.1.1 documents the design of the auxiliary feedwater system and credits eight sources of water that can provide backup means of supply in the event that its primary source of water, the condensate storage tank, becomes exhausted. One of the sources included is the Diablo Canyon Creek. Diablo Canyon Technical Specification 5.4.1 states: "Written procedures shall be established, implemented, and maintained covering the following activities: [a.] The applicable procedures recommended in NRC Regulatory Guide 1.33, Revision 2, Appendix A, February 1978". NRC Regulatory Guide 1.33, Revision 2, Appendix A, describes procedures under Items 3l (instructions for shutdown, startup, and operation, including system filling, of the auxiliary feedwater system) and 6j (loss of feedwater system or feedwater system failure). Contrary to Technical Specification 5.4.1, on March 4, 2010, the team identified that the licensee did not have an established procedure for accomplish the task identified in the Final Safety Analysis Report Update, Section 6.5.2.1.1 for taking water from the Diablo Canyon Creek to be a supply for the auxiliary feedwater system. This item is identified in the licensee's corrective action document Notification 50298563.

Failure to provide a procedure or instructions and acceptance criteria to perform an emergency makeup water alignment to the auxiliary feedwater system is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality, and the lack of having this procedure affects the ability to ensure the availability, reliability, and capability of the auxiliary feedwater system to respond to initiating events to prevent undesirable consequences, (i.e., core damage, and it was within the licensee's ability to correct this problem.) Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding was determined to have very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. The team concluded that this finding had a crosscutting aspect in the area of problem identification and resolution, in that the licensees' corrective action program thoroughly evaluates problems such that the resolutions address causes and the extent of conditions, as necessary. Per licensee Notification 50298563, changes were made to pumping systems associated with the Diablo Canyon Creek in 2007, which affected the ability to pump water through the discussed credited lineup supporting the auxiliary feedwater system. This effect was not identified as part of the changes, so no review of procedures related to the emergency auxiliary feedwater system alignment in question was performed. Since these actions occurred within the last three years, this performance characteristic reflects current performance [P.1 (c)].

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation

**Failure to Update Feedwater Rupture Accident Analysis in the Final Safety Analysis Report Update**
The team identified a Severity Level IV noncited violation of 10 CFR 50.71, "Maintenance of records, making of reports." Paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." In the Diablo Canyon Final Safety Analysis Report Update section addressing the feedwater line break accident, it states that operator actions are credited with precluding the operation of pressurizer safety valves based on determinations in Westinghouse study WCAP-11667 (1998) (Final Safety Analysis Report Update, Section 15.4.2.2.2). Review of this study, and associated correspondence on the topic during 2006 indicated that the Westinghouse study did not state that operator actions could be credited for this event, but analysis of the worst case pressurizer overfill accidents by the licensee may show that this is the bounding case for such accidents, and that it did not need to be addressed in the feedwater line break analysis. In 2006, the licensee indicated that they would revise the Final Safety Analysis Report Update text to remove this reference to the Westinghouse study, which had been in the Final Safety Analysis Report Update since Revision 16. Contrary to the above, since 2006 (Final Safety Analysis Report Update Revision 16), the licensee failed to update Final Safety Analysis Report Update, Section 15.4.2.2.2. The licensee has entered this issue into their corrective action process as Notification 50301747.

Failure to periodically update the Final Safety Analysis Report Update with a known error is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this issue was to be evaluated using the traditional enforcement process because the performance deficiency was a failure to meet a requirement or standard, had the potential for impacting the NRC's ability to perform its regulatory function, and the concern was within the licensee's ability to foresee and correct and should have been prevented. The team used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, "Reactor Operations," dated January 14, 2005, to evaluate the significances of this violation. The team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The inspectors reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Text to Reflect Credited Design Class I Makeup Flowpath to Component Cooling Water Expansion Tank in the Final Safety Analysis Report Update**
The team identified a Severity Level IV noncited violation of 10 CFR 50.71, "Maintenance of records, making of reports." Title 10 CFR 50.71, paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." In the Final Safety Analysis Report Update Table 9.2-7, Component 5, it states, "This 250 gpm, Design Class I, makeup water flowpath, described under Makeup Provisions in Subsection 2.3.3 (Section 9.2.2.3.3), can be started within 10 minutes." Final Safety Analysis Report Update, Section 9.2.2.3.3 states, "All piping and valves in the makeup path from the condensate storage tank (including their cross-connections) and the firewater tank, through the makeup water transfer pumps up to and including the makeup valves on the component cooling water system lines, are Design Class I." Text later in the section implied that the flow path from the firewater tank was not Design Class I. Review by the licensee staff revealed that the only Design Class I flow path to provide makeup to the component cooling water expansion tank is via the condensate storage tank. This revealed that the text provided in Final Safety Analysis Report Update, Section 9.2.2.3.3 stating that both the condensate storage tank and firewater tank makeup paths are credited is incorrect. Contrary to above, since 1984 (Final Safety Analysis Report Update, Revision 0), the licensee did not update Final Safety Analysis Report Update, Section 9.2.2.3.3 to correct the error of including firewater as a possible makeup path to the component cooling water expansion tank. The licensee has entered this issue into their corrective action process as Notification 50301884.

Failure to periodically update the Final Safety Analysis Report Update with a known error is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this performance deficiency was to be evaluated using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. Using General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, Reactor Operations, dated January 14, 2005, to evaluate the significances of this violation, the team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The team reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Effectively Implement the Seismically-induced Systems Interaction Program**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric personnel failed to effectively implement the Seismically Induced System Interaction Program. The Seismic Interaction Program is part of the design basis mitigation strategy for a potential 7.5 magnitude Hosgri earthquake and is required by Procedure AD4.ID3, "SISIP Housekeeping Activities." The inspectors identified three examples of transient equipment and materials improperly staged in seismically induced system interaction target areas. Pacific Gas and Electric had not analyzed the transient equipment to assess the risk to safety related components as required by plant procedures. Pacific Gas and Electric entered this finding into the corrective action program as Notification 50299740.

The finding is more than minor because the failure to follow the Seismically Induced System Interaction Program is associated with the Mitigating Systems Cornerstone external events protection attribute and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding had very low safety significance because none of the examples of improperly staged equipment resulted in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. The inspectors concluded this finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee's past actions to address Seismically Induced System Interaction Program deficiencies were not effective [P.1(d)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report with the Current Plant Design Bases**
The inspectors identified a noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with the current design basis. The inspectors identified that the current Final Safety Analysis Report Update, Revision 18, Sections 3.1, 6.4, 6.5, and 9.4 did not capture the current design basis for the control room, component cooling water, and auxiliary feedwater systems. The failure of the licensee to provide current design basis information in the Final Safety Analysis Report Update had an adverse impact on the plant modification process, the licensee's ability to assess operability for degraded plant systems, and the NRC's ability to ensure that regulatory requirements were met. The licensee entered this violation into the corrective action program as Notifications 50308588, 50306131, 5030799, and 50307476.

The inspectors evaluated this violation using the traditional enforcement process because the issue affected the NRC's ability to perform its regulatory function. The inspectors concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. The inspectors concluded the violation is Severity Level IV because the erroneous information was not used to make

an unacceptable change to the facility or procedures that would have resulted in greater than very low safety significance under the Significance Determination Process. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of condition of previous similar violations and take appropriate corrective actions [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovering a condition that could have prevented the fulfillment of a safety function. On November 22, 2005, the licensee determined that plant operators may not have had the capability to align either residual heat removal train to the cold leg recirculation mode of emergency core cooling following certain small break loss of coolant accidents. Plant engineers determined that the residual heat removal containment sump suction valve operators were inadequately sized to open against the differential pressure generated by the pumps operating in recirculation for an extended period. Plant engineers identified this condition during a follow up of industry operating experience. The licensee initially concluded that the condition was not reportable because the operating experience was not applicable to Diablo Canyon. The licensee failed to re-screen the issue for reportability after determining that the plant was susceptible to the condition. The licensee entered this issue into the corrective action program as Notifications 50301839 and 50295784.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded the violation was a Severity Level IV because the licensee failed to submit a required licensee event report. The inspectors did not assign a crosscutting aspect because the performance deficiency represented a latent issue.
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation Following the Failure of Both Motor-Driven Auxiliary Feedwater Trains**
The inspectors identified a noncited violation of 10 CFR, Part 50, Appendix B, Criteria XVI, "Corrective Actions," after Pacific Gas and Electric failed to implement adequate corrective actions following a protection system failure. On June 29, 2009, a protection system card failure resulted in the inoperability of both motor-driven auxiliary feedwater trains. The licensee concluded that the failure of the auxiliary feedwater trains were expected as part of the protection system design and limited corrective actions to replacing the failed card. The inspectors concluded that the protection system design did not meet the design basis, which required that no single active failure would prevent the auxiliary feedwater system from meeting the safety function. The licensee entered this issue into the corrective action program as Notifications 50251823, 50298491 and 50254412.

The inspectors concluded that the finding is greater than minor because the vulnerability of auxiliary feedwater to a single failure is associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the finding to have very low safety significance because the condition did not represent a loss of system safety function. While the single failure of the protection system card resulted in the inoperability of both motor-driven auxiliary feedwater trains, the turbine-driven auxiliary feedwater train was available to perform the safety function. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the auxiliary feedwater failure such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance: SL-IV** Mar 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit a Licensee Event Report following the Common-Cause Failure of Independent Trains or Channels**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(1) after Pacific Gas and Electric failed to submit a required licensee event report within 60 days after discovery of a common-cause failure of three control room radiation monitors. The inspectors concluded that monitors failed on October 13, 2009, as a result of water intrusion due to heavy rains. The inspectors concluded that common cause failure of the radiation monitors was reportable under 10 CFR 50.73(a)(2)(vii). Pacific Gas and Electric subsequently reported the event on February 17, 2010, as Licensee Event Report 2010-001-00, Control Room Ventilation Pressurization Due to Radiation Detector Failures. The licensee entered this issue into the corrective action program as Notification 50301839.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. Consistent with the guidance in Section IV.A.3 and Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the inspectors concluded that this was a Severity Level IV noncited violation because the licensee failed to submit a required licensee event report. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to thoroughly evaluate the failure of the radiation monitor failures to ensure NRC reportability requirements were met [P.1(c)].
Inspection Report# : 2010002 *(pdf)*

**Significance:** G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Design and Configuration Control Requirements**
The inspection team identified a noncited violation of 10 CFR 50, Appendix B, Criterion III, Design Control, which requires licensees to implement measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These design control measures include verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. Specifically, on February 16, 2008, plant engineering personnel failed to implement the design control process for a modification to the Unit 2 residual heat removal containment sump valves when they inappropriately used maintenance procedures to reduce the valve stroke lengths from 15.5 to 13.8 inches. The invalid design change resulted in the inoperability of both emergency core cooling trains between April 8, 2008, (when the plant entered Mode 4) and October 22, 2009. The reduced sump valve stroke length also caused a portion of the sump valve disc to remain in the residual heat removal suction flow path, reducing the available residual heat removal pump net positive suction head. The licensee entered this condition into their corrective action program as Notification 50277252.

The inspection team concluded that the failure of plant engineering to use the design control process was a performance deficiency within the licensee's ability to foresee and correct. The finding is more than minor because it affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding required a Phase 2 analysis because the finding represented the loss of a safety system function. The Phase 2 analysis determined that this finding was potentially greater than Green; therefore, a Phase 3 analysis was completed by a regional senior reactor analyst. The Phase 3 analysis determined that this issue was of very low safety significance (Green), owing principally to the fact that operators could have opened the affected valves locally with a very high probability of success. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the failure of the valves to meet the specified stroke time to ensure that the resolution fully addressed the causes and extent of condition, as necessary [P.1(c)].

Inspection Report# : 2009009 *(pdf)*

**Significance:** G Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Conduct an Adequate Post-Modification Test**
The inspection team identified a noncited violation of 10 CFR 50, Appendix B, Criterion XI, Test Control, which requires that a test program be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service. Specifically, the licensee failed to perform testing to assure that the interlock circuitry associated with the residual heat removal containment sump valves (SI-2-8982A and B) would perform satisfactorily in service following a modification on February 16, 2008, that changed the stroke lengths. As a consequence, remote operation of the valves needed to initiate high pressure recirculation was lost for an entire operating cycle. The licensee entered this issue into their corrective action program as Notification 50277252.

The failure to establish adequate post-modification testing requirements was a performance deficiency within the licensee's ability to foresee and correct. The finding is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences was affected. Using Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," the finding required a Phase 2 analysis because the finding represented the loss of a safety system function. The Phase 2 analysis determined that this finding was potentially greater than Green; therefore, a Phase 3 analysis was completed by a regional senior reactor analyst. The Phase 3 analysis determined that this issue was of very low safety significance (Green), owing principally to the fact that operators could have opened the affected valves locally with a very high probability of success. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the operating experience component because the licensee failed to implement a corrective action program with a threshold sufficient to identify issues associated with the failure to meet sump valve post-modification test acceptance criteria [P.1(a)].

Inspection Report# : 2009009 *(pdf)*

**Significance: SL-IV** Jan 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate a Change to the Facility as Described in the Final Safety Report Update Associated with the Addition of Manual Actions in the Safety Analysis**
The inspection team identified a noncited violation of 10 CFR 50.59, which states that a licensee may make changes to the facility as described in the final safety analysis report without obtaining a license amendment if the change does not result in a departure from a method of evaluation described in the final safety analysis report used in establishing the design bases or in the safety analyses. This regulation further requires the licensee to include a written evaluation providing the basis for concluding that a license amendment is not required. On November 21, 2005, the licensee failed to provide a written evaluation concluding that a license amendment was not required for a change to the facility as described in the final safety analysis report. Specifically, the licensee identified a condition where large differential pressure across the residual heat removal suction containment sump valves could cause them to fail to open during certain small break loss of coolant accidents. On October 5, 2005, the licensee revised Emergency Operating Procedure E-1, "Loss of Reactor or Secondary Coolant," to add an operator action to align component cooling water to the residual heat removal heat exchanger. On June 16, 2009, the licensee again revised Emergency Operating Procedure E-1 to specify that operator action to align component cooling water within 30 minutes was a time critical operator action. The licensee did not evaluate either change to determine if prior NRC approval was required for the new manual actions. The licensee entered this issue into their corrective action program as Notification 50276288.

The failure of the licensee to perform a 10 CFR 50.59 evaluation of a new manual action supporting the plant's design basis was a performance deficiency within the licensee's ability to foresee and correct. The inspectors evaluated this issue using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. The inspectors concluded that the issue was more than minor because of a reasonable likelihood that the change to the facility would require Commission review and approval prior to implementation. The inspectors also evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609.04, "Phase 1 Initial Screening and Characterization of Findings." The inspectors concluded that the issue affected the Mitigating Systems Cornerstone and screened Green because the finding was a design or qualification deficiency confirmed not to result in loss of operability. The issue was classified as Severity Level IV because the violation of 10 CFR 50.59 involved conditions resulting in very low safety significance by the significance determination process. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the change to the facility as described in the Final Safety Analysis Report Update to determine if prior NRC approval was required [P.1(c)].

Inspection Report# : 2009009 *(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

**Significance: N/A** Jul 27, 2010
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**
The team concluded that the notification process facilitates the initiation, tracking, and trending of concerns and that the licensee correctly identified deficiencies that were conditions adverse to quality and entered them into the corrective action program in accordance with the licensee's corrective action program guidance and NRC requirements. Prioritization of issues was appropriate. The licensee was inconsistent in the effectiveness of evaluating issues once they were identified. The team's assessment was there was limited effective interdepartmental communication, a lack of cross discipline peer checks, and a failure to assign the appropriate resources to evaluate cross-departmental problems/issues. As a result, the licensee's performance in resolving problems and effective utilization of operating experience was negatively impacted. The licensee performed effective quality assurance audits and self-assessments, as demonstrated by self-identification of poor corrective action program performance and identification of ineffective corrective actions. However, because of challenges in performing evaluations, the licensee had difficulty properly addressing some of these issues. Overall the team concluded that implementation of the corrective action program was adequate with improvements warranted.

The team determined that site personnel were willing to raise safety issues and document them in the corrective action program. The team noted that workers at the site felt free to report problems to their management and the NRC, but were reluctant to take safety concerns to the Employee Concern Program. Additionally, the function and processes associated with the Employee Concern Program was not understood by a majority of the personnel interviewed.

Inspection Report# : 2010006 *(pdf)*

Last modified : March 03, 2011

# Diablo Canyon 2
# 1Q/2011 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Preferred Offsite Power System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the preferred offsite power system design basis was correctly translated into electrical dynamic loading Calculations 357A-DC, “Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis,” Revision 12 and 359-DC, “Offsite Power Dynamic Analysis,” Revision 8. The licensee did not include the limiting load flow cases representing the largest total onsite demand for both units as required by the plant design basis. On July 7, 2010, the NRC clarified that the Diablo Canyon current licensing basis required the preferred offsite power system to have adequate capacity and capability to supply the most limiting loading requirements, including a dual unit trip. The licensee subsequently entered the condition into the corrective action program as Notification 50289590 and revised the station dynamic loading analysis to reflect the increased onsite power demand.

The inspectors concluded that the failure to ensure that the dynamic loading analysis included all design basis requirements was a performance deficiency. This performance deficiency is more than minor because the finding was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because the inspectors were unable to conclude that the preferred offsite power system had not been inoperable for greater than the allowed Technical Specification outage time, a senior reactor analyst performed a bounding Phase 3 analysis. The Phase 3 analysis demonstrated that the subject finding was of very low safety significance (Green), because of the small increase of probability of a loss of offsite power that the finding represented. This finding had a crosscutting
aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the current licensing basis requirements to ensure that resolutions addressed causes and extent of conditions, as necessary.
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain the integrity of Door 155 in the rated condition. On December 9, 2010, the inspectors identified that the fire door was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain Door 155 in a configuration that would provide at least a 1½-hour rated fire barrier. The inspectors previously identified that Door 155 was degraded as a fire barrier in 2009. The licensee entered the violation into the corrective action program as Notification 50367381 and took immediate corrective actions to restore the fire barrier to the rated condition and to implement weekly plant fire door walkdowns.

The inspectors concluded that the finding was more than minor because the degraded fire barrier affected the

Mitigating Systems Cornerstone external factors attribute and objective to prevent undesirable consequences due to fire. The inspectors determined that the finding was within the fire confinement category and that the fire barrier was moderately degraded. The inspectors concluded that the finding was of very low safety significance (Green) because there was a non-degraded automatic full area water-based suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take effective corrective actions to following the previous occurrence of the violation [P.1(d)].

Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Transient Combustibles Procedure**
The inspectors identified a noncited violation of Diablo Canyon Unit 2 Facility Operating License Condition 2.C.(5), “Fire Protection,” after Pacific Gas and Electric failed to ensure procedures for controlling flammable and combustible materials adequately incorporated requirements of the fire hazard analysis. On October 18, 2010, the inspectors identified that transient combustible materials staged in the Unit 1 12 kilovolt switchgear room did not have an approved transient combustibles permit. The licensee stated that the combustibles permit procedure did not require a permit for the room while Unit 1 was shutdown. However, the plant fire hazards design basis described safe shutdown equipment in the room that would be needed to support a safe shutdown of the operating unit, specifically the Unit 2 startup bus located in the room. The inspectors determined that the licensee’s transient combustibles permit procedure was inadequate because the procedure did not require a permit for the Unit1 12 kilovolt switchgear room when Unit 2 was operating. The licensee entered the issue into the corrective action program as Notification 50366302 and performed an evaluation of the transient combustibles stored in the area.

The inspectors concluded that this finding was more than minor because it affected the Mitigating Systems Cornerstone external factors attribute objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors determined that the finding was within the fire prevention and administrative controls category and represented a low degradation level due to the minimal impact on the effectiveness and reliability of the affected systems. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening, the low degradation rating, and only equipment needed to reach and maintain cold shutdown conditions was affected. This finding had a crosscutting aspect in the area of human performance associated with the resources component because the licensee failed to ensure that the design documentation adequately identified the Unit 2 startup bus as equipment required for safe shutdown for Unit 2 [H.2 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determinations**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to adequately evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, “Operability Determination.” On October 15, 2010, the inspectors identified a less than adequate technical evaluation supporting Prompt Operability Assessment 50350918, “Unit 2 - Insulation in Bio-Wall Penetration.” Engineering personnel failed to adequately evaluate the extent of condition after technicians identified about 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous insulation in the Unit 1 reactor coolant loop biological shield wall penetrations. This fibrous material could have potentially been transported and plugged the emergency core cooling containment sump screen. The licensee performed the prompt operability assessment for Unit 2, which was operating at the time. The inspectors concluded that the engineering personnel inappropriately applied the leak-before-break methodology to exclude about 87 percent of this material from the extent of condition review in the prompt operability assessment.
The second example involved Prompt Operability Assessment Notification 50355265, “RHR Sump Margin,” which was completed by the licensee on October 23, 2010. In this example, engineering personnel failed to identify and

demonstrate that the specified safety function of the refueling water storage tank could be maintained as required by the plant operability procedure. The inspectors identified that the post accident flow path from the reactor cavity to the containment sump was blocked by a large shield plug. This blockage
reduced the amount of post accident inventory available at the containment sump at the time of transition from injection to recirculation mode of emergency core cooling operation. Engineering personnel failed to demonstrate that the safety function to ensure full sump submergence was maintained with the blocked flow path. Full submergence of the sump was used by the NRC as the basis for approval of Technical Specification 3.5.4, “Refueling Water Storage Tank,” inventory requirements. The licensee entered the violation into the corrective action program as Notification 50369117 and revised the prompt operability assessments using assumptions consistent with the current licensing bases.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because Pacific Gas and Electric did not use conservative assumptions in decisions to demonstrate component operability in either example [H.1(b)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Containment Recirculation Sump Design Control**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure Calculation STA-255, “Minimum Required Refueling Water Storage Tank Level for GE Sumps,” Revision 2, demonstrated adequate available refueling water storage tank inventory. On October 19, 2010, the inspectors identified that emergency core cooling post accident flow path from the reactor cavity to the containment sumps was blocked by a large steel plug on Unit 1. The accident analysis assumed this 35 square foot path was open to allow coolant from a pipe break inside the biological shield to communicate with containment sumps during the recirculation mode of emergency core cooling. The licensee credited the inventory from the reactor cavity when determining the minimum required refueling water storage tank volume in Calculation STA-255. Pacific Gas and Electric used Calculation STA-255 as the basis for determining the minimum required refueling water storage tank volume specified by Technical Specification 3.5.4, “Refueling Water Storage Tank.” The inspectors identified that the recirculation flow path was also blocked on Unit 2. The inspectors concluded that the most significant contributor to the violation was inaccurate plant drawings used by plant engineers during the performance of Calculation STA-255. The licensee’s corrective actions included completion of a prompt operability assessment justifying continued operation of Unit 2 and replacement of the shield plug with a movable platform on Unit 1 prior to plant restart.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone plant modification design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the performance deficiency involved a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the resources component because Pacific Gas and Electric failed to use complete, accurate and up-to-date drawing for Calculation STA 255 [H.2(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Emergency Diesel Generator Surveillance Testing**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” after Pacific Gas and Electric failed to develop and implement an adequate testing program for the emergency diesel

generators that met design requirements and recommendations. Specifically, in December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design bases were met. Pacific Gas and Electric updated the load calculations, but failed to make the necessary revisions to Surveillance Test Procedure STP M-9D1, "Diesel Generator Full Load Rejection Test." As a result, Pacific Gas and Electric failed to test several of the emergency diesel generators at the complete load as required by Regulatory Guide 1.108, Revision 1, which is part of the current licensing bases. The licensee entered this into the corrective action program as Notification 50368801, determined there was no loss of safety function for the affected components, and applied the provisions of Surveillance Requirement 3.0.3 for a missed surveillance test. The inspectors concluded the most significant contributor to the finding was less than adequate diesel generator loading evaluations to support corrective action from previous violations associated with the emergency diesel generator testing.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the equipment control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative surveillance test such that the resolution addressed the fundamental basis for the surveillance [P.1 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Quality Verification Audits**
The inspectors identified a noncited violation of 10CFR Appendix B, Criterion XVIII, "Audits", which required that a comprehensive system of planned and periodic audits be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program as well as follow up action, including re-audit of deficient areas, where indicated. Contrary to this requirement, Pacific Gas and Electric failed to ensure that a comprehensive system of planned and periodic audits were carried out to verify compliance with all aspects of the quality assurance program, determine the effectiveness of the program, and perform necessary follow up actions. Specifically, the 2008 Quality Verification audit of the corrective action program failed to adequately address an adverse trend in the problem evaluation process documented in NRC Inspection report 2008005, which identified eleven examples of an adverse trend in problem evaluation. The licensee entered this into their corrective action program as Notification 50365083 and determined
there was no loss of safety function for the affected components. The inspectors concluded the most significant contributor to the finding was a less than adequate evaluation of the corrective action trending program.

This finding was more than minor because it was associated with the equipment control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the performance deficiency was of very low safety significance (Green) it was a deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to coordinate and communicate the results from assessments to affected personnel, and track the corrective actions to address issues commensurate with their significance [P.3(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify a Degraded Fire Barrier**
The inspectors identified a noncited violation of the Diablo Canyon Facility Operating License Condition (5), "Fire Protection," after Pacific Gas and Electric failed to maintain the integrity of a fire door in the rated configuration. On

August 19, 2010, the inspectors identified that Fire Door 223 was inoperable. Fire Door 223 was required to provide a 3-hour rated barrier between Fire Areas 5-A-4 and 5-B-4. A fire in either of these areas could have prevented operation of the auxiliary feedwater, auxiliary saltwater, or component cooling water pumps or steam generator level control from the remote shutdown panel. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to either maintain Fire Door 223 operable or implement compensatory actions within one hour. The inspectors concluded the most significant contributor to the finding was that licensee personnel did not identify and enter the degraded fire door into the Corrective Action Program. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notification 50336901 and completed repairs to the door on August 23, 2010.

The inspectors concluded that the performance deficiency was more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door was a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated fire barrier function. The inspectors concluded the finding was of very low safety significance because the degraded barrier would have provided a minimum of 20 minutes fire endurance protection and ignition sources and combustible materials were positioned that had a fire spread to secondary combustibles, the degraded barrier would not have been subject to direct flame impingement. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a low threshold for identifying and entering issues into the Corrective Action Program [P.1(a)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: FIN Finding
**Inadequate Risk Management During a Planned Auxiliary Saltwater Pump Outage**
The inspectors identified a finding after Pacific Gas and Electric failed to adequately manage risk during planned maintenance activity as required by Procedure AD7.DC6, "On-line Maintenance Risk Management." On April 5, 2010, work control personnel requested that plant operators simultaneously remove Auxiliary Saltwater Pump 2-2 and Component Cooling Water Heat Exchanger 2-2 from service for two scheduled maintenance activities. Plant operators identified that the combination of the auxiliary saltwater pump and component cooling water heat exchanger out of service at the same time would result in an elevated maintenance risk (Yellow). Procedure AD7.DC6, "On-line Maintenance Risk Management", Section 2.1, required that the licensee manage plant risk during on-line maintenance by minimizing the number of risk significant equipment simultaneously removed from service. The inspectors concluded that these two maintenance activities could have been performed in series rather than in parallel without affecting the duration either component was unavailable for maintenance. The licensee entered the performance deficiency into the corrective action program as Notification 50309451.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," because the work scope unnecessarily placed the plant into a higher licensee-established risk category and required additional risk management actions. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, "Maintenance Risk Assessment and Risk Management Significance Determination Process." This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement adequate corrective actions to prevent unnecessarily entering elevated plant risk for the planned maintenance [P.1(d)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation

**Inadequate Risk Assessment during Planned Maintenance Activities**
The inspectors identified a noncited violation of 10 CFR 50.65 after Pacific Gas and Electric failed to perform a risk assessment after plant conditions had changed. On July 13, 2010, Pacific Gas and Electric identified that station personnel failed to complete Technical Specification Surveillance Requirement 3.3.4.2, "Remote Shutdown System," within the specified frequency for both Units. As provided by Surveillance Requirement 3.0.3, the licensee performed a risk evaluation to extend the required surveillance completion time beyond twenty-four hours. The licensee initiated the missed surveillance tests and identified results were outside acceptance criteria. On July 26, 2010, Operations personnel declared several remote shutdown system functions inoperable because reasonable expectation no longer existed that remote shutdown system could perform its safety function. Pacific Gas and Electric failed to reassess the effect on plant risk resulting from inoperable remote shutdown system functions before continuing with scheduled maintenance. A subsequent risk assessment concluded that plant risk was in a higher risk category due to planned maintenance activities conducted during this time frame. The licensee entered the performance deficiency into the corrective action program as Notification 50331841.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," because the overall elevated plant risk would put the plant into a higher licensee-established risk category. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, "Maintenance Risk Assessment and Risk Management Significance Determination Process." This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee failed to follow its maintenance risk procedure and reassess plant risk due to changing plant conditions [H.4(b)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination**
The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to promptly evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, "Operability Determination." The first example involved the failure of engineering personnel to promptly notify plant operations of the failure of the emergency diesel generators to meet licensing and design frequency and voltage recovery requirements. This issue was identified by the NRC on May 11, 2010, but not evaluated for the effect on diesel operability until September 9, 2010. The second example also involved the failure of engineering personnel to promptly notify plant operations to evaluate a nonconforming condition associated with a common cross-tie line that connected both auxiliary saltwater trains. This issue was identified by the NRC on July 22, 2010, but not evaluated for the effect on auxiliary saltwater operability until August 4, 2010. In both examples, engineering personnel failed to follow Procedure OM7.ID12, "Operability Determination," Section 5.1, which required any individual identifying a degraded or nonconforming condition that potentially impacts operability of a system, structure or component to ensure that operations shift management is informed. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notifications 50340417 and 50335847.

The inspectors concluded that the performance deficiency is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. The finding was of very low safety significance (Green) because neither of the two examples was subsequently determined to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming conditions for operability [P.1(c)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the AuxiliarySaltwater System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to maintain adequate design control measures associated with the auxiliary saltwater system. The inspectors identified that the auxiliary saltwater system design did not comply with the plant design bases as described the Final Safety Analysis Report Update. Specifically, an auxiliary saltwater vent line did not meet the requirements established of General Design Criteria 1, “Quality Standards and Records,” and Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants.” The licensee entered the performance deficiency into the corrective action program as Notification 50328942.

This performance deficiency is greater than minor because the design control attribute of the mitigating systems cornerstone and the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Mitigating Systems Cornerstone, the inspectors concluded the finding was of very low significance (Green) because it was a design deficiency confirmed not to result in the loss of operability or functionality. The inspectors concluded that the finding does not have a crosscutting aspect since the performance deficiency is not reflective of current plant performance.
Inspection Report# : 2010004 *(pdf)*

**Significance:** Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to maintain adequate design control measures associated with the emergency diesel generating air system. Specifically, failure of non-seismically qualified air compressor unloader sensing lines during a seismic event could impact the safety function of the emergency diesel generators. Subsequent analysis of the nonconforming condition performed by the licensee determined the piping would not fail during a postulated seismic event. The licensee entered this issue into the corrective action program as Notifications 50307496, 50307497, 50307504, 50307670, 50308204, and 50308824.

The finding was more than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones the finding was potentially risk significant for a seismic initiating event requiring a Phase 3 analysis. The analyst estimated the nonrecovery probabilities for operators failing to isolate air between the receiver and the compressor prior to air pressure depletion, and operators failing to manually open fuel transfer valves to makeup to the diesel day tank. The final quantitative result was calculated to be 1.06 x 10-6. However, using a qualitative evaluation of the bounding assumptions, the analyst determined that the best available information indicated that the finding was of very low risk significance (Green). The team determined that the finding was reflective of current plant performance because it had been recently identified during the license renewal inspection and had a human performance crosscutting aspect related to decision making because the licensee did not use conservative assumptions when evaluating this nonconforming condition in previous evaluations.

Inspection Report# : 2010006 *(pdf)*

**Significance:** Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Proficiency of Operators to Meet the Time Critical Operator Actions**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving

the failure to ensure that operators are able to implement specified actions in response to operational events and accidents. Specifically, operators could not achieve actions within the analysis time estimates for the cold leg recirculation phase of a loss of coolant accident response and the steam generator tube rupture response as described in the licensee's safety analysis report.

The finding is more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding represented a potential loss of a safety function requiring a Phase 2 analysis. Because the probability of human error is not effectively addressed by a Phase 2 analysis, a Phase 3 analysis was performed. The senior reactor analyst reviewed the actual timing of the walkdowns associated with the steam generator tube rupture time critical actions. The analyst determined that, while the licensee failed to meet the specific cooldown timing documented in the Final Safety Analysis Report, the total time to start cooling the reactor was well within the total critical timing of the event. The analyst found no impact on safety in delaying the cooldown of the reactor for one minute given that the other time critical actions were performed more quickly than required. Therefore, the analyst determined that this portion of the finding was of very low safety significance because it does not represent an actual loss of safety function (Green). The senior reactor analyst reviewed the issue related to the assumed action times associated with switching over to containment sump recirculation lineup for their emergency core cooling system pumps during a large break loss of coolant accident. The analyst noted that this time critical action was only required if a large-break loss of coolant accident occurred simultaneously with the failure of an residual heat removal pump to stop automatically, requiring local isolation of the pump. Given that the frequency of the initial conditions for the time critical action are below the Green/White threshold, the change in core damage frequency associated with this finding must be of very low safety significance (Green). The team determined that the finding was reflective of current plant performance because the licensee participated in a recent industry-wide study on time critical operator actions, but did not implement any of the group's recommendations. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process related to verifying the validity of the underlying assumptions used to evaluate the feasibility of operators implementing time critical operator actions.

Inspection Report# : 2010006 *(pdf)*

**Significance: SL-IV** Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Complete and Accurate Information for a Requested License Amendment**
The team identified a noncited violation of 10 CFR 50.9(a), "Completeness and Accuracy of Information" with multiple examples. Specifically, information supplied to the NRC in License Amendment Request 01-10, dated February 24, 2010, related to the revision of Technical Specification 3.8.1, "AC Sources - Operating," were not complete and accurate in all material respects. Following NRC questioning of the discrepancies the licensee withdrew the amendment request.

The finding is more than minor because the inaccurate information was material to the NRC. Specifically, this information was under review by the NRC to evaluate specific changes to the surveillance requirements associated with the emergency diesel generators. Following management review, this violation was determined to be of very low safety-significance because the amendment request was withdrawn before the NRC amended the facility technical specifications. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement VII, paragraph D.1, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV noncited violation. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not adequately evaluate the extent of condition and take appropriate corrective actions after the NRC identified a similar violation.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation

**Untimely and Inadequate Corrective Actions for the Emergency Diesel Generators**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," with two examples for the failure of the licensee to promptly identify and correct nonconforming conditions related to the emergency diesel generators meeting the design basis. The first example resulted from the failure to identify that instrument inaccuracies were not accounted for in the bounding calculations. The second example involved the failure to identify that the worst case loading calculations exceeded the emergency diesel generator operating load limit.

The failure to promptly identify and correct the design deficiencies associated with the emergency diesel generators was a performance deficiency. This finding is greater than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, "Significant Determination Process," the team performed a Phase 1 analysis to analyze the significance of this finding and determined the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent an actual loss of safety function of the system or train, did not result in the loss of one or more trains of nontechnical specification equipment, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process or conduct an adequate effectiveness review to verify the validity of the underlying assumptions for a safety-significant decision.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Appropriately Evaluate Failed Residual Heat Removal Surveillance Test**
The team identified a noncited violation of Technical Specification 5.4.1.a for failure to appropriately evaluate and correct a condition adverse to quality, as instructed by Surveillance Test Procedure P-RHR-A22," Comprehensive Testing of Residual Heat Removal Pump." Specifically, the licensee failed to recognize a deviation in differential pressure towards the alert range, following the February 9, 2008, comprehensive surveillance test of the 2-2 residual heat removal pump. Continued degradation of the 2-2 residual heat removal pump resulted in failure of the October 9, 2009, comprehensive surveillance test due to the differential pressure exceeding the action limit. The licensee entered this issue into the corrective action program as Notification 50308225.

The finding is more than minor because it was associated with the equipment reliability attribute of the Mitigating Systems Cornerstone and it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating Systems Cornerstone. The finding was determined to be of very low safety significance (Green) because: (1) it was a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The team determined that this finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program, because the licensee failed to appropriately evaluate the 2009 residual heat removal surveillance test failure such that the resolution identified and corrected the cause of the failure.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Corrective Actions Following Identification of a Non-conservative Technical Specification**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria XVI, "Corrective Action,"

after Pacific Gas and Electric failed to implement prompt corrective actions after identifying a nonconservative technical specification. In December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design basis were met. On January 9, 2009, the licensee entered this condition into the corrective action program. On April 9, 2009, Pacific Gas and Electric concluded that Technical Specification Surveillance Requirement 3.8.1, "AC Sources – Operating," was not adequate to preserve plant safety and applied the provisions of Technical Specification Surveillance Requirement 3.0.3, and Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety." The licensee did not complete the necessary actions to correct the deficient technical specification by submitting an adequate license amendment request. The inspectors concluded the most significant contributor to the finding was a less than adequate engineering evaluation to support the new emergency diesel generator loading profiles following the previous violation. The licensee entered the performance deficiency into the corrective action program as Notification 50232181.

The inspectors determined that the performance deficiency is more than minor because if left uncorrected, the failure to implement prompt corrective actions has the potential to lead to a more significant safety concern. The inspectors concluded the finding was of very low safety significance because the finding was a design deficiency confirmed not to result in the loss of operability or functionality. The finding is associated with the Mitigating Systems Cornerstone. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative technical specification such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 26, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Report a Condition that Could Have Prevented the Fulfillment of a Safety Function**
The inspectors identified a noncited violation of 10 CFR 50.73(a)(2)(i)(B) and 10 CFR 50.73(a)(2)(v)(B) and after Pacific Gas and Electric failed to submit a required licensee event report within 60 days following discovery of a condition prohibited by the plant technical specifications and a condition that could have prevented the fulfillment of a safety function. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme, required by Technical Specification 3.3.5, "Loss of Power Diesel Generator Start Instrumentation," was inadequate to protect operating engineering safety feature pump motors. The licensee concluded that sustained degraded voltage could result in an overcurrent condition affecting equipment powered from the preferred offsite power supply. This condition was required to be reported to the NRC because the degraded voltage protection scheme rendered engineered safety feature pumps inoperable for a period in excess of the allowable technical specification out of service time and the condition resulted in the loss of the degraded voltage protection scheme safety function on all three vital 4 kV power buses.

The inspectors evaluated this finding using the traditional enforcement process because the failure to submit a required event report affected the NRC's ability to perform its regulatory function. The inspectors concluded the violation was a Severity Level IV because the licensee failed to submit an adequate licensee event report. The inspectors determined that the violation was also a finding under the reactor oversight process because licensee personnel failed to adequately evaluate a condition adverse to quality for operability and reportability, as required by station procedures. The inspectors concluded that the finding is more than minor because the failure to properly evaluate degraded plant equipment for past operability and reportability could reasonably be seen to lead to a more significant condition. The inspectors concluded that the finding had very low safety significance because the failure to adequately evaluate the condition did not result in an actual loss of a system safety function or equipment required by technical specifications, or involve the loss or degradation of equipment specifically designed to mitigate a seismic, flooding, or severe weather initiating event, and did not involve the total loss of any safety function that contributes to an external event initiated core damage accident sequence. This finding has a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the degraded voltage protection scheme such that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Inspection Report# : 2010003 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.59 for the licensee's failure to demonstrate that prior NRC approval was not required prior to making changes to the facility degraded voltage protection scheme as described in the Final Safety Analysis Report Update. In response to this violation, the licensee re-performed the corresponding safety analysis to demonstrate that the subject change to the facility degraded voltage protection scheme was consistent with General Design Criteria 17, "Electric Power Systems." The violation is in the licensee's corrective action program as Notification 50306053.

The failure of Pacific Gas and Electric to perform a 10 CFR 50.59 evaluation of modifications to the offsite power protection scheme, in accordance with NEI 96-07, was a performance deficiency. The violation was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The violation screened as very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding has a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee did not adopt the requirement to demonstrate that the proposed action was safe in order to proceed rather than a requirement to demonstrate that the proposed action was unsafe in order to disapprove the action, in that the Plant Safety Review Committee did not require that a 50.59 evaluation be performed to demonstrate that the proposed action was safe in order to proceed [H.1(b)].
Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Adequately Evaluate Changes to the Diesel Testing as Described in the Final Safety Analysis Report Update**
The team identified two examples of a Severity Level IV noncited violation of 10 CFR 50.59 after the licensee failed to perform an adequate evaluation to demonstrate that prior NRC approval was not required before making changes to the frequency and voltage recovery criteria and to the diesel testing commitments as described in the Final Safety Analysis Report Update. Specifically, the 1998 Final Safety Analysis Report Update identified a change from Safety Guide 9 to Regulatory Guide 1.9, Revision 2. The scope involved the removal of the KWS delay and included new requirements for voltage and frequency response. This resulted in a reduction in acceptance criteria. The team also identified a second example where the licensee failed to evaluate the 2005 Final Safety Analysis Report Update change from Regulatory Guide 1.9, Revision 2 to Revision 3 for diesel testing and interval frequency. Using NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, the team concluded that these changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases. In addition, the licensee's 50.59 evaluation, for DCP E-049425, Revision 0, "EDG Starting, and Loading Capability" was less than adequate to conclude that prior NRC approval was not required for the changes. The licensee has entered these issues into their corrective action program as Notification 50302481.

The failure of Pacific Gas and Electric to perform an adequate 10 CFR 50.59 evaluation prior to changing the facility as described in the Final Safety Analysis Report Update is a performance deficiency. The violation was more than minor because of a reasonable likelihood the change to the facility would require Commission review and approval prior to implementation. The violation screened as very low safety significance (Green) because the finding was not a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent a loss of system safety function, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The finding has a crossuctting aspect in the area of problem identification and resolution associated with the corrective action program component. In License Amendment Request 10-01, dated February 24, 2010, the licencee did not thoroughly evaluate the original problem of using the 10 CFR 50.59 evaluation process to justify using Regulatory Guide 1.9, Revision 2, Section C, Position 4, as an exception to meeting the frequency and voltage criteria identified in Safety Guide 9 [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-Conservative Decision Making resulted in a Violation of Technical Specification**
On March 10, 2010, the inspectors identified a noncited violation of Technical Specification 3.7.7, "Vital Component Cooling Water System," after both Unit 2 component cooling water loops were inoperable longer than permitted during power operations. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme was inadequate to ensure minimum required voltage would be available to operating engineered safety feature pumps during a degraded offsite power grid. The licensee concluded that operating pumps could trip and lock out on over current before the protection scheme would automatically transfer power to the emergency diesel generators. The licensee declared the 230kV offsite power systems inoperable and took compensatory actions to enable the automatic transfer of busses with operating engineered safety feature pumps directly to the diesel generators following a unit trip. On March 10, 2010, the inspectors identified that operating component cooling water pump 2-3 was still aligned to automatically transfer to 230kV offsite power source following a unit trip. The licensee had previously removed component cooling water pump 2-2 from service for maintenance on March 7, 2010. Technical Specification 3.7.7, "Vital Component Cooling Water System," required a minimum of two operable component cooling water pumps to establish operability of a vital component cooling water loop. Contrary to Technical Specification 3.7.7, on March 10, 2010, the licensee operated Unit 2 without an operable vital component cooling water loop for greater than 14 hours. The licensee has entered this issue into their corrective action program as Notification 50304802.

Either the failure of Pacific Gas and Electric to restore at least two operable component cooling water pumps or to have placed Unit 2 in Mode 3 within six hours, as required by plant Technical Specification 3.7.7, was a performance deficiency. The performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of equipment performance, of ensuring the availability, reliability, and capability of safety systems that respond to initiating events to prevent undesirable consequences (i.e., core damage), and it was within the licencee's ability to correct this problem. The inspectors used Inspection Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," to analyze the finding because the violation represents the actual loss of safety function for greater than the technical specification allowed outage time. The finding was of very low safety significance (Green) based on a bounding qualitative evaluation using Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," The inspectors based this conclusion on the low probability of an actual degraded grid condition coincidental with an accident or anticipated operational occurrence during the 14-hour exposure that the vital component cooling water loops were unavailable due to the performance deficiency. The inspectors concluded that this finding had a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the degraded voltage protection scheme [H.1(b).

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update the Final Safety Analysis Report Update with the Current Plant Design Bases**
The inspectors identified a Severity Level IV noncited violation of 10 CFR 50.71 after Pacific Gas and Electric failed to include the current plant design basis for the 230kV degraded voltage protection scheme in the Final Safety Analysis Report Update. Title 10 CFR 50.71 (e) states in part, "Each person licensed to operate a nuclear power reactor shall update periodically, as provided in paragraphs (e) (3) and (4) of this ection, the Final Safety Analysis Report Update originally submitted as part of the application for the operating license, to assure that the information included in the report contains the latest information developed. Contrary to the above, on March 14, 2010, the

inspectors identified that Pacific Gas and Electric failed to update the Final Safety Analysis Report Update to include complete design basis information for the offsite degraded voltage protection scheme. The inspectors identified that Final Safety Analysis Report Update did not include the design basis for the allowance time delay or the limiting voltage setpoints. The licensee has entered this issue into their corrective action process as Notification 50313763.

Failure to include the current plant design basis for the 230kV degraded voltage protection scheme in the Final Safety Analysis Report Update is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this issue was to be ealuated using the traditional enforcement process because the performance deficiency was a failure to meet a requirement or standard, had the potential for impacting the NRC's ability to perform its regulatory function, and the concern was within the licensee's ability to foresee and correct and should have been prevented. The team used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I "Reactor Operations," dated January 14,2 005, to evaluate the significance of this violation. The team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation because the erroneous information was not used to make any unacceptable change to the facility or procedures. Using Inspection Manual Chapter 0609.04, “Phase 1 Initial Screening and Characterization of Findings,” the team concluded that the issue screened as having low safety significance (Green) under the Significance Determination. Because this violation is of very low safety significance and was entered into the licensee’s corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. Because the violation included a performance deficiency, the inspectors also concluded the issue was a finding under the Reactor Oversight Process and had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately evaluate the extent of the condition and take appropriate corrective actions after the NRC identified a similar violation [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination Associated With the Offsite Degraded Voltage Protection Scheme**
On February 27, 2010, the inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to complete an adequate operability evaluation, as required by Procedure OM7.ID12, “Operability Determination,” Revision 14. The inspectors identified that the offsite power degraded voltage protection scheme time delay was inconsistent with key assumptions in the accident analysis. The licensee entered this nonconforming condition into the corrective action program as Notification 50301167 on February 24, 2010. Plant operators subsequently requested plant engineering to perform an operability determination of the nonconforming condition per Operability Determination Procedure OM7.ID12. On February 27, 2010, the plant operating authority concluded that the protection scheme was operable based on the information provided in the operability determination. Contrary to the above, on March 2, 2010, the inspectors concluded that the licensee’s operability determination was inadequate to demonstrate protection scheme operability and was not performed as required by Operability Determination Procedure OM7.ID12. Plant engineering only addressed the capability of the protection scheme at normal grid voltage following a mechanical failure of the 230 kV load tap changer. Operability Determination Procedure OM7.ID12, Section 5.3, “Write the Prompt Operability Assessment (POA),” required that the licensee address the potential effect of the nonconforming condition to perform the specified safety function. The licensee has entered this finding into the corrective action program as Notification 50319258.

Failure to complete an adequate operability evaluation, as required by Procedure OM7.ID12, “Operability Determination,” Revision 14, is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality, and the failure to perform an adequate operability evaluation affects the ability to ensure operability of the protection scheme at normal grid voltage following a mechanical failure of the 230 kV load tap changer. The inspectors used Inspection Manual Chapter 609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” to analyze the finding because the violation represent the actual loss of safety function for greater than the Technical Specification 3.3.5 allowed outage time. Using Appendix M, of the "Significance Determination Process Using Qualitative Criteria," the inspectors concluded that the finding was of very low safety significance (Green) based on a bounding qualitative evaluation. The inspectors based this conclusion on the low probability of an actual degraded grid condition coincidental with an accident or anticipated operational occurrence during the exposure time that protection scheme was available due to the performance deficiency. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming condition for operability and reportability [P.1(c)].

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Second Level Undervoltage Relay Time Delay to Initiate Load Shed and Sequencing Upon the Diesel Generator is Adequate to Assure Plant Safety**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure that plant conditions were consistent with design calculation inputs and assumptions. The licensee failed to assure and verify that Technical Specification 3.3.5 (SR3.3.5.3) pertaining to the second level undervoltage relay time delay to initiate load shed and sequencing upon the diesel generator was adequate to assure plant safety. Supplemental Safety Evaluation Report 09, Section 8.1, requires that a second level of under voltage protection for the onsite power system be provided. Subsection (1)(c)(i), reads: "The allowable second level undervoltage relay time delay, including margin, shall not exceed the maximum time delay that is assumed in the Final Safety Analysis Report Update accident analyses." Contrary to the above, as of March 4, 2010, the licensee failed to adequately implement the requirements of Supplemental Safety Evaluation Report 09. The second level undervoltage relay time delay setpoint for the emergency diesel generator of less than or equal to 20 seconds, assuming a safety injection signal concurrent with a degraded off site power source, exceeded the Final Safety Analysis Report Update accident analysis. This item is identified in the licensee's corrective action document Notification 50301167.

Failure to ensure that plant conditions were consistent with design calculation inputs and assumptions is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix E, Section 3 Example j, the violation was determined to be more than minor because the engineering calculation error results in a condition where there is now a reasonable doubt on the operability of a system or component. These deficiencies represented reasonable doubt regarding the mitigation of an accident by being in an unanalyzed condition. Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 2, the finding was determined to have very low safety significance (Green), did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. The team reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance:** G Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Drawings and Procedures to Align Emergency Makeup Water Supply from Diablo Canyon Creek to Support the Auxiliary Feedwater System**
The team identified a noncited violation of Diablo Canyon Technical Specification 5.4.1. "Procedures," for failure to have a procedure. The Diablo Canyon Final Safety Analysis Report Update, Revision 18, Section 6.5.2.1.1 documents the design of the auxiliary feedwater system and credits eight sources of water that can provide backup means of supply in the event that its primary source of water, the condensate storage tank, becomes exhausted. One of the sources included is the Diablo Canyon Creek. Diablo Canyon Technical Specification 5.4.1 states: "Written procedures shall be established, implemented, and maintained covering the following activities: [a.] The applicable procedures recommended in NRC Regulatory Guide 1.33, Revision 2, Appendix A, February 1978". NRC Regulatory Guide 1.33, Revision 2, Appendix A, describes procedures under Items 3l (instructions for shutdown, startup, and operation, including system filling, of the auxiliary feedwater system) and 6j (loss of feedwater system or feedwater

system failure). Contrary to Technical Specification 5.4.1, on March 4, 2010, the team identified that the licensee did not have an established procedure for accomplish the task identified in the Final Safety Analysis Report Update, Section 6.5.2.1.1 for taking water from the Diablo Canyon Creek to be a supply for the auxiliary feedwater system. This item is identified in the licensee's corrective action document Notification 50298563.

Failure to provide a procedure or instructions and acceptance criteria to perform an emergency makeup water alignment to the auxiliary feedwater system is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the performance deficiency is more than minor because it is associated with the Mitigating Systems Cornerstone attribute of procedure quality, and the lack of having this procedure affects the ability to ensure the availability, reliability, and capability of the auxiliary feedwater system to respond to initiating events to prevent undesirable consequences, (i.e., core damage, and it was within the licensee's ability to correct this problem.) Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding was determined to have very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a technical specification allowed outage time, and did not affect external event mitigation. The team concluded that this finding had a crosscutting aspect in the area of problem identification and resolution, in that the licensees' corrective action program thoroughly evaluates problems such that the resolutions address causes and the extent of conditions, as necessary. Per licensee Notification 50298563, changes were made to pumping systems associated with the Diablo Canyon Creek in 2007, which affected the ability to pump water through the discussed credited lineup supporting the auxiliary feedwater system. This effect was not identified as part of the changes, so no review of procedures related to the emergency auxiliary feedwater system alignment in question was performed. Since these actions occurred within the last three years, this performance characteristic reflects current performance [P.1 (c)].

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Feedwater Rupture Accident Analysis in the Final Safety Analysis Report Update**
The team identified a Severity Level IV noncited violation of 10 CFR 50.71, "Maintenance of records, making of reports." Paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." In the Diablo Canyon Final Safety Analysis Report Update section addressing the feedwater line break accident, it states that operator actions are credited with precluding the operation of pressurizer safety valves based on determinations in Westinghouse study WCAP-11667 (1998) (Final Safety Analysis Report Update, Section 15.4.2.2.2). Review of this study, and associated correspondence on the topic during 2006 indicated that the Westinghouse study did not state that operator actions could be credited for this event, but analysis of the worst case pressurizer overfill accidents by the licensee may show that this is the bounding case for such accidents, and that it did not need to be addressed in the feedwater line break analysis. In 2006, the licensee indicated that they would revise the Final Safety Analysis Report Update text to remove this reference to the Westinghouse study, which had been in the Final Safety Analysis Report Update since Revision 16. Contrary to the above, since 2006 (Final Safety Analysis Report Update Revision 16), the licensee failed to update Final Safety Analysis Report Update, Section 15.4.2.2.2. The licensee has entered this issue into their corrective action process as Notification 50301747.

Failure to periodically update the Final Safety Analysis Report Update with a known error is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this issue was to be evaluated using the traditional enforcement process because the performance deficiency was a failure to meet a requirement or standard, had the potential for impacting the NRC's ability to perform its regulatory function, and the concern was within the licensee's ability to foresee and correct and should have been prevented. The team used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, "Reactor Operations," dated January 14, 2005, to evaluate the significances of this violation. The team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The inspectors reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

**Significance: SL-IV** Jun 10, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Text to Reflect Credited Design Class I Makeup Flowpath to Component Cooling Water Expansion Tank in the Final Safety Analysis Report Update**
The team identified a Severity Level IV noncited violation of 10 CFR 50.71, "Maintenance of records, making of reports." Title 10 CFR 50.71, paragraph (e) states, "Each person licensed to operate a nuclear power reactor shall update periodically the final safety analysis report originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." In the Final Safety Analysis Report Update Table 9.2-7, Component 5, it states, "This 250 gpm, Design Class I, makeup water flowpath, described under Makeup Provisions in Subsection 2.3.3 (Section 9.2.2.3.3), can be started within 10 minutes." Final Safety Analysis Report Update, Section 9.2.2.3.3 states, "All piping and valves in the makeup path from the condensate storage tank (including their cross-connections) and the firewater tank, through the makeup water transfer pumps up to and including the makeup valves on the component cooling water system lines, are Design Class I." Text later in the section implied that the flow path from the firewater tank was not Design Class I. Review by the licensee staff revealed that the only Design Class I flow path to provide makeup to the component cooling water expansion tank is via the condensate storage tank. This revealed that the text provided in Final Safety Analysis Report Update, Section 9.2.2.3.3 stating that both the condensate storage tank and firewater tank makeup paths are credited is incorrect. Contrary to above, since 1984 (Final Safety Analysis Report Update, Revision 0), the licensee did not update Final Safety Analysis Report Update, Section 9.2.2.3.3 to correct the error of including firewater as a possible makeup path to the component cooling water expansion tank. The licensee has entered this issue into their corrective action process as Notification 50301884.

Failure to periodically update the Final Safety Analysis Report Update with a known error is a performance deficiency. Using Inspection Manual Chapter 0612, Appendix B, the team determined that this performance deficiency was to be evaluated using the traditional enforcement process because the performance deficiency had the potential for impacting the NRC's ability to perform its regulatory function. Using General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, Reactor Operations, dated January 14, 2005, to evaluate the significances of this violation, the team concluded that the violation is more than minor because the incorrect Final Safety Analysis Report Update information had a potential impact on safety and licensed activities. Using Supplement I, Section D, Item 6, of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. The team reviewed the finding for crosscutting aspects and none were identified.

Inspection Report# : 2010007 *(pdf)*

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# Barrier Integrity

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Design Basis of Containment Fan Cooler Unit Cooling Coil Casings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, prior to December 15, 2010, the licensee failed to assure that the design basis function of the containment fan cooler unit casings was translated into specifications, drawings, procedures, and instructions. The licensee has entered this violation into the corrective action program as Notification 50384801.

The inspectors determined that the failure to establish measures to assure that the design basis function of the containment fan cooler unit cooling coil casings was translated into specifications, drawings, procedures, and instructions was a performance deficiency. The finding was more than minor because it adversely affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 4, "Initial Screening and Characterization of Findings," the finding was determined to be of very low safety significance (Green) because it did not represent a degradation of the barrier function of the control room against a smoke or toxic barrier, an open pathway in the physical integrity of reactor containment, or an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined that this finding has a crosscutting aspect in the area of human performance because the licensee failed to ensure that personnel, equipment, procedures and other resources were available to assure nuclear safety by maintaining complete, accurate and up-to-date design documentation.
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Auxiliary Building Ventilation System Control Panel Modification**
The inspectors reviewed a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the design basis requirements for single failure criteria were correctly translated into auxiliary building ventilation system controls modifications. On January 10, 2011, a single failure of a Unit 2 auxiliary building ventilation Train “A” damper resulted in the loss of system safety function for both trains. The loss of safety function occurred because of a logic error in the programmable logic controllers. The licensee programmed and installed the logic controllers in November 2010 for Unit 1 and in November 2009 for Unit 2. The inspectors identified that the engineering department performed a less than adequate review to identify the single point vulnerability during the modification review process. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50370698, replaced the failed damper, and implemented compensatory actions to mitigate the design deficiency. The licensee plans to implement corrective actions to program the logic controller program consistent with the design basis requirements.

The inspectors concluded that the failure to ensure that the modification met design basis requirements was a performance deficiency. This performance deficiency is more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers and radiological barriers, including the Auxiliary Building, protect the public from radionuclide releases caused by accidents or events. The inspectors determined that the finding had very low safety significance because the finding only represents degradation to the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance associated with work practices because the licensee did not ensure human error prevention techniques, such as self and peer checking, were effectively used in the preparation of the modification.
Inspection Report# : 2011002 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

**Significance: N/A** Jul 27, 2010
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**
The team concluded that the notification process facilitates the initiation, tracking, and trending of concerns and that the licensee correctly identified deficiencies that were conditions adverse to quality and entered them into the corrective action program in accordance with the licensee's corrective action program guidance and NRC requirements. Prioritization of issues was appropriate. The licensee was inconsistent in the effectiveness of evaluating issues once they were identified. The team's assessment was there was limited effective interdepartmental communication, a lack of cross discipline peer checks, and a failure to assign the appropriate resources to evaluate cross-departmental problems/issues. As a result, the licensee's performance in resolving problems and effective utilization of operating experience was negatively impacted. The licensee performed effective quality assurance audits and self-assessments, as demonstrated by self-identification of poor corrective action program performance and identification of ineffective corrective actions. However, because of challenges in performing evaluations, the licensee had difficulty properly addressing some of these issues. Overall the team concluded that implementation of the corrective action program was adequate with improvements warranted.

The team determined that site personnel were willing to raise safety issues and document them in the corrective action program. The team noted that workers at the site felt free to report problems to their management and the NRC, but were reluctant to take safety concerns to the Employee Concern Program. Additionally, the function and processes associated with the Employee Concern Program was not understood by a majority of the personnel interviewed.

Inspection Report# : 2010006 *(pdf)*

Last modified : June 07, 2011

# Diablo Canyon 2
# 2Q/2011 Plant Inspection Findings

## Initiating Events

## Mitigating Systems

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Fire Hazard Evaluations**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to implement the required compensatory actions described in Equipment Control Guideline 18.7, “Fire Rated Assemblies.” On December 28, 2010, the licensee blocked open Fire Doors 175 and 182-2, entrances to the Unit 1 and 2 safety injection pump room to address auxiliary building ventilation flow balance problems. The supporting engineering evaluation failed to identify that the doors were rated fire barriers as described in the fire hazard analysis. If a fire had occurred, these blocked open doors would have allowed smoke and hot gases to pass from fire area AB-1 to impact equipment in adjacent fire areas 3-B-2 (Unit 1) and 3-D-2 (Unit 2). Equipment Control Guideline 18.7 required the licensee to either establish a continuous fire watch on at least one side of the inoperable fire doors or verify that the fire detection or automatic suppression system on at least one side of the fire doors was operable and establish an hourly fire watch. The licensee took corrective actions to establish the required fire watches and enter the finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of Pacific Gas and Electric to maintain the fire doors in the rated configuration as described in the Final Safety Analysis Report Update “Fire Hazard Analysis,” was a performance deficiency. This finding was more than minor because the degraded fire barriers affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only affected the ability to reach and maintain cold shutdown conditions. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems associated with modification of the safety injection pump room fire doors such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation of New Security Modifications**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to adequately evaluate the impact of protected area boundary modifications. These modifications affected the ability of plant operators to transfer water from the raw water storage reservoirs to the auxiliary feedwater system using temporary hoses. Plant engineers authorized a series of security modifications which included the installation of physical intrusion barriers, including delay fences and razor wire between the raw water reservoirs and the auxiliary feedwater system. The licensing basis evaluation did not address raw water makeup to the auxiliary feedwater system using temporary hoses as described in Final Safety Analysis Report Update Section 6.5, “Auxiliary Feedwater System,” and Section 3.7.6, “Seismic Evaluation to Demonstrate Compliance with the Hosgri Earthquake Requirements Utilizing a Dedicated Shutdown Flowpath.” The licensee took immediate corrective actions to establish a route for the temporary hoses, including preplanned security compensatory measures, and entered this finding into the corrective action program as Notification 50410997.

The failure to adequately evaluate the impact of the security modifications on the plant licensing and design bases was a performance deficiency. This performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of Problem Identification and Resolution, associated with the Corrective Action Program component, because the licensee failed to thoroughly evaluate the security modifications such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Preferred Offsite Power System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the preferred offsite power system design basis was correctly translated into electrical dynamic loading Calculations 357A-DC, “Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis,” Revision 12 and 359-DC, “Offsite Power Dynamic Analysis,” Revision 8. The licensee did not include the limiting load flow cases representing the largest total onsite demand for both units as required by the plant design basis. On July 7, 2010, the NRC clarified that the Diablo Canyon current licensing basis required the preferred offsite power system to have adequate capacity and capability to supply the most limiting loading requirements, including a dual unit trip. The licensee subsequently entered the condition into the corrective action program as Notification 50289590 and revised the station dynamic loading analysis to reflect the increased onsite power demand. The inspectors concluded that the failure to ensure that the dynamic loading analysis included all design basis requirements was a performance deficiency. This performance deficiency is more than minor because the finding was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because the inspectors were unable to conclude that the preferred offsite power system had not been inoperable for greater than the allowed Technical Specification outage time, a senior reactor analyst performed a bounding Phase 3 analysis. The Phase 3 analysis demonstrated that the subject finding was of very low safety significance (Green), because of the small increase of probability of a loss of offsite power that the finding represented. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the current licensing basis requirements to ensure that resolutions addressed causes and extent of conditions, as necessary. [P.1(c)]
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain the integrity of Door 155 in the rated condition. On December 9, 2010, the inspectors identified that the fire door was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain Door 155 in a configuration that would provide at least a 1½-hour rated fire barrier. The inspectors previously identified that Door 155 was degraded as a fire barrier in 2009. The licensee entered the violation into the corrective action program as Notification 50367381 and took immediate corrective actions to restore the fire barrier to the rated condition and to implement weekly plant fire door walkdowns.

The inspectors concluded that the finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute and objective to prevent undesirable consequences due to fire. The inspectors determined that the finding was within the fire confinement category and that the fire barrier was moderately degraded. The inspectors concluded that the finding was of very low safety significance (Green) because there was a non-degraded automatic full area water-based suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take effective corrective actions to following the previous occurrence of the violation [P.1(d)].

Inspection Report# : 2010005 *(pdf)*

**Significance:** Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Transient Combustibles Procedure**
The inspectors identified a noncited violation of Diablo Canyon Unit 2 Facility Operating License Condition 2.C.(5), “Fire Protection,” after Pacific Gas and Electric failed to ensure procedures for controlling flammable and combustible materials adequately incorporated requirements of the fire hazard analysis. On October 18, 2010, the inspectors identified that transient combustible materials staged in the Unit 1 12 kilovolt switchgear room did not have an approved transient combustibles permit. The licensee stated that the combustibles permit procedure did not require a permit for the room while Unit 1 was shutdown. However, the plant fire hazards design basis described safe shutdown equipment in the room that would be needed to support a safe shutdown of the operating unit, specifically the Unit 2 startup bus located in the room. The inspectors determined that the licensee’s transient combustibles permit procedure was inadequate because the procedure did not require a permit for the Unit1 12 kilovolt switchgear room when Unit 2 was operating. The licensee entered the issue into the corrective action program as Notification 50366302 and performed an evaluation of the transient combustibles stored in the area.

The inspectors concluded that this finding was more than minor because it affected the Mitigating Systems Cornerstone external factors attribute objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors determined that the finding was within the fire prevention and administrative controls category and represented a low degradation level due to the minimal impact on the effectiveness and reliability of the affected systems. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening, the low degradation rating, and only equipment needed to reach and maintain cold shutdown conditions was affected. This finding had a crosscutting aspect in the area of human performance associated with the resources component because the licensee failed to ensure that the design documentation adequately identified the Unit 2 startup bus as equipment required for safe shutdown for Unit 2 [H.2 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determinations**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to adequately evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, “Operability Determination.” On October 15, 2010, the inspectors identified a less than adequate technical evaluation supporting Prompt Operability Assessment 50350918, “Unit 2 - Insulation in Bio-Wall Penetration.” Engineering personnel failed to adequately evaluate the extent of condition after technicians identified about 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous insulation in the Unit 1 reactor coolant loop biological shield wall penetrations. This fibrous material could have potentially been transported and plugged the emergency core cooling containment sump screen. The licensee performed the prompt operability assessment for Unit 2, which was operating at the time. The inspectors concluded that the engineering personnel inappropriately applied the leak-before-break methodology to exclude about 87 percent of this material from the extent of condition review in the prompt operability assessment.
The second example involved Prompt Operability Assessment Notification 50355265, “RHR Sump Margin,” which was completed by the licensee on October 23, 2010. In this example, engineering personnel failed to identify and demonstrate that the specified safety function of the refueling water storage tank could be maintained as required by the plant operability procedure. The inspectors identified that the post accident flow path from the reactor cavity to the

containment sump was blocked by a large shield plug. This blockage
reduced the amount of post accident inventory available at the containment sump at the time of transition from injection to recirculation mode of emergency core cooling operation. Engineering personnel failed to demonstrate that the safety function to ensure full sump submergence was maintained with the blocked flow path. Full submergence of the sump was used by the NRC as the basis for approval of Technical Specification 3.5.4, “Refueling Water Storage Tank,” inventory requirements. The licensee entered the violation into the corrective action program as Notification 50369117 and revised the prompt operability assessments using assumptions consistent with the current licensing bases.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because Pacific Gas and Electric did not use conservative assumptions in decisions to demonstrate component operability in either example [H.1(b)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Containment Recirculation Sump Design Control**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure Calculation STA-255, “Minimum Required Refueling Water Storage Tank Level for GE Sumps,” Revision 2, demonstrated adequate available refueling water storage tank inventory. On October 19, 2010, the inspectors identified that emergency core cooling post accident flow path from the reactor cavity to the containment sumps was blocked by a large steel plug on Unit 1. The accident analysis assumed this 35 square foot path was open to allow coolant from a pipe break inside the biological shield to communicate with containment sumps during the recirculation mode of emergency core cooling. The licensee credited the inventory from the reactor cavity when determining the minimum required refueling water storage tank volume in Calculation STA-255. Pacific Gas and Electric used Calculation STA-255 as the basis for determining the minimum required refueling water storage tank volume specified by Technical Specification 3.5.4, “Refueling Water Storage Tank.” The inspectors identified that the recirculation flow path was also blocked on Unit 2. The inspectors concluded that the most significant contributor to the violation was inaccurate plant drawings used by plant engineers during the performance of Calculation STA-255. The licensee’s corrective actions included completion of a prompt operability assessment justifying continued operation of Unit 2 and replacement of the shield plug with a movable platform on Unit 1 prior to plant restart.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone plant modification design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the performance deficiency involved a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the resources component because Pacific Gas and Electric failed to use complete, accurate and up-to-date drawing for Calculation STA 255 [H.2(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Emergency Diesel Generator Surveillance Testing**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” after Pacific Gas and Electric failed to develop and implement an adequate testing program for the emergency diesel generators that met design requirements and recommendations. Specifically, in December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design bases were

met. Pacific Gas and Electric updated the load calculations, but failed to make the necessary revisions to Surveillance Test Procedure STP M-9D1, “Diesel Generator Full Load Rejection Test.” As a result, Pacific Gas and Electric failed to test several of the emergency diesel generators at the complete load as required by Regulatory Guide 1.108, Revision 1, which is part of the current licensing bases. The licensee entered this into the corrective action program as Notification 50368801, determined there was no loss of safety function for the affected components, and applied the provisions of Surveillance Requirement 3.0.3 for a missed surveillance test. The inspectors concluded the most significant contributor to the finding was less than adequate diesel generator loading evaluations to support corrective action from previous violations associated with the emergency diesel generator testing.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the equipment control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative surveillance test such that the resolution addressed the fundamental basis for the surveillance [P.1 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Quality Verification Audits**
The inspectors identified a noncited violation of 10CFR Appendix B, Criterion XVIII, “Audits”, which required that a comprehensive system of planned and periodic audits be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program as well as follow up action, including re-audit of deficient areas, where indicated. Contrary to this requirement, Pacific Gas and Electric failed to ensure that a comprehensive system of planned and periodic audits were carried out to verify compliance with all aspects of the quality assurance program, determine the effectiveness of the program, and perform necessary follow up actions. Specifically, the 2008 Quality Verification audit of the corrective action program failed to adequately address an adverse trend in the problem evaluation process documented in NRC Inspection report 2008005, which identified eleven examples of an adverse trend in problem evaluation. The licensee entered this into their corrective action program as Notification 50365083 and determined
there was no loss of safety function for the affected components. The inspectors concluded the most significant contributor to the finding was a less than adequate evaluation of the corrective action trending program.

This finding was more than minor because it was associated with the equipment control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the performance deficiency was of very low safety significance (Green) it was a deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to coordinate and communicate the results from assessments to affected personnel, and track the corrective actions to address issues commensurate with their significance [P.3(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Identify a Degraded Fire Barrier**
The inspectors identified a noncited violation of the Diablo Canyon Facility Operating License Condition (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain the integrity of a fire door in the rated configuration. On August 19, 2010, the inspectors identified that Fire Door 223 was inoperable. Fire Door 223 was required to provide a 3-hour rated barrier between Fire Areas 5-A-4 and 5-B-4. A fire in either of these areas could have prevented

operation of the auxiliary feedwater, auxiliary saltwater, or component cooling water pumps or steam generator level control from the remote shutdown panel. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to either maintain Fire Door 223 operable or implement compensatory actions within one hour. The inspectors concluded the most significant contributor to the finding was that licensee personnel did not identify and enter the degraded fire door into the Corrective Action Program. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notification 50336901 and completed repairs to the door on August 23, 2010.

The inspectors concluded that the performance deficiency was more than minor because the degraded fire barrier affected the mitigating systems cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors determined that the inoperable door was a fire confinement category finding and that the fire barrier was moderately degraded because the door would not perform the rated fire barrier function. The inspectors concluded the finding was of very low safety significance because the degraded barrier would have provided a minimum of 20 minutes fire endurance protection and ignition sources and combustible materials were positioned that had a fire spread to secondary combustibles, the degraded barrier would not have been subject to direct flame impingement. This finding has a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not implement a low threshold for identifying and entering issues into the Corrective Action Program [P.1(a)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: FIN Finding
**Inadequate Risk Management During a Planned Auxiliary Saltwater Pump Outage**
The inspectors identified a finding after Pacific Gas and Electric failed to adequately manage risk during planned maintenance activity as required by Procedure AD7.DC6, "On-line Maintenance Risk Management." On April 5, 2010, work control personnel requested that plant operators simultaneously remove Auxiliary Saltwater Pump 2-2 and Component Cooling Water Heat Exchanger 2-2 from service for two scheduled maintenance activities. Plant operators identified that the combination of the auxiliary saltwater pump and component cooling water heat exchanger out of service at the same time would result in an elevated maintenance risk (Yellow). Procedure AD7.DC6, "On-line Maintenance Risk Management", Section 2.1, required that the licensee manage plant risk during on-line maintenance by minimizing the number of risk significant equipment simultaneously removed from service. The inspectors concluded that these two maintenance activities could have been performed in series rather than in parallel without affecting the duration either component was unavailable for maintenance. The licensee entered the performance deficiency into the corrective action program as Notification 50309451.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," because the work scope unnecessarily placed the plant into a higher licensee-established risk category and required additional risk management actions. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, "Maintenance Risk Assessment and Risk Management Significance Determination Process." This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee failed to implement adequate corrective actions to prevent unnecessarily entering elevated plant risk for the planned maintenance [P.1(d)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Risk Assessment during Planned Maintenance Activities**
The inspectors identified a noncited violation of 10 CFR 50.65 after Pacific Gas and Electric failed to perform a risk

assessment after plant conditions had changed. On July 13, 2010, Pacific Gas and Electric identified that station personnel failed to complete Technical Specification Surveillance Requirement 3.3.4.2, “Remote Shutdown System,” within the specified frequency for both Units. As provided by Surveillance Requirement 3.0.3, the licensee performed a risk evaluation to extend the required surveillance completion time beyond twenty-four hours. The licensee initiated the missed surveillance tests and identified results were outside acceptance criteria. On July 26, 2010, Operations personnel declared several remote shutdown system functions inoperable because reasonable expectation no longer existed that remote shutdown system could perform its safety function. Pacific Gas and Electric failed to reassess the effect on plant risk resulting from inoperable remote shutdown system functions before continuing with scheduled maintenance. A subsequent risk assessment concluded that plant risk was in a higher risk category due to planned maintenance activities conducted during this time frame. The licensee entered the performance deficiency into the corrective action program as Notification 50331841.

The inspectors determined that the performance deficiency is more than minor because the performance deficiency affected the Mitigating Systems Cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Also, the finding is similar to Example 7.e in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues,” because the overall elevated plant risk would put the plant into a higher licensee-established risk category. The inspectors concluded that the finding is of very low safety significance (Green) based on an actual incremental core damage probability deficit of less than 1x10-6 and an evaluation using Flowchart 1 of Appendix K of Inspection Manual Chapter 0609, “Maintenance Risk Assessment and Risk Management Significance Determination Process.” This finding had a crosscutting aspect in the area of human performance associated with the work practices component because the licensee failed to follow its maintenance risk procedure and reassess plant risk due to changing plant conditions [H.4(b)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determination**
The inspectors identified a noncited violation of 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to promptly evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, “Operability Determination.” The first example involved the failure of engineering personnel to promptly notify plant operations of the failure of the emergency diesel generators to meet licensing and design frequency and voltage recovery requirements. This issue was identified by the NRC on May 11, 2010, but not evaluated for the effect on diesel operability until September 9, 2010. The second example also involved the failure of engineering personnel to promptly notify plant operations to evaluate a nonconforming condition associated with a common cross-tie line that connected both auxiliary saltwater trains. This issue was identified by the NRC on July 22, 2010, but not evaluated for the effect on auxiliary saltwater operability until August 4, 2010. In both examples, engineering personnel failed to follow Procedure OM7.ID12, “Operability Determination,” Section 5.1, which required any individual identifying a degraded or nonconforming condition that potentially impacts operability of a system, structure or component to ensure that operations shift management is informed. The licensee entered the performance deficiency associated with this finding into the corrective action program as Notifications 50340417 and 50335847.

The inspectors concluded that the performance deficiency is more than minor because the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. The finding was of very low safety significance (Green) because neither of the two examples was subsequently determined to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because Pacific Gas and Electric did not thoroughly evaluate the nonconforming conditions for operability [P.1(c)].
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Sep 25, 2010
Identified By: NRC

Item Type: NCV NonCited Violation
**Inadequate Design Control for the AuxiliarySaltwater System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to maintain adequate design control measures associated with the auxiliary saltwater system. The inspectors identified that the auxiliary saltwater system design did not comply with the plant design bases as described the Final Safety Analysis Report Update. Specifically, an auxiliary saltwater vent line did not meet the requirements established of General Design Criteria 1, “Quality Standards and Records,” and Regulatory Guide 1.26, “Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants.” The licensee entered the performance deficiency into the corrective action program as Notification 50328942.

This performance deficiency is greater than minor because the design control attribute of the mitigating systems cornerstone and the cornerstone’s objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences were affected. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Mitigating Systems Cornerstone, the inspectors concluded the finding was of very low significance (Green) because it was a design deficiency confirmed not to result in the loss of operability or functionality. The inspectors concluded that the finding does not have a crosscutting aspect since the performance deficiency is not reflective of current plant performance.
Inspection Report# : 2010004 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Emergency Diesel Generator**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to maintain adequate design control measures associated with the emergency diesel generating air system. Specifically, failure of non-seismically qualified air compressor unloader sensing lines during a seismic event could impact the safety function of the emergency diesel generators. Subsequent analysis of the nonconforming condition performed by the licensee determined the piping would not fail during a postulated seismic event. The licensee entered this issue into the corrective action program as Notifications 50307496, 50307497, 50307504, 50307670, 50308204, and 50308824.

The finding was more than minor because it affected the mitigating systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using the Significance Determination Process (SDP) Phase 1 Screening Worksheet for the Initiating Events, Mitigating Systems, and Barriers Cornerstones the finding was potentially risk significant for a seismic initiating event requiring a Phase 3 analysis. The analyst estimated the nonrecovery probabilities for operators failing to isolate air between the receiver and the compressor prior to air pressure depletion, and operators failing to manually open fuel transfer valves to makeup to the diesel day tank. The final quantitative result was calculated to be 1.06 x 10-6. However, using a qualitative evaluation of the bounding assumptions, the analyst determined that the best available information indicated that the finding was of very low risk significance (Green). The team determined that the finding was reflective of current plant performance because it had been recently identified during the license renewal inspection and had a human performance crosscutting aspect related to decision making because the licensee did not use conservative assumptions when evaluating this nonconforming condition in previous evaluations.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain Proficiency of Operators to Meet the Time Critical Operator Actions**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the failure to ensure that operators are able to implement specified actions in response to operational events and accidents. Specifically, operators could not achieve actions within the analysis time estimates for the cold leg recirculation phase of a loss of coolant accident response and the steam generator tube rupture response as described

in the licensee’s safety analysis report.

The finding is more than minor because it affected the Mitigating Systems Cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding represented a potential loss of a safety function requiring a Phase 2 analysis. Because the probability of human error is not effectively addressed by a Phase 2 analysis, a Phase 3 analysis was performed. The senior reactor analyst reviewed the actual timing of the walkdowns associated with the steam generator tube rupture time critical actions. The analyst determined that, while the licensee failed to meet the specific cooldown timing documented in the Final Safety Analysis Report, the total time to start cooling the reactor was well within the total critical timing of the event. The analyst found no impact on safety in delaying the cooldown of the reactor for one minute given that the other time critical actions were performed more quickly than required. Therefore, the analyst determined that this portion of the finding was of very low safety significance because it does not represent an actual loss of safety function (Green). The senior reactor analyst reviewed the issue related to the assumed action times associated with switching over to containment sump recirculation lineup for their emergency core cooling system pumps during a large break loss of coolant accident. The analyst noted that this time critical action was only required if a large-break loss of coolant accident occurred simultaneously with the failure of an residual heat removal pump to stop automatically, requiring local isolation of the pump. Given that the frequency of the initial conditions for the time critical action are below the Green/White threshold, the change in core damage frequency associated with this finding must be of very low safety significance (Green). The team determined that the finding was reflective of current plant performance because the licensee participated in a recent industry-wide study on time critical operator actions, but did not implement any of the group’s recommendations. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process related to verifying the validity of the underlying assumptions used to evaluate the feasibility of operators implementing time critical operator actions.

Inspection Report# : 2010006 *(pdf)*

**Significance: SL-IV** Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Submit Complete and Accurate Information for a Requested License Amendment**
The team identified a noncited violation of 10 CFR 50.9(a), “Completeness and Accuracy of Information” with multiple examples. Specifically, information supplied to the NRC in License Amendment Request 01-10, dated February 24, 2010, related to the revision of Technical Specification 3.8.1, "AC Sources - Operating," were not complete and accurate in all material respects. Following NRC questioning of the discrepancies the licensee withdrew the amendment request.

The finding is more than minor because the inaccurate information was material to the NRC. Specifically, this information was under review by the NRC to evaluate specific changes to the surveillance requirements associated with the emergency diesel generators. Following management review, this violation was determined to be of very low safety-significance because the amendment request was withdrawn before the NRC amended the facility technical specifications. Because this issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. Consistent with the guidance in Section IV.A.3 and Supplement VII, paragraph D.1, of the NRC Enforcement Policy, this finding was determined to be a Severity Level IV noncited violation. The finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program because the licensee did not adequately evaluate the extent of condition and take appropriate corrective actions after the NRC identified a similar violation.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Untimely and Inadequate Corrective Actions for the Emergency Diesel Generators**
The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” with two examples for the failure of the licensee to promptly identify and correct nonconforming conditions related to the

emergency diesel generators meeting the design basis. The first example resulted from the failure to identify that instrument inaccuracies were not accounted for in the bounding calculations. The second example involved the failure to identify that the worst case loading calculations exceeded the emergency diesel generator operating load limit.

The failure to promptly identify and correct the design deficiencies associated with the emergency diesel generators was a performance deficiency. This finding is greater than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone's objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, "Significant Determination Process," the team performed a Phase 1 analysis to analyze the significance of this finding and determined the finding is of very low safety significance because the condition was a design or qualification deficiency confirmed not to result in loss of operability or functionality, did not represent an actual loss of safety function of the system or train, did not result in the loss of one or more trains of nontechnical specification equipment, and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding had a crosscutting aspect in the area of human performance, decision making, because the licensee did not use conservative assumptions in the decision making process or conduct an adequate effectiveness review to verify the validity of the underlying assumptions for a safety-significant decision.

Inspection Report# : 2010006 *(pdf)*

**Significance:** G Jul 27, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Appropriately Evaluate Failed Residual Heat Removal Surveillance Test**
The team identified a noncited violation of Technical Specification 5.4.1.a for failure to appropriately evaluate and correct a condition adverse to quality, as instructed by Surveillance Test Procedure P-RHR-A22," Comprehensive Testing of Residual Heat Removal Pump." Specifically, the licensee failed to recognize a deviation in differential pressure towards the alert range, following the February 9, 2008, comprehensive surveillance test of the 2-2 residual heat removal pump. Continued degradation of the 2-2 residual heat removal pump resulted in failure of the October 9, 2009, comprehensive surveillance test due to the differential pressure exceeding the action limit. The licensee entered this issue into the corrective action program as Notification 50308225.

The finding is more than minor because it was associated with the equipment reliability attribute of the Mitigating Systems Cornerstone and it adversely affected the associated cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding in accordance with Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a for the Mitigating Systems Cornerstone. The finding was determined to be of very low safety significance (Green) because: (1) it was a design or qualification issue confirmed not to result in a loss of operability or functionality; (2) did not represent an actual loss of safety function of the system or train; (3) did not result in the loss of one or more trains of nontechnical specification equipment; and (4) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The team determined that this finding had a crosscutting aspect in the area of problem identification and resolution, corrective action program, because the licensee failed to appropriately evaluate the 2009 residual heat removal surveillance test failure such that the resolution identified and corrected the cause of the failure.

Inspection Report# : 2010006 *(pdf)*

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# Barrier Integrity

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: FIN Finding
**Inadequate Review of Severe Accident Management Guidelines**

The inspectors identified a finding after Pacific Gas and Electric failed to periodically review and update the severe accident management guidelines. Procedure OM10.ID5, "Severe Accident Management," required the licensee to review and update the severe accident management guidelines biennially to ensure that any changes in plant design or procedures, experience in severe accident management requalification training, and any changes in industry understanding of severe accidents were incorporated into the severe accident management guidelines. As a result of the licensee's failure to implement the periodic review, the severe accident management guidelines did not incorporate the latest owners' group guidance or recent plant design and hardware changes. The licensee took corrective actions to implement the biennial reviews and entered this finding into the corrective action program as Notification 50399554.

Pacific Gas and Electric's failure to follow procedural requirements for periodic review of the severe accident management guidelines was a performance deficiency. The finding was more than minor because if left uncorrected, the failure to review and update the severe accident management guidelines has the potential to lead to a more significant safety concern. This finding affected the barrier integrity cornerstone because the severe accident management guidelines are procedures that would be used to maintain the functionality of the containment should a severe accident occur. The inspectors concluded that the finding was of very low safety significance because it did not represent a degradation of the radiological, smoke, or toxic atmosphere barrier function; or represent an actual open pathway in the physical integrity of the reactor containment; or involve the function of the containment hydrogen igniters. The finding did not have any crosscutting aspects because the performance deficiency occurred more than three years ago and is not indicative of current licensee performance in that the licensee has improved the design review process since the performance deficiency occurred.
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Design Basis of Containment Fan Cooler Unit Cooling Coil Casings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, prior to December 15, 2010, the licensee failed to assure that the design basis function of the containment fan cooler unit casings was translated into specifications, drawings, procedures, and instructions. The licensee has entered this violation into the corrective action program as Notification 50384801.

The inspectors determined that the failure to establish measures to assure that the design basis function of the containment fan cooler unit cooling coil casings was translated into specifications, drawings, procedures, and instructions was a performance deficiency. The finding was more than minor because it adversely affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 4, "Initial Screening and Characterization of Findings," the finding was determined to be of very low safety significance (Green) because it did not represent a degradation of the barrier function of the control room against a smoke or toxic barrier, an open pathway in the physical integrity of reactor containment, or an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined that this finding has a crosscutting aspect in the area of human performance because the licensee failed to ensure that personnel, equipment, procedures and other resources were available to assure nuclear safety by maintaining complete, accurate and up-to-date design documentation. [H.2(c)]
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Auxiliary Building Ventilation System Control Panel Modification**
The inspectors reviewed a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to ensure that the design basis requirements for single failure criteria were correctly translated into auxiliary building ventilation system controls modifications. On January 10, 2011, a single failure of a Unit 2 auxiliary building ventilation Train "A" damper resulted in the loss of system safety function

for both trains. The loss of safety function occurred because of a logic error in the programmable logic controllers. The licensee programmed and installed the logic controllers in November 2010 for Unit 1 and in November 2009 for Unit 2. The inspectors identified that the engineering department performed a less than adequate review to identify the single point vulnerability during the modification review process. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50370698, replaced the failed damper, and implemented compensatory actions to mitigate the design deficiency. The licensee plans to implement corrective actions to program the logic controller program consistent with the design basis requirements.

The inspectors concluded that the failure to ensure that the modification met design basis requirements was a performance deficiency. This performance deficiency is more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers and radiological barriers, including the Auxiliary Building, protect the public from radionuclide releases caused by accidents or events. The inspectors determined that the finding had very low safety significance because the finding only represents degradation to the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance associated with work practices because the licensee did not ensure human error prevention techniques, such as self and peer checking, were effectively used in the preparation of the modification. [H.4(a)]
Inspection Report# : 2011002 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedures for Testing HEPA Ventilation Units**
The inspectors identified a noncited violation of Technical Specification 5.4.1(a) for the failure to follow procedures for testing and using the high-efficiency particulate air ventilation units used to prevent personal contamination. Licensee immediate actions included removing all high-efficiency particulate air ventilation units installed for the Unit 2 outage and testing all high-efficiency particulate air ventilation units as required by procedure. This matter was placed in the licensee's corrective action program as Notifications 50399479, 50399560, and 50399682.

This failure to follow procedures was a performance deficiency. The finding was more than minor because it was associated with the program and process attribute of the occupational radiation safety cornerstone. The finding affected the objective to ensure adequate protection of the worker's health and safety from exposure to unintended radiation from radioactive material during routine civilian nuclear reactor operation. Using the Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined the finding was of very low safety significance because (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This finding was determined to have a crosscutting aspect in the area of human performance, associated with work practices, because the licensee did not effectively communicate expectations regarding procedural compliance and the personnel following the procedures [H.4(b)].
Inspection Report# : 2011003 *(pdf)*

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## Public Radiation Safety

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# Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

**Significance: N/A** Jul 27, 2010
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**
The team concluded that the notification process facilitates the initiation, tracking, and trending of concerns and that the licensee correctly identified deficiencies that were conditions adverse to quality and entered them into the corrective action program in accordance with the licensee's corrective action program guidance and NRC requirements. Prioritization of issues was appropriate. The licensee was inconsistent in the effectiveness of evaluating issues once they were identified. The team's assessment was there was limited effective interdepartmental communication, a lack of cross discipline peer checks, and a failure to assign the appropriate resources to evaluate cross-departmental problems/issues. As a result, the licensee's performance in resolving problems and effective utilization of operating experience was negatively impacted. The licensee performed effective quality assurance audits and self-assessments, as demonstrated by self-identification of poor corrective action program performance and identification of ineffective corrective actions. However, because of challenges in performing evaluations, the licensee had difficulty properly addressing some of these issues. Overall the team concluded that implementation of the corrective action program was adequate with improvements warranted.

The team determined that site personnel were willing to raise safety issues and document them in the corrective action program. The team noted that workers at the site felt free to report problems to their management and the NRC, but were reluctant to take safety concerns to the Employee Concern Program. Additionally, the function and processes associated with the Employee Concern Program was not understood by a majority of the personnel interviewed.

Inspection Report# : 2010006 *(pdf)*

Last modified : October 14, 2011

# Diablo Canyon 2
# 3Q/2011 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), "Fire Protection," after the licensee failed to maintain the integrity of a fire barrier. On July 21, 2011, the inspectors identified that Fire Door B43-2, entrance to the Residual Heat Removal Pump Room 2-2, was inoperable. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to maintain the fire barrier in the rated configuration or establish prescribed compensatory actions. The door was held open due to Auxiliary Building ventilation flow balance problems. The ventilation problems had affected the fire door since January 12, 2011. The inspectors performed an extent of condition evaluation and identified eight additional fire doors impacted by the flow balance problems. The licensee took immediate action to restore the fire door to the rated condition and entered the problem into the corrective action programs as Notification 50416374.

The inspectors concluded that the failure of the licensee to maintain a fire door in the rated configuration was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the licensee had maintained an automatic full area water-based fire suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not take timely corrective actions to correct Auxiliary Building ventilation flow balance issues.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Surveillances on Fire Barriers**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), "Fire Protection," after the licensee failed to identify and correct the failure to perform required surveillance testing on fire-rated assemblies. On August 16, 2011, the inspectors identified that the licensee had not performed Equipment Control Guideline 18.7, "Fire Rated Assemblies," surveillance testing on Fire Door 329-2, entrance to the 125VDC Battery 2-1 Room, and Fire Door 332-2, entrance to the 125VDC Battery 2-2 Room, within the required frequency. The inspectors also identified that both fire doors were degraded and did not meet the surveillance acceptance criteria. The licensee implemented the required compensatory actions for both fire doors and entered this finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of the licensee to perform required surveillance tests on fire-rated assemblies was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the exposed fire areas did not contain any potential damage targets unique from those in the exposing fire area. This finding had a

crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately prioritize and perform an extent of condition review of previous problems associated with excluding fire barriers from the Equipment Control Guideline requirements.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Fire Hazard Evaluations**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), "Fire Protection," after Pacific Gas and Electric failed to implement the required compensatory actions described in Equipment Control Guideline 18.7, "Fire Rated Assemblies." On December 28, 2010, the licensee blocked open Fire Doors 175 and 182-2, entrances to the Unit 1 and 2 safety injection pump room to address auxiliary building ventilation flow balance problems. The supporting engineering evaluation failed to identify that the doors were rated fire barriers as described in the fire hazard analysis. If a fire had occurred, these blocked open doors would have allowed smoke and hot gases to pass from fire area AB-1 to impact equipment in adjacent fire areas 3-B-2 (Unit 1) and 3-D-2 (Unit 2). Equipment Control Guideline 18.7 required the licensee to either establish a continuous fire watch on at least one side of the inoperable fire doors or verify that the fire detection or automatic suppression system on at least one side of the fire doors was operable and establish an hourly fire watch. The licensee took corrective actions to establish the required fire watches and enter the finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of Pacific Gas and Electric to maintain the fire doors in the rated configuration as described in the Final Safety Analysis Report Update "Fire Hazard Analysis," was a performance deficiency. This finding was more than minor because the degraded fire barriers affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only affected the ability to reach and maintain cold shutdown conditions. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems associated with modification of the safety injection pump room fire doors such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation of New Security Modifications**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to adequately evaluate the impact of protected area boundary modifications. These modifications affected the ability of plant operators to transfer water from the raw water storage reservoirs to the auxiliary feedwater system using temporary hoses. Plant engineers authorized a series of security modifications which included the installation of physical intrusion barriers, including delay fences and razor wire between the raw water reservoirs and the auxiliary feedwater system. The licensing basis evaluation did not address raw water makeup to the auxiliary feedwater system using temporary hoses as described in Final Safety Analysis Report Update Section 6.5, "Auxiliary Feedwater System," and Section 3.7.6, "Seismic Evaluation to Demonstrate Compliance with the Hosgri Earthquake Requirements Utilizing a Dedicated Shutdown Flowpath." The licensee took immediate corrective actions to establish a route for the temporary hoses, including preplanned security compensatory measures, and entered this finding into the corrective action program as Notification 50410997.

The failure to adequately evaluate the impact of the security modifications on the plant licensing and design bases was a performance deficiency. This performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was
confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of Problem Identification and Resolution, associated with the Corrective Action Program component, because the

licensee failed to thoroughly evaluate the security modifications such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Preferred Offsite Power System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the preferred offsite power system design basis was correctly translated into electrical dynamic loading Calculations 357A-DC, “Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis,” Revision 12 and 359-DC, “Offsite Power Dynamic Analysis,” Revision 8. The licensee did not include the limiting load flow cases representing the largest total onsite demand for both units as required by the plant design basis. On July 7, 2010, the NRC clarified that the Diablo Canyon current licensing basis required the preferred offsite power system to have adequate capacity and capability to supply the most limiting loading requirements, including a dual unit trip. The licensee subsequently entered the condition into the corrective action program as Notification 50289590 and revised the station dynamic loading analysis to reflect the increased onsite power demand. The inspectors concluded that the failure to ensure that the dynamic loading analysis included all design basis requirements was a performance deficiency. This performance deficiency is more than minor because the finding was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because the inspectors were unable to conclude that the preferred offsite power system had not been inoperable for greater than the allowed Technical Specification outage time, a senior reactor analyst performed a bounding Phase 3 analysis. The Phase 3 analysis demonstrated that the subject finding was of very low safety significance (Green), because of the small increase of probability of a loss of offsite power that the finding represented. This finding had a crosscutting
aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the current licensing basis requirements to ensure that resolutions addressed causes and extent of conditions, as necessary. [P.1(c)]
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to maintain the integrity of Door 155 in the rated condition. On December 9, 2010, the inspectors identified that the fire door was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain Door 155 in a configuration that would provide at least a 1½-hour rated fire barrier. The inspectors previously identified that Door 155 was degraded as a fire barrier in 2009. The licensee entered the violation into the corrective action program as Notification 50367381 and took immediate corrective actions to restore the fire barrier to the rated condition and to implement weekly plant fire door walkdowns.

The inspectors concluded that the finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute and objective to prevent undesirable consequences due to fire. The inspectors determined that the finding was within the fire confinement category and that the fire barrier was moderately degraded. The inspectors concluded that the finding was of very low safety significance (Green) because there was a non-degraded automatic full area water-based suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not take effective corrective actions to following the previous occurrence of the violation [P.1(d)].

Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Transient Combustibles Procedure**
The inspectors identified a noncited violation of Diablo Canyon Unit 2 Facility Operating License Condition 2.C.(5), “Fire Protection,” after Pacific Gas and Electric failed to ensure procedures for controlling flammable and combustible materials adequately incorporated requirements of the fire hazard analysis. On October 18, 2010, the inspectors identified that transient combustible materials staged in the Unit 1 12 kilovolt switchgear room did not have an approved transient combustibles permit. The licensee stated that the combustibles permit procedure did not require a permit for the room while Unit 1 was shutdown. However, the plant fire hazards design basis described safe shutdown equipment in the room that would be needed to support a safe shutdown of the operating unit, specifically the Unit 2 startup bus located in the room. The inspectors determined that the licensee’s transient combustibles permit procedure was inadequate because the procedure did not require a permit for the Unit1 12 kilovolt switchgear room when Unit 2 was operating. The licensee entered the issue into the corrective action program as Notification 50366302 and performed an evaluation of the transient combustibles stored in the area.

The inspectors concluded that this finding was more than minor because it affected the Mitigating Systems Cornerstone external factors attribute objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. The inspectors determined that the finding was within the fire prevention and administrative controls category and represented a low degradation level due to the minimal impact on the effectiveness and reliability of the affected systems. The inspectors concluded that the finding was of very low safety significance (Green) based on a qualitative screening, the low degradation rating, and only equipment needed to reach and maintain cold shutdown conditions was affected. This finding had a crosscutting aspect in the area of human performance associated with the resources component because the licensee failed to ensure that the design documentation adequately identified the Unit 2 startup bus as equipment required for safe shutdown for Unit 2 [H.2 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Operability Determinations**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to adequately evaluate two nonconforming conditions for operability as required by Procedure OM7.ID12, “Operability Determination.” On October 15, 2010, the inspectors identified a less than adequate technical evaluation supporting Prompt Operability Assessment 50350918, “Unit 2 - Insulation in Bio-Wall Penetration.” Engineering personnel failed to adequately evaluate the extent of condition after technicians identified about 632 pounds of Temp-Mat and 60 pounds of Min-K fibrous insulation in the Unit 1 reactor coolant loop biological shield wall penetrations. This fibrous material could have potentially been transported and plugged the emergency core cooling containment sump screen. The licensee performed the prompt operability assessment for Unit 2, which was operating at the time. The inspectors concluded that the engineering personnel inappropriately applied the leak-before-break methodology to exclude about 87 percent of this material from the extent of condition review in the prompt operability assessment.
The second example involved Prompt Operability Assessment Notification 50355265, “RHR Sump Margin,” which was completed by the licensee on October 23, 2010. In this example, engineering personnel failed to identify and demonstrate that the specified safety function of the refueling water storage tank could be maintained as required by the plant operability procedure. The inspectors identified that the post accident flow path from the reactor cavity to the containment sump was blocked by a large shield plug. This blockage
reduced the amount of post accident inventory available at the containment sump at the time of transition from injection to recirculation mode of emergency core cooling operation. Engineering personnel failed to demonstrate that the safety function to ensure full sump submergence was maintained with the blocked flow path. Full submergence of the sump was used by the NRC as the basis for approval of Technical Specification 3.5.4, “Refueling Water Storage Tank,” inventory requirements. The licensee entered the violation into the corrective action program as Notification 50369117 and revised the prompt operability assessments using assumptions consistent with the current licensing bases.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone initial design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the decision making component because Pacific Gas and Electric did not use conservative assumptions in decisions to demonstrate component operability in either example [H.1(b)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Containment Recirculation Sump Design Control**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after Pacific Gas and Electric failed to ensure Calculation STA-255, "Minimum Required Refueling Water Storage Tank Level for GE Sumps," Revision 2, demonstrated adequate available refueling water storage tank inventory. On October 19, 2010, the inspectors identified that emergency core cooling post accident flow path from the reactor cavity to the containment sumps was blocked by a large steel plug on Unit 1. The accident analysis assumed this 35 square foot path was open to allow coolant from a pipe break inside the biological shield to communicate with containment sumps during the recirculation mode of emergency core cooling. The licensee credited the inventory from the reactor cavity when determining the minimum required refueling water storage tank volume in Calculation STA-255. Pacific Gas and Electric used Calculation STA-255 as the basis for determining the minimum required refueling water storage tank volume specified by Technical Specification 3.5.4, "Refueling Water Storage Tank." The inspectors identified that the recirculation flow path was also blocked on Unit 2. The inspectors concluded that the most significant contributor to the violation was inaccurate plant drawings used by plant engineers during the performance of Calculation STA-255. The licensee's corrective actions included completion of a prompt operability assessment justifying continued operation of Unit 2 and replacement of the shield plug with a movable platform on Unit 1 prior to plant restart.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone plant modification design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the performance deficiency involved a design deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of human performance associated with the resources component because Pacific Gas and Electric failed to use complete, accurate and up-to-date drawing for Calculation STA 255 [H.2(c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** G Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Emergency Diesel Generator Surveillance Testing**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," after Pacific Gas and Electric failed to develop and implement an adequate testing program for the emergency diesel generators that met design requirements and recommendations. Specifically, in December 2008, the inspectors identified that the diesel generator loading calculations were inadequate to demonstrate that the design bases were met. Pacific Gas and Electric updated the load calculations, but failed to make the necessary revisions to Surveillance Test Procedure STP M-9D1, "Diesel Generator Full Load Rejection Test." As a result, Pacific Gas and Electric failed to test several of the emergency diesel generators at the complete load as required by Regulatory Guide 1.108, Revision 1, which is part of the current licensing bases. The licensee entered this into the corrective action program as Notification 50368801, determined there was no loss of safety function for the affected components, and applied the provisions of Surveillance Requirement 3.0.3 for a missed surveillance test. The inspectors concluded the most significant contributor to the finding was less than adequate diesel generator loading evaluations to support corrective action from previous violations associated with the emergency diesel generator testing.

The inspectors concluded that the performance deficiency was more than minor because the finding affected the equipment control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance (Green) because it did not represent an actual loss of safety function of a single train for greater than its technical specification allowed outage time. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component because the licensee failed to perform an adequate evaluation of the nonconservative surveillance test such that the resolution addressed the fundamental basis for the surveillance [P.1 (c)].
Inspection Report# : 2010005 *(pdf)*

**Significance:** Dec 31, 2010
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Quality Verification Audits**
The inspectors identified a noncited violation of 10CFR Appendix B, Criterion XVIII, “Audits”, which required that a comprehensive system of planned and periodic audits be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program as well as follow up action, including re-audit of deficient areas, where indicated. Contrary to this requirement, Pacific Gas and Electric failed to ensure that a comprehensive system of planned and periodic audits were carried out to verify compliance with all aspects of the quality assurance program, determine the effectiveness of the program, and perform necessary follow up actions. Specifically, the 2008 Quality Verification audit of the corrective action program failed to adequately address an adverse trend in the problem evaluation process documented in NRC Inspection report 2008005, which identified eleven examples of an adverse trend in problem evaluation. The licensee entered this into their corrective action program as Notification 50365083 and determined
there was no loss of safety function for the affected components. The inspectors concluded the most significant contributor to the finding was a less than adequate evaluation of the corrective action trending program.

This finding was more than minor because it was associated with the equipment control attribute of the Mitigating Systems cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined the performance deficiency was of very low safety significance (Green) it was a deficiency confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee failed to coordinate and communicate the results from assessments to affected personnel, and track the corrective actions to address issues commensurate with their significance [P.3(c)].
Inspection Report# : 2010005 *(pdf)*

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# Barrier Integrity

**Significance:** Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Control Room Habitability System in the Design Configuration**
The inspectors identified a Green noncited violation of Technical Specification 5.5.19, “Control Room Envelope Habitability Program,” after the licensee failed to maintain the Unit 1 control room ventilation train in the design configuration. The inspectors identified that Unit 1 control room ventilation system was in a degraded/non-conforming condition on August 31, 2011. The inspectors observed airflow bypassing the control room inlet header through disconnected ductwork. Technical Specification 5.5.19 required the licensee to maintain the habitability system in the most limited configuration used during the tracer gas in-leakage test. The disconnected ductwork was a more limiting condition than the tested configuration. The licensee took corrective action to declare the control room envelope inoperable and entered the finding into the corrective action program as Notification 50425114.

The inspectors determined that the failure of the licensee to maintain the control room habitability system in the design configuration was a performance deficiency. This finding was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a crosscutting aspect in the area of human performance associated with work control in that the licensee failed to appropriately plan work activities consistent with nuclear safety.
Inspection Report# : 2011004 *(pdf)*

**Significance:** Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow a Procedural Requirement for Reactivity Manipulation**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Procedures,” after operations personnel conducted a reactivity manipulation during shift turnover. Procedure OP1.ID3, “Reactivity Management Program,” required plant operators to suspend reactivity manipulations during shift turnover. On March 27, 2011, plant operators conducted a continuous dilution during shift turnover. The licensee entered this condition into the corrective action program as Notification 50407054.

The inspectors concluded that the failure of operations personnel to follow Procedure OP1.ID3 was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the procedure adherence area of the human performance attribute of the barrier integrity cornerstone and affected the objective to provide reasonable assurance that design barriers will protect the public from radionuclide releases. The inspectors concluded that the finding was of very low safety significance (Green) because only the fuel barrier was affected by the performance deficiency. The finding has a crosscutting aspect in the area of human performance, associated with work practices component, because the licensee failed to define and effectively communicate expectations regarding procedural compliance.
Inspection Report# : 2011004 *(pdf)*

**Significance:** Jun 26, 2011
Identified By: NRC
Item Type: FIN Finding
**Inadequate Review of Severe Accident Management Guidelines**
The inspectors identified a finding after Pacific Gas and Electric failed to periodically review and update the severe accident management guidelines. Procedure OM10.ID5, “Severe Accident Management,” required the licensee to review and update the severe accident management guidelines biennially to ensure that any changes in plant design or procedures, experience in severe accident management requalification training, and any changes in industry understanding of severe accidents were incorporated into the severe accident management guidelines. As a result of the licensee’s failure to implement the periodic review, the severe accident management guidelines did not incorporate the latest owners’ group guidance or recent plant design and hardware changes. The licensee took corrective actions to implement the biennial reviews and entered this finding into the corrective action program as Notification 50399554.

Pacific Gas and Electric’s failure to follow procedural requirements for periodic review of the severe accident management guidelines was a performance deficiency. The finding was more than minor because if left uncorrected, the failure to review and update the severe accident management guidelines has the potential to lead to a more significant safety concern. This finding affected the barrier integrity cornerstone because the severe accident management guidelines are procedures that would be used to maintain the functionality of the containment should a severe accident occur. The inspectors concluded that the finding was of very low safety significance because it did not represent a degradation of the radiological, smoke, or toxic atmosphere barrier function; or represent an actual open pathway in the physical integrity of the reactor containment; or involve the function of the containment hydrogen igniters. The finding did not have any crosscutting aspects because the performance deficiency occurred more than three years ago and is not indicative of current licensee performance in that the licensee has improved the design review process since the performance deficiency occurred.

Inspection Report# : 2011003 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Design Basis of Containment Fan Cooler Unit Cooling Coil Casings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, prior to December 15, 2010, the licensee failed to assure that the design basis function of the containment fan cooler unit casings was translated into specifications, drawings, procedures, and instructions. The licensee has entered this violation into the corrective action program as Notification 50384801.

The inspectors determined that the failure to establish measures to assure that the design basis function of the containment fan cooler unit cooling coil casings was translated into specifications, drawings, procedures, and instructions was a performance deficiency. The finding was more than minor because it adversely affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 4, "Initial Screening and Characterization of Findings," the finding was determined to be of very low safety significance (Green) because it did not represent a degradation of the barrier function of the control room against a smoke or toxic barrier, an open pathway in the physical integrity of reactor containment, or an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined that this finding has a crosscutting aspect in the area of human performance because the licensee failed to ensure that personnel, equipment, procedures and other resources were available to assure nuclear safety by maintaining complete, accurate and up-to-date design documentation. [H.2(c)]
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Auxiliary Building Ventilation System Control Panel Modification**
The inspectors reviewed a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the design basis requirements for single failure criteria were correctly translated into auxiliary building ventilation system controls modifications. On January 10, 2011, a single failure of a Unit 2 auxiliary building ventilation Train “A” damper resulted in the loss of system safety function for both trains. The loss of safety function occurred because of a logic error in the programmable logic controllers. The licensee programmed and installed the logic controllers in November 2010 for Unit 1 and in November 2009 for Unit 2. The inspectors identified that the engineering department performed a less than adequate review to identify the single point vulnerability during the modification review process. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50370698, replaced the failed damper, and implemented compensatory actions to mitigate the design deficiency. The licensee plans to implement corrective actions to program the logic controller program consistent with the design basis requirements.

The inspectors concluded that the failure to ensure that the modification met design basis requirements was a performance deficiency. This performance deficiency is more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers and radiological barriers, including the Auxiliary Building, protect the public from radionuclide releases caused by accidents or events. The inspectors determined that the finding had very low safety significance because the finding only represents degradation to the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance associated with work practices because the licensee did not ensure human error prevention techniques, such as self and peer checking, were effectively used in the preparation of the modification. [H.4(a)]
Inspection Report# : 2011002 *(pdf)*

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## Emergency Preparedness

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Ensure Emergency Response Organization Qualifications**
A noncited violation of 10 CFR 50.47(b)(10) was identified for the licensee's failure to ensure a range of protective actions is available for emergency workers during emergencies. Specifically, an operator filled an on-shift emergency response organization watch position with expired self-contained breathing apparatus respiratory protection qualifications. The licensee has entered this issue into the corrective action program as Notification 50420127.

The failure to ensure that an emergency response organization on-shift watch stander was respiratory protection qualified is a performance deficiency. This finding is greater than minor because it affects the emergency response organization readiness attribute of the emergency preparedness cornerstone to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it was not a loss of a planning standard function. The finding had a human performance crosscutting aspect of conservative assumptions under the decision making component because the licensee did not ensure that personnel filling the minimum shift staffing emergency response organization positions were qualified to take the watch.
Inspection Report# : 2011004 *(pdf)*

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## Occupational Radiation Safety

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedures for Testing HEPA Ventilation Units**
The inspectors identified a noncited violation of Technical Specification 5.4.1(a) for the failure to follow procedures for testing and using the high-efficiency particulate air ventilation units used to prevent personal contamination. Licensee immediate actions included removing all high-efficiency particulate air ventilation units installed for the Unit 2 outage and testing all high-efficiency particulate air ventilation units as required by procedure. This matter was placed in the licensee's corrective action program as Notifications 50399479, 50399560, and 50399682.

This failure to follow procedures was a performance deficiency. The finding was more than minor because it was associated with the program and process attribute of the occupational radiation safety cornerstone. The finding affected the objective to ensure adequate protection of the worker's health and safety from exposure to unintended radiation from radioactive material during routine civilian nuclear reactor operation. Using the Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined the finding was of very low safety significance because (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This finding was determined to have a crosscutting aspect in the area of human performance, associated with work practices, because the licensee did not effectively communicate expectations regarding procedural compliance and the personnel following the procedures [H.4(b)].
Inspection Report# : 2011003 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : January 04, 2012

# Diablo Canyon 2
# 4Q/2011 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), “Fire Protection,” after the licensee failed to maintain the integrity of a fire barrier. On July 21, 2011, the inspectors identified that Fire Door B43-2, entrance to the Residual Heat Removal Pump Room 2-2, was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain the fire barrier in the rated configuration or establish prescribed compensatory actions. The door was held open due to Auxiliary Building ventilation flow balance problems. The ventilation problems had affected the fire door since January 12, 2011. The inspectors performed an extent of condition evaluation and identified eight additional fire doors impacted by the flow balance problems. The licensee took immediate action to restore the fire door to the rated condition and entered the problem into the corrective action programs as Notification 50416374.

The inspectors concluded that the failure of the licensee to maintain a fire door in the rated configuration was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the licensee had maintained an automatic full area water-based fire suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not take timely corrective actions to correct Auxiliary Building ventilation flow balance issues.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Surveillances on Fire Barriers**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), “Fire Protection,” after the licensee failed to identify and correct the failure to perform required surveillance testing on fire-rated assemblies. On August 16, 2011, the inspectors identified that the licensee had not performed Equipment Control Guideline 18.7, “Fire Rated Assemblies,” surveillance testing on Fire Door 329-2, entrance to the 125VDC Battery 2-1 Room, and Fire Door 332-2, entrance to the 125VDC Battery 2-2 Room, within the required frequency. The inspectors also identified that both fire doors were degraded and did not meet the surveillance acceptance criteria. The licensee implemented the required compensatory actions for both fire doors and entered this finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of the licensee to perform required surveillance tests on fire-rated assemblies was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the exposed fire areas did not contain any potential damage targets unique from those in the exposing fire area. This finding had a

crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately prioritize and perform an extent of condition review of previous problems associated with excluding fire barriers from the Equipment Control Guideline requirements.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Fire Hazard Evaluations**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), “Fire Protection,” after Pacific Gas and Electric failed to implement the required compensatory actions described in Equipment Control Guideline 18.7, “Fire Rated Assemblies.” On December 28, 2010, the licensee blocked open Fire Doors 175 and 182-2, entrances to the Unit 1 and 2 safety injection pump room to address auxiliary building ventilation flow balance problems. The supporting engineering evaluation failed to identify that the doors were rated fire barriers as described in the fire hazard analysis. If a fire had occurred, these blocked open doors would have allowed smoke and hot gases to pass from fire area AB-1 to impact equipment in adjacent fire areas 3-B-2 (Unit 1) and 3-D-2 (Unit 2). Equipment Control Guideline 18.7 required the licensee to either establish a continuous fire watch on at least one side of the inoperable fire doors or verify that the fire detection or automatic suppression system on at least one side of the fire doors was operable and establish an hourly fire watch. The licensee took corrective actions to establish the required fire watches and enter the finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of Pacific Gas and Electric to maintain the fire doors in the rated configuration as described in the Final Safety Analysis Report Update “Fire Hazard Analysis,” was a performance deficiency. This finding was more than minor because the degraded fire barriers affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only affected the ability to reach and maintain cold shutdown conditions. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems associated with modification of the safety injection pump room fire doors such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation of New Security Modifications**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to adequately evaluate the impact of protected area boundary modifications. These modifications affected the ability of plant operators to transfer water from the raw water storage reservoirs to the auxiliary feedwater system using temporary hoses. Plant engineers authorized a series of security modifications which included the installation of physical intrusion barriers, including delay fences and razor wire between the raw water reservoirs and the auxiliary feedwater system. The licensing basis evaluation did not address raw water makeup to the auxiliary feedwater system using temporary hoses as described in Final Safety Analysis Report Update Section 6.5, “Auxiliary Feedwater System,” and Section 3.7.6, “Seismic Evaluation to Demonstrate Compliance with the Hosgri Earthquake Requirements Utilizing a Dedicated Shutdown Flowpath.” The licensee took immediate corrective actions to establish a route for the temporary hoses, including preplanned security compensatory measures, and entered this finding into the corrective action program as Notification 50410997.

The failure to adequately evaluate the impact of the security modifications on the plant licensing and design bases was a performance deficiency. This performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was
confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of Problem Identification and Resolution, associated with the Corrective Action Program component, because the

licensee failed to thoroughly evaluate the security modifications such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Preferred Offsite Power System**
The inspectors identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the preferred offsite power system design basis was correctly translated into electrical dynamic loading Calculations 357A-DC, “Units 1 and 2 Load Flow, Short Circuit and Motor Starting Analysis,” Revision 12 and 359-DC, “Offsite Power Dynamic Analysis,” Revision 8. The licensee did not include the limiting load flow cases representing the largest total onsite demand for both units as required by the plant design basis. On July 7, 2010, the NRC clarified that the Diablo Canyon current licensing basis required the preferred offsite power system to have adequate capacity and capability to supply the most limiting loading requirements, including a dual unit trip. The licensee subsequently entered the condition into the corrective action program as Notification 50289590 and revised the station dynamic loading analysis to reflect the increased onsite power demand. The inspectors concluded that the failure to ensure that the dynamic loading analysis included all design basis requirements was a performance deficiency. This performance deficiency is more than minor because the finding was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because the inspectors were unable to conclude that the preferred offsite power system had not been inoperable for greater than the allowed Technical Specification outage time, a senior reactor analyst performed a bounding Phase 3 analysis. The Phase 3 analysis demonstrated that the subject finding was of very low safety significance (Green), because of the small increase of probability of a loss of offsite power that the finding represented. This finding had a crosscutting
aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate the current licensing basis requirements to ensure that resolutions addressed causes and extent of conditions, as necessary. [P.1(c)]
Inspection Report# : 2011002 *(pdf)*

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# Barrier Integrity

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Control Room Habitability System in the Design Configuration**
The inspectors identified a Green noncited violation of Technical Specification 5.5.19, “Control Room Envelope Habitability Program,” after the licensee failed to maintain the Unit 1 control room ventilation train in the design configuration. The inspectors identified that Unit 1 control room ventilation system was in a degraded/non-conforming condition on August 31, 2011. The inspectors observed airflow bypassing the control room inlet header through disconnected ductwork. Technical Specification 5.5.19 required the licensee to maintain the habitability system in the most limited configuration used during the tracer gas in-leakage test. The disconnected ductwork was a more limiting condition than the tested configuration. The licensee took corrective action to declare the control room envelope inoperable and entered the finding into the corrective action program as Notification 50425114.

The inspectors determined that the failure of the licensee to maintain the control room habitability system in the design configuration was a performance deficiency. This finding was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a crosscutting aspect in the area of human performance associated with work control in that the licensee failed to

appropriately plan work activities consistent with nuclear safety.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow a Procedural Requirement for Reactivity Manipulation**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," after operations personnel conducted a reactivity manipulation during shift turnover. Procedure OP1.ID3, "Reactivity Management Program," required plant operators to suspend reactivity manipulations during shift turnover. On March 27, 2011, plant operators conducted a continuous dilution during shift turnover. The licensee entered this condition into the corrective action program as Notification 50407054.

The inspectors concluded that the failure of operations personnel to follow Procedure OP1.ID3 was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the procedure adherence area of the human performance attribute of the barrier integrity cornerstone and affected the objective to provide reasonable assurance that design barriers will protect the public from radionuclide releases. The inspectors concluded that the finding was of very low safety significance (Green) because only the fuel barrier was affected by the performance deficiency. The finding has a crosscutting aspect in the area of human performance, associated with work practices component, because the licensee failed to define and effectively communicate expectations regarding procedural compliance.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: FIN Finding
**Inadequate Review of Severe Accident Management Guidelines**
The inspectors identified a finding after Pacific Gas and Electric failed to periodically review and update the severe accident management guidelines. Procedure OM10.ID5, "Severe Accident Management," required the licensee to review and update the severe accident management guidelines biennially to ensure that any changes in plant design or procedures, experience in severe accident management requalification training, and any changes in industry understanding of severe accidents were incorporated into the severe accident management guidelines. As a result of the licensee's failure to implement the periodic review, the severe accident management guidelines did not incorporate the latest owners' group guidance or recent plant design and hardware changes. The licensee took corrective actions to implement the biennial reviews and entered this finding into the corrective action program as Notification 50399554.

Pacific Gas and Electric's failure to follow procedural requirements for periodic review of the severe accident management guidelines was a performance deficiency. The finding was more than minor because if left uncorrected, the failure to review and update the severe accident management guidelines has the potential to lead to a more significant safety concern. This finding affected the barrier integrity cornerstone because the severe accident management guidelines are procedures that would be used to maintain the functionality of the containment should a severe accident occur. The inspectors concluded that the finding was of very low safety significance because it did not represent a degradation of the radiological, smoke, or toxic atmosphere barrier function; or represent an actual open pathway in the physical integrity of the reactor containment; or involve the function of the containment hydrogen igniters. The finding did not have any crosscutting aspects because the performance deficiency occurred more than three years ago and is not indicative of current licensee performance in that the licensee has improved the design review process since the performance deficiency occurred.
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Design Basis of Containment Fan Cooler Unit Cooling Coil Casings**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which

states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Specifically, prior to December 15, 2010, the licensee failed to assure that the design basis function of the containment fan cooler unit casings was translated into specifications, drawings, procedures, and instructions. The licensee has entered this violation into the corrective action program as Notification 50384801.

The inspectors determined that the failure to establish measures to assure that the design basis function of the containment fan cooler unit cooling coil casings was translated into specifications, drawings, procedures, and instructions was a performance deficiency. The finding was more than minor because it adversely affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 4, "Initial Screening and Characterization of Findings," the finding was determined to be of very low safety significance (Green) because it did not represent a degradation of the barrier function of the control room against a smoke or toxic barrier, an open pathway in the physical integrity of reactor containment, or an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined that this finding has a crosscutting aspect in the area of human performance because the licensee failed to ensure that personnel, equipment, procedures and other resources were available to assure nuclear safety by maintaining complete, accurate and up-to-date design documentation. [H.2(c)]
Inspection Report# : 2011002 *(pdf)*

**Significance:** G Mar 27, 2011
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Design Control for the Auxiliary Building Ventilation System Control Panel Modification**
The inspectors reviewed a self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after Pacific Gas and Electric failed to ensure that the design basis requirements for single failure criteria were correctly translated into auxiliary building ventilation system controls modifications. On January 10, 2011, a single failure of a Unit 2 auxiliary building ventilation Train “A” damper resulted in the loss of system safety function for both trains. The loss of safety function occurred because of a logic error in the programmable logic controllers. The licensee programmed and installed the logic controllers in November 2010 for Unit 1 and in November 2009 for Unit 2. The inspectors identified that the engineering department performed a less than adequate review to identify the single point vulnerability during the modification review process. Pacific Gas and Electric entered this issue into the corrective action program as Notification 50370698, replaced the failed damper, and implemented compensatory actions to mitigate the design deficiency. The licensee plans to implement corrective actions to program the logic controller program consistent with the design basis requirements.

The inspectors concluded that the failure to ensure that the modification met design basis requirements was a performance deficiency. This performance deficiency is more than minor because it was associated with the design control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance that physical design barriers and radiological barriers, including the Auxiliary Building, protect the public from radionuclide releases caused by accidents or events. The inspectors determined that the finding had very low safety significance because the finding only represents degradation to the radiological barrier function provided for the auxiliary building. This finding had a crosscutting aspect in the area of human performance associated with work practices because the licensee did not ensure human error prevention techniques, such as self and peer checking, were effectively used in the preparation of the modification. [H.4(a)]
Inspection Report# : 2011002 *(pdf)*

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# Emergency Preparedness

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Ensure Emergency Response Organization Qualifications**

A noncited violation of 10 CFR 50.47(b)(10) was identified for the licensee's failure to ensure a range of protective actions is available for emergency workers during emergencies. Specifically, an operator filled an on-shift emergency response organization watch position with expired self-contained breathing apparatus respiratory protection qualifications. The licensee has entered this issue into the corrective action program as Notification 50420127.

The failure to ensure that an emergency response organization on-shift watch stander was respiratory protection qualified is a performance deficiency. This finding is greater than minor because it affects the emergency response organization readiness attribute of the emergency preparedness cornerstone to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it was not a loss of a planning standard function. The finding had a human performance crosscutting aspect of conservative assumptions under the decision making component because the licensee did not ensure that personnel filling the minimum shift staffing emergency response organization positions were qualified to take the watch.
Inspection Report# : 2011004 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedures for Testing HEPA Ventilation Units**
The inspectors identified a noncited violation of Technical Specification 5.4.1(a) for the failure to follow procedures for testing and using the high-efficiency particulate air ventilation units used to prevent personal contamination. Licensee immediate actions included removing all high-efficiency particulate air ventilation units installed for the Unit 2 outage and testing all high-efficiency particulate air ventilation units as required by procedure. This matter was placed in the licensee's corrective action program as Notifications 50399479, 50399560, and 50399682.

This failure to follow procedures was a performance deficiency. The finding was more than minor because it was associated with the program and process attribute of the occupational radiation safety cornerstone. The finding affected the objective to ensure adequate protection of the worker's health and safety from exposure to unintended radiation from radioactive material during routine civilian nuclear reactor operation. Using the Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined the finding was of very low safety significance because (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This finding was determined to have a crosscutting aspect in the area of human performance, associated with work practices, because the licensee did not effectively communicate expectations regarding procedural compliance and the personnel following the procedures [H.4(b)].
Inspection Report# : 2011003 *(pdf)*

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# Public Radiation Safety

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# Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

Last modified : March 02, 2012

# Diablo Canyon 2
# 1Q/2012 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate OperabilityDetermination**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after operations personnel declared diesel generator 2-3 operable after failing to meet all surveillance test acceptance criterion. On December 22, 2011, diesel generator 2-3 did not meet frequency acceptance criteria during technical specification surveillance testing. Plant operators declared the diesel operable after applying an engineering evaluation. The inspectors identified that the evaluation was not appropriate to the conditions of the failed test. The licensee’s corrective actions included corrective maintenance, re-performance of the surveillance test, and entering the condition into the corrective action program as Notifications 50449027 and 50449504.

The failure of operations personnel to recognize that diesel generator surveillance results indicated that the system was not fully operable was a performance deficiency. This finding was more than minor because the licensee’s engineering evaluation created a reasonable doubt that the system was operable, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues.” The inspectors concluded that the finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in the loss of operability or functionality of a single train for greater than the technical specification outage time, did not represent an actual loss of safety function, and was not potentially risk significant due to a seismic, flooding, or severe weather event. The most significant contributor to this performance deficiency was that operators did not review and understand the diesel generator surveillance results sufficiently to recognize that the condition did not match the previously-evaluated condition that was used to conclude the diesel generator remained operable. Therefore, this finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component [P.1(c)].
Inspection Report# : 2012002 *(pdf)*

**Significance:** G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform an Operability Determination for New Seismic Information**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to evaluate the affect of new seismic information on the operability of plant structures, systems and components. On January 7, 2011, the licensee completed and submitted to the NRC a report to the detailing the results of a deterministic reevaluation of the local seismology. This report concluded that an earthquake on three local faults could produce greater vibratory ground motion than bound by the safe shutdown earthquake as described in the Final Safety Evaluation Report Update. Quality Procedure OM7.ID12, “Operability Determinations,” required plant operators to assess the impact of nonconforming conditions for the affect on plant structures, systems and components without delay. On June 22, 2011, the licensee entered the condition into the corrective action program as Notification 50410266 and completed an operability determination on June 24, 2011.

The inspectors determined that the licensee’s failure to evaluate the new seismic information against the plant design and licensing bases was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The senior reactor analyst evaluated the significance of the finding using a Phase 3 analysis because the inspectors were unable to confirm that the operability of plant systems was not impacted. The senior reactor analyst concluded that the finding was of very low risk significance (Green) because no significant change in overall core damage frequency resulted from the new seismic hazards. This finding had a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee used non-conservative assumptions in deciding not to evaluate the new seismic information against the current plant design and licensing bases (H.1.b).

Inspection Report# : 2011005 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), “Fire Protection,” after the licensee failed to maintain the integrity of a fire barrier. On July 21, 2011, the inspectors identified that Fire Door B43-2, entrance to the Residual Heat Removal Pump Room 2-2, was inoperable. Equipment Control Guideline 18.7, “Fire Rated Assemblies,” required the licensee to maintain the fire barrier in the rated configuration or establish prescribed compensatory actions. The door was held open due to Auxiliary Building ventilation flow balance problems. The ventilation problems had affected the fire door since January 12, 2011. The inspectors performed an extent of condition evaluation and identified eight additional fire doors impacted by the flow balance problems. The licensee took immediate action to restore the fire door to the rated condition and entered the problem into the corrective action programs as Notification 50416374.

The inspectors concluded that the failure of the licensee to maintain a fire door in the rated configuration was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the licensee had maintained an automatic full area water-based fire suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not take timely corrective actions to correct Auxiliary Building ventilation flow balance issues.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Surveillances on Fire Barriers**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), “Fire Protection,” after the licensee failed to identify and correct the failure to perform required surveillance testing on fire-rated assemblies. On August 16, 2011, the inspectors identified that the licensee had not performed Equipment Control Guideline 18.7, “Fire Rated Assemblies,” surveillance testing on Fire Door 329-2, entrance to the 125VDC Battery 2-1 Room, and Fire Door 332-2, entrance to the 125VDC Battery 2-2 Room, within the required frequency. The inspectors also identified that both fire doors were degraded and did not meet the surveillance acceptance criteria. The licensee implemented the required compensatory actions for both fire doors and entered this finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of the licensee to perform required surveillance tests on fire-rated assemblies was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the exposed fire areas did not contain any potential damage targets unique from those in the exposing fire area. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program

component because the licensee did not adequately prioritize and perform an extent of condition review of previous problems associated with excluding fire barriers from the Equipment Control Guideline requirements.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Fire Hazard Evaluations**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (5), "Fire Protection," after Pacific Gas and Electric failed to implement the required compensatory actions described in Equipment Control Guideline 18.7, "Fire Rated Assemblies." On December 28, 2010, the licensee blocked open Fire Doors 175 and 182-2, entrances to the Unit 1 and 2 safety injection pump room to address auxiliary building ventilation flow balance problems. The supporting engineering evaluation failed to identify that the doors were rated fire barriers as described in the fire hazard analysis. If a fire had occurred, these blocked open doors would have allowed smoke and hot gases to pass from fire area AB-1 to impact equipment in adjacent fire areas 3-B-2 (Unit 1) and 3-D-2 (Unit 2). Equipment Control Guideline 18.7 required the licensee to either establish a continuous fire watch on at least one side of the inoperable fire doors or verify that the fire detection or automatic suppression system on at least one side of the fire doors was operable and establish an hourly fire watch. The licensee took corrective actions to establish the required fire watches and enter the finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of Pacific Gas and Electric to maintain the fire doors in the rated configuration as described in the Final Safety Analysis Report Update "Fire Hazard Analysis," was a performance deficiency. This finding was more than minor because the degraded fire barriers affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only affected the ability to reach and maintain cold shutdown conditions. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not thoroughly evaluate problems associated with modification of the safety injection pump room fire doors such that the resolutions addressed causes and extent of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less Than Adequate Evaluation of New Security Modifications**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to adequately evaluate the impact of protected area boundary modifications. These modifications affected the ability of plant operators to transfer water from the raw water storage reservoirs to the auxiliary feedwater system using temporary hoses. Plant engineers authorized a series of security modifications which included the installation of physical intrusion barriers, including delay fences and razor wire between the raw water reservoirs and the auxiliary feedwater system. The licensing basis evaluation did not address raw water makeup to the auxiliary feedwater system using temporary hoses as described in Final Safety Analysis Report Update Section 6.5, "Auxiliary Feedwater System," and Section 3.7.6, "Seismic Evaluation to Demonstrate Compliance with the Hosgri Earthquake Requirements Utilizing a Dedicated Shutdown Flowpath." The licensee took immediate corrective actions to establish a route for the temporary hoses, including preplanned security compensatory measures, and entered this finding into the corrective action program as Notification 50410997.

The failure to adequately evaluate the impact of the security modifications on the plant licensing and design bases was a performance deficiency. This performance deficiency was more than minor because the finding affected the Mitigating Systems Cornerstone design control attribute and objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded that the finding was of very low safety significance (Green) because the finding was confirmed not to result in the loss of operability or functionality. This finding had a crosscutting aspect in the area of Problem Identification and Resolution, associated with the Corrective Action Program component, because the licensee failed to thoroughly evaluate the security modifications such that the resolutions addressed causes and extent

of conditions, as necessary [P.1(c)].
Inspection Report# : 2011003 *(pdf)*

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# Barrier Integrity

**Significance: TBD** Mar 23, 2012
Identified By: NRC
Item Type: AV Apparent Violation
**Incomplete and Inaccurate Information Provided to the NRC in Response to Generic Letter 2003-01, "Control Room Habitability"**
The inspectors identified a Green finding and Severity Level III violation of 10 CFR 50.9, “Completeness and Accuracy of Information,” after Pacific Gas and Electric failed to submitted complete and accurate information in response to Generic Letter 2003-01, “Control Room Habitability.” Generic Letter 2003-01 requested that the licensee submit information demonstrating that the control room habitability system was in compliance with the current licensing and design bases. The licensee was specifically requested to verify that the most limiting unfiltered in-leakage into the control room envelope was no more than the value assumed in the design basis radiological analyses for control room habitability. On April 22, 2005, the licensee reported to the NRC that testing performed in the most limiting configuration for operator dose demonstrated that there was no unfiltered in-leakage into the control room envelope. This was material because the NRC used this information to close out Generic Letter 2003-01. In September 2011, the inspectors identified that the control room test results were greater than the value assumed in the design basis radiological analysis and that the licensee’s testing was not performed in the most limiting configuration for operator dose. Using the actual control room in-leakage rates, the inspectors concluded that the resultant operator dose could have exceeded the limit established by current licensing and design bases during an accident.

The inspectors concluded that the failure of Pacific Gas and Electric to provide complete and accurate information in response to Generic Letter 2003-01 was a performance deficiency. The finding was more than minor because the information was material to the NRC’s decision making processes. The inspectors screened the issue through the Reactor Oversight Process because the finding included a performance deficiency that was reasonably within the licensee’s ability to control. The inspectors concluded that the finding was of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The inspectors also screened the issue through the traditional enforcement process because the violation impacted the regulatory process. The inspectors concluded that the violation was a Severity Level III because had the licensee provided complete and accurate information in their letter dated April 22, 2005, the NRC would have likely reconsidered a regulatory position or undertaken a substantial further inquiry. The inspectors did not identify a cross-cutting aspect because the performance deficiency was not reflective of present performance.
Inspection Report# : 2012002 *(pdf)*

**Significance:** G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Evaluation of a Nonconforming Control Room Habitability Train**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after operations personnel failed to adequately evaluate the operability and extent of condition of a nonconforming control room habitability train. Beginning on August 30, 2011, the inspectors identified several nonconforming conditions associated with the habitability system, including disconnected ductwork, two 12 inch diameter openings in the envelope boundary, and less than adequate control room envelope pressurization and tracer gas surveillance tests. On November 7, 2011, the licensee re-performed the tracer gas test and observed gross unfiltered in-leakage into the control room envelope. Plant operators declared the habitability system inoperable. The licensee restored system operability after implementing a series of compensatory measures. The licensee entered the finding into the corrective action program as Notification 50425114 and plans to restore the system to the current licensing basis condition.

The inspectors concluded that the failure of plant operators to adequately evaluate the operability and extent of a nonconforming condition was a performance deficiency. This finding was more than minor because the licensee’s operability evaluation created a reasonable doubt that the system was capable of performing the specified safety function, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues." The inspectors concluded that the finding was of very low safety significance because only the radiological barrier function of the control room was affected. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the degraded control room ventilation train for operability and extent of condition [P.1(c)].

Inspection Report# : 2011005 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Control Room Habitability System in the Design Configuration**
The inspectors identified a Green noncited violation of Technical Specification 5.5.19, "Control Room Envelope Habitability Program," after the licensee failed to maintain the Unit 1 control room ventilation train in the design configuration. The inspectors identified that Unit 1 control room ventilation system was in a degraded/non-conforming condition on August 31, 2011. The inspectors observed airflow bypassing the control room inlet header through disconnected ductwork. Technical Specification 5.5.19 required the licensee to maintain the habitability system in the most limited configuration used during the tracer gas in-leakage test. The disconnected ductwork was a more limiting condition than the tested configuration. The licensee took corrective action to declare the control room envelope inoperable and entered the finding into the corrective action program as Notification 50425114.

The inspectors determined that the failure of the licensee to maintain the control room habitability system in the design configuration was a performance deficiency. This finding was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a crosscutting aspect in the area of human performance associated with work control in that the licensee failed to appropriately plan work activities consistent with nuclear safety.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow a Procedural Requirement for Reactivity Manipulation**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, "Procedures," after operations personnel conducted a reactivity manipulation during shift turnover. Procedure OP1.ID3, "Reactivity Management Program," required plant operators to suspend reactivity manipulations during shift turnover. On March 27, 2011, plant operators conducted a continuous dilution during shift turnover. The licensee entered this condition into the corrective action program as Notification 50407054.

The inspectors concluded that the failure of operations personnel to follow Procedure OP1.ID3 was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the procedure adherence area of the human performance attribute of the barrier integrity cornerstone and affected the objective to provide reasonable assurance that design barriers will protect the public from radionuclide releases. The inspectors concluded that the finding was of very low safety significance (Green) because only the fuel barrier was affected by the performance deficiency. The finding has a crosscutting aspect in the area of human performance, associated with work practices component, because the licensee failed to define and effectively communicate expectations regarding procedural compliance.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Jun 26, 2011
Identified By: NRC

Item Type: FIN Finding
**Inadequate Review of Severe Accident Management Guidelines**
The inspectors identified a finding after Pacific Gas and Electric failed to periodically review and update the severe accident management guidelines. Procedure OM10.ID5, "Severe Accident Management," required the licensee to review and update the severe accident management guidelines biennially to ensure that any changes in plant design or procedures, experience in severe accident management requalification training, and any changes in industry understanding of severe accidents were incorporated into the severe accident management guidelines. As a result of the licensee's failure to implement the periodic review, the severe accident management guidelines did not incorporate the latest owners' group guidance or recent plant design and hardware changes. The licensee took corrective actions to implement the biennial reviews and entered this finding into the corrective action program as Notification 50399554.

Pacific Gas and Electric's failure to follow procedural requirements for periodic review of the severe accident management guidelines was a performance deficiency. The finding was more than minor because if left uncorrected, the failure to review and update the severe accident management guidelines has the potential to lead to a more significant safety concern. This finding affected the barrier integrity cornerstone because the severe accident management guidelines are procedures that would be used to maintain the functionality of the containment should a severe accident occur. The inspectors concluded that the finding was of very low safety significance because it did not represent a degradation of the radiological, smoke, or toxic atmosphere barrier function; or represent an actual open pathway in the physical integrity of the reactor containment; or involve the function of the containment hydrogen igniters. The finding did not have any crosscutting aspects because the performance deficiency occurred more than three years ago and is not indicative of current licensee performance in that the licensee has improved the design review process since the performance deficiency occurred.
Inspection Report# : 2011003 *(pdf)*

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# Emergency Preparedness

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Ensure Emergency Response Organization Qualifications**
A noncited violation of 10 CFR 50.47(b)(10) was identified for the licensee's failure to ensure a range of protective actions is available for emergency workers during emergencies. Specifically, an operator filled an on-shift emergency response organization watch position with expired self-contained breathing apparatus respiratory protection qualifications. The licensee has entered this issue into the corrective action program as Notification 50420127.

The failure to ensure that an emergency response organization on-shift watch stander was respiratory protection qualified is a performance deficiency. This finding is greater than minor because it affects the emergency response organization readiness attribute of the emergency preparedness cornerstone to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it was not a loss of a planning standard function. The finding had a human performance crosscutting aspect of conservative assumptions under the decision making component because the licensee did not ensure that personnel filling the minimum shift staffing emergency response organization positions were qualified to take the watch.
Inspection Report# : 2011004 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Jun 26, 2011
Identified By: NRC
Item Type: NCV NonCited Violation

**Failure to Follow Procedures for Testing HEPA Ventilation Units**
The inspectors identified a noncited violation of Technical Specification 5.4.1(a) for the failure to follow procedures for testing and using the high-efficiency particulate air ventilation units used to prevent personal contamination. Licensee immediate actions included removing all high-efficiency particulate air ventilation units installed for the Unit 2 outage and testing all high-efficiency particulate air ventilation units as required by procedure. This matter was placed in the licensee's corrective action program as Notifications 50399479, 50399560, and 50399682.

This failure to follow procedures was a performance deficiency. The finding was more than minor because it was associated with the program and process attribute of the occupational radiation safety cornerstone. The finding affected the objective to ensure adequate protection of the worker's health and safety from exposure to unintended radiation from radioactive material during routine civilian nuclear reactor operation. Using the Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined the finding was of very low safety significance because (1) it was not associated with as low as is reasonably achievable (ALARA) planning or work controls, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This finding was determined to have a crosscutting aspect in the area of human performance, associated with work practices, because the licensee did not effectively communicate expectations regarding procedural compliance and the personnel following the procedures [H.4(b)].
Inspection Report# : 2011003 *(pdf)*

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## Public Radiation Safety

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## Physical Protection

Although the NRC is actively overseeing the Security cornerstone, the Commission has decided that certain findings pertaining to security cornerstone will not be publicly available to ensure that potentially useful information is not provided to a possible adversary. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : May 29, 2012

# Diablo Canyon 2
# 2Q/2012 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure for the Control of Tools for Use on Stainless Steel**
Inspectors identified a non-cited violation of Technical Specification 5.4.1.e, for the failure to follow procedures that ensured hand files and wire brushes designated for stainless steel weld preparation were stored and maintained separately from hand files and wire brushes used on carbon steel. Specifically, the inspectors determined that the licensee was not segregating tools as required by Procedure MA1.ID12, "Control of Tools for Use on Stainless Steel," Revision 1, because inspectors observed rust deposits on stainless steel components in the plant. This indicated that carbon steel contaminated tools may have been used on these systems. The licensee took corrective actions to segregate the stainless steel tools that were mixed with tools used on carbon steel. The licensee established segregated locations in tool rooms for the separation of abrasive tools, trained tool room attendants to properly store and mark abrasive tools designated for use on stainless steel and evaluated the systems with indications of rust deposits. This issue was entered into the licensee's corrective action program as Notifications 50475217 and 50475779.
Failure to assure that hand files and wire brushes designated for exclusive use on stainless steel were stored separately from tools used on other materials was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and, if left uncorrected, could become a more significant safety concern. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: FIN Finding
**Feedwater System Weld Flaw**
The inspectors identified a finding for failure to follow applicable ASME Code requirements prior to returning the feedwater system to service after code repairs for flow accelerated corrosion. The licensee failed to recognize a rejectable indication in feedwater piping weld 2K16-550-30 FW 33 observable in the original acceptance radiography film. The licensee entered the issue into their corrective action program as Notifications 50473769 and 50475897 and re-examined the radiographic films for welds performed during Refueling Outage 2R16. A random re-examination of other radiographic films will be completed at a later date.

This finding was more than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and directly affected the cornerstone objective of limiting events that challenge plant stability. Based on the results of the engineering evaluation that was performed when the flaw was recognized, the inspectors determined that the structural integrity of the feedwater piping was not affected. Based on the results of a significance determination process Phase 1 evaluation, the finding was determined to be of very low safety significance (Green) because it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or an internal/external flood. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure human error prevention techniques, such as self- and peer-checking were used so that work activities are performed safely.

Inspection Report# : 2012003 *(pdf)*

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# Mitigating Systems

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Preferrd Offsite Power System Design Control**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after plant engineers failed to adequately translate regulatory requirements and the design bases into the offsite power interface calculation on May 6, 2011. As a result, the licensee failed to demonstrate that the 230 kilo-Volt preferred offsite power source had adequate capacity and capability to supply the minimum required terminal voltage to plant engineering safety features following a limiting transmission system contingency. The licensee took corrective actions to limit the plant load that would automatically transfer to the preferred power source following a unit trip and entered the condition into the corrective action program as Notification 50492766.
The failure to ensure that the 230 kV power system had adequate capability and capability as defined in the current licensing basis requirements was a performance deficiency. This performance deficiency was more than minor because it was associated with the modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded thisfinding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a 50.59 Evaluation**
The inspectors identified a non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," because the licensee failed to document an evaluation providing a basis that changes made to the facility and associated changes to Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," did not require prior NRC approval. When a 50.59 review was performed, the licensee incorrectly concluded that only a screening was needed. Plant operators use Procedure OP J-2:VIII to determine the operability of the preferred offsite power system for various transmission system configurations. This change accepted a reduction in the preferred offsite power capacity and capability, below the minimum specified by the current licensing basis, due to local service area load growth. This condition would have likely required prior NRC approval had a 50.59 evaluation been performed. The licensee entered this finding into the corrective action program as Notification 50492767.
The failure to perform a 50.59 evaluation was also a performance deficiency. The inspectors concluded that this issue involved traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This performance deficiency is more than minor because it was associated with modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions to adopt the licensing basis requirement during decision making.

Inspection Report# : 2012003 *(pdf)*

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate OperabilityDetermination**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after operations personnel declared diesel generator 2-3 operable after failing to meet all surveillance test acceptance criterion. On December 22, 2011, diesel generator 2-3 did not meet frequency acceptance criteria during technical specification surveillance testing. Plant operators declared the diesel operable after applying an engineering evaluation. The inspectors identified that the evaluation was not appropriate to the conditions of the failed test. The licensee’s corrective actions included corrective maintenance, re-performance of the surveillance test, and entering the condition into the corrective action program as Notifications 50449027 and 50449504.

The failure of operations personnel to recognize that diesel generator surveillance results indicated that the system was not fully operable was a performance deficiency. This finding was more than minor because the licensee’s engineering evaluation created a reasonable doubt that the system was operable, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues.” The inspectors concluded that the finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in the loss of operability or functionality of a single train for greater than the technical specification outage time, did not represent an actual loss of safety function, and was not potentially risk significant due to a seismic, flooding, or severe weather event. The most significant contributor to this performance deficiency was that operators did not review and understand the diesel generator surveillance results sufficiently to recognize that the condition did not match the previously-evaluated condition that was used to conclude the diesel generator remained operable. Therefore, this finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component [P.1(c)].
Inspection Report# : 2012002 *(pdf)*

**Significance:** G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform an Operability Determination for New Seismic Information**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after Pacific Gas and Electric failed to evaluate the affect of new seismic information on the operability of plant structures, systems and components. On January 7, 2011, the licensee completed and submitted to the NRC a report to the detailing the results of a deterministic reevaluation of the local seismology. This report concluded that an earthquake on three local faults could produce greater vibratory ground motion than bound by the safe shutdown earthquake as described in the Final Safety Evaluation Report Update. Quality Procedure OM7.ID12, “Operability Determinations,” required plant operators to assess the impact of nonconforming conditions for the affect on plant structures, systems and components without delay. On June 22, 2011, the licensee entered the condition into the corrective action program as Notification 50410266 and completed an operability determination on June 24, 2011.

The inspectors determined that the licensee’s failure to evaluate the new seismic information against the plant design and licensing bases was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The senior reactor analyst evaluated the significance of the finding using a Phase 3 analysis because the inspectors were unable to confirm that the operability of plant systems was not impacted. The senior reactor analyst concluded that the finding was of very low risk significance (Green) because no significant change in overall core damage frequency resulted from the new seismic hazards. This finding had a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee used non-conservative assumptions in deciding not to evaluate the new seismic information against the current plant design and licensing bases (H.1.b).

Inspection Report# : 2011005 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain a Fire Barrier**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), "Fire Protection," after the licensee failed to maintain the integrity of a fire barrier. On July 21, 2011, the inspectors identified that Fire Door B43-2, entrance to the Residual Heat Removal Pump Room 2-2, was inoperable. Equipment Control Guideline 18.7, "Fire Rated Assemblies," required the licensee to maintain the fire barrier in the rated configuration or establish prescribed compensatory actions. The door was held open due to Auxiliary Building ventilation flow balance problems. The ventilation problems had affected the fire door since January 12, 2011. The inspectors performed an extent of condition evaluation and identified eight additional fire doors impacted by the flow balance problems. The licensee took immediate action to restore the fire door to the rated condition and entered the problem into the corrective action programs as Notification 50416374.

The inspectors concluded that the failure of the licensee to maintain a fire door in the rated configuration was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the licensee had maintained an automatic full area water-based fire suppression system in the exposed fire area. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not take timely corrective actions to correct Auxiliary Building ventilation flow balance issues.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Surveillances on Fire Barriers**
The inspectors identified a noncited violation of Diablo Canyon Facility Operating License Condition 2.C (4), "Fire Protection," after the licensee failed to identify and correct the failure to perform required surveillance testing on fire-rated assemblies. On August 16, 2011, the inspectors identified that the licensee had not performed Equipment Control Guideline 18.7, "Fire Rated Assemblies," surveillance testing on Fire Door 329-2, entrance to the 125VDC Battery 2-1 Room, and Fire Door 332-2, entrance to the 125VDC Battery 2-2 Room, within the required frequency. The inspectors also identified that both fire doors were degraded and did not meet the surveillance acceptance criteria. The licensee implemented the required compensatory actions for both fire doors and entered this finding into the corrective action program as Notification 50409975.

The inspectors concluded that the failure of the licensee to perform required surveillance tests on fire-rated assemblies was a performance deficiency. This finding was more than minor because the degraded fire barrier affected the Mitigating Systems Cornerstone external factors attribute objective to prevent undesirable consequences due to fire. The inspectors concluded that the finding was of very low safety significance (Green) because the exposed fire areas did not contain any potential damage targets unique from those in the exposing fire area. This finding had a crosscutting aspect in the area of problem identification and resolution associated with the corrective action program component because the licensee did not adequately prioritize and perform an extent of condition review of previous problems associated with excluding fire barriers from the Equipment Control Guideline requirements.
Inspection Report# : 2011004 *(pdf)*

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## Barrier Integrity

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: VIO Violation
**Incomplete and Inaccurate Information Provided to the NRC in Response to Generic Letter 2003-01, "Control Room Habitability"**
The inspectors identified a Green finding and Severity Level III violation of 10 CFR 50.9, “Completeness and Accuracy of Information,” after Pacific Gas and Electric failed to submitted complete and accurate information in response to Generic Letter 2003-01, “Control Room Habitability.” Generic Letter 2003-01 requested that the licensee submit information demonstrating that the control room habitability system was in compliance with the current licensing and design bases. The licensee was specifically requested to verify that the most limiting unfiltered in-leakage into the control room envelope was no more than the value assumed in the design basis radiological analyses for control room habitability. On April 22, 2005, the licensee reported to the NRC that testing performed in the most limiting configuration for operator dose demonstrated that there was no unfiltered in-leakage into the control room envelope. This was material because the NRC used this information to close out Generic Letter 2003-01. In September 2011, the inspectors identified that the control room test results were greater than the value assumed in the design basis radiological analysis and that the licensee’s testing was not performed in the most limiting configuration for operator dose. Using the actual control room in-leakage rates, the inspectors concluded that the resultant operator dose could have exceeded the limit established by current licensing and design bases during an accident.

The inspectors concluded that the failure of Pacific Gas and Electric to provide complete and accurate information in response to Generic Letter 2003-01 was a performance deficiency. The finding was more than minor because the information was material to the NRC’s decision making processes. The inspectors screened the issue through the Reactor Oversight Process because the finding included a performance deficiency that was reasonably within the licensee’s ability to control. The inspectors concluded that the finding was of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The inspectors also screened the issue through the traditional enforcement process because the violation impacted the regulatory process. The inspectors concluded that the violation was a Severity Level III because had the licensee provided complete and accurate information in their letter dated April 22, 2005, the NRC would have likely reconsidered a regulatory position or undertaken a substantial further inquiry. The inspectors did not identify a cross-cutting aspect because the performance deficiency was not reflective of present performance.
Inspection Report# : 2012002 *(pdf)*

**Significance:** G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Evaluation of a Nonconforming Control Room Habitability Train**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” after operations personnel failed to adequately evaluate the operability and extent of condition of a nonconforming control room habitability train. Beginning on August 30, 2011, the inspectors identified several nonconforming conditions associated with the habitability system, including disconnected ductwork, two 12 inch diameter openings in the envelope boundary, and less than adequate control room envelope pressurization and tracer gas surveillance tests. On November 7, 2011, the licensee re-performed the tracer gas test and observed gross unfiltered in-leakage into the control room envelope. Plant operators declared the habitability system inoperable. The licensee restored system operability after implementing a series of compensatory measures. The licensee entered the finding into the corrective action program as Notification 50425114 and plans to restore the system to the current licensing basis condition.

The inspectors concluded that the failure of plant operators to adequately evaluate the operability and extent of a nonconforming condition was a performance deficiency. This finding was more than minor because the licensee’s operability evaluation created a reasonable doubt that the system was capable of performing the specified safety function, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues.” The inspectors concluded that the finding was of very low safety significance because only the radiological barrier function of the control room was affected. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the degraded control room ventilation train for operability and extent of condition [P.1(c)].

Inspection Report# : 2011005 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Maintain the Control Room Habitability System in the Design Configuration**
The inspectors identified a Green noncited violation of Technical Specification 5.5.19, “Control Room Envelope Habitability Program,” after the licensee failed to maintain the Unit 1 control room ventilation train in the design configuration. The inspectors identified that Unit 1 control room ventilation system was in a degraded/non-conforming condition on August 31, 2011. The inspectors observed airflow bypassing the control room inlet header through disconnected ductwork. Technical Specification 5.5.19 required the licensee to maintain the habitability system in the most limited configuration used during the tracer gas in-leakage test. The disconnected ductwork was a more limiting condition than the tested configuration. The licensee took corrective action to declare the control room envelope inoperable and entered the finding into the corrective action program as Notification 50425114.

The inspectors determined that the failure of the licensee to maintain the control room habitability system in the design configuration was a performance deficiency. This finding was more than minor because it was associated with the configuration control attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. The inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a crosscutting aspect in the area of human performance associated with work control in that the licensee failed to appropriately plan work activities consistent with nuclear safety.
Inspection Report# : 2011004 *(pdf)*

**Significance:** G Sep 25, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow a Procedural Requirement for Reactivity Manipulation**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion V, “Procedures,” after operations personnel conducted a reactivity manipulation during shift turnover. Procedure OP1.ID3, “Reactivity Management Program,” required plant operators to suspend reactivity manipulations during shift turnover. On March 27, 2011, plant operators conducted a continuous dilution during shift turnover. The licensee entered this condition into the corrective action program as Notification 50407054.

The inspectors concluded that the failure of operations personnel to follow Procedure OP1.ID3 was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the procedure adherence area of the human performance attribute of the barrier integrity cornerstone and affected the objective to provide reasonable assurance that design barriers will protect the public from radionuclide releases. The inspectors concluded that the finding was of very low safety significance (Green) because only the fuel barrier was affected by the performance deficiency. The finding has a crosscutting aspect in the area of human performance, associated with work practices component, because the licensee failed to define and effectively communicate expectations regarding procedural compliance.
Inspection Report# : 2011004 *(pdf)*

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## Emergency Preparedness

**Significance:** G Sep 25, 2011
Identified By: NRC

Item Type: NCV NonCited Violation
**Failure to Ensure Emergency Response Organization Qualifications**
A noncited violation of 10 CFR 50.47(b)(10) was identified for the licensee’s failure to ensure a range of protective actions is available for emergency workers during emergencies. Specifically, an operator filled an on-shift emergency response organization watch position with expired self-contained breathing apparatus respiratory protection qualifications. The licensee has entered this issue into the corrective action program as Notification 50420127.

The failure to ensure that an emergency response organization on-shift watch stander was respiratory protection qualified is a performance deficiency. This finding is greater than minor because it affects the emergency response organization readiness attribute of the emergency preparedness cornerstone to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The finding is of very low safety significance because it was not a loss of a planning standard function. The finding had a human performance crosscutting aspect of conservative assumptions under the decision making component because the licensee did not ensure that personnel filling the minimum shift staffing emergency response organization positions were qualified to take the watch.
Inspection Report# : 2011004 *(pdf)*

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# Occupational Radiation Safety

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# Public Radiation Safety

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# Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

**Significance: N/A** Mar 24, 2012
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**
The inspection team concluded that the implementation of the corrective action program and overall performance related to identifying, evaluating, and resolving problems at Diablo Canyon was generally effective. Licensee identified problems were entered into the corrective action program at an appropriately low threshold. Problems were effectively prioritized and evaluated commensurate with the safety significance of the problems. Corrective actions were effectively implemented in a timely manner commensurate with their importance to safety and addressed the identified causes of problems. Lessons learned from industry operating experience were effectively reviewed and applied when appropriate. Audits and self-assessments were effectively used to identify problems and appropriate actions. Finally, Diablo Canyon effectively established and maintained a Safety Conscious Work Environment.
Inspection Report# : 2012007 *(pdf)*

Last modified : September 12, 2012

# Diablo Canyon 2
# 3Q/2012 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure for the Control of Tools for Use on Stainless Steel**
Inspectors identified a non-cited violation of Technical Specification 5.4.1.e, for the failure to follow procedures that ensured hand files and wire brushes designated for stainless steel weld preparation were stored and maintained separately from hand files and wire brushes used on carbon steel. Specifically, the inspectors determined that the licensee was not segregating tools as required by Procedure MA1.ID12, "Control of Tools for Use on Stainless Steel," Revision 1, because inspectors observed rust deposits on stainless steel components in the plant. This indicated that carbon steel contaminated tools may have been used on these systems. The licensee took corrective actions to segregate the stainless steel tools that were mixed with tools used on carbon steel. The licensee established segregated locations in tool rooms for the separation of abrasive tools, trained tool room attendants to properly store and mark abrasive tools designated for use on stainless steel and evaluated the systems with indications of rust deposits. This issue was entered into the licensee's corrective action program as Notifications 50475217 and 50475779.
Failure to assure that hand files and wire brushes designated for exclusive use on stainless steel were stored separately from tools used on other materials was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and, if left uncorrected, could become a more significant safety concern. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: FIN Finding
**Feedwater System Weld Flaw**
The inspectors identified a finding for failure to follow applicable ASME Code requirements prior to returning the feedwater system to service after code repairs for flow accelerated corrosion. The licensee failed to recognize a rejectable indication in feedwater piping weld 2K16-550-30 FW 33 observable in the original acceptance radiography film. The licensee entered the issue into their corrective action program as Notifications 50473769 and 50475897 and re-examined the radiographic films for welds performed during Refueling Outage 2R16. A random re-examination of other radiographic films will be completed at a later date.

This finding was more than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and directly affected the cornerstone objective of limiting events that challenge plant stability. Based on the results of the engineering evaluation that was performed when the flaw was recognized, the inspectors

determined that the structural integrity of the feedwater piping was not affected. Based on the results of a significance determination process Phase 1 evaluation, the finding was determined to be of very low safety significance (Green) because it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or an internal/external flood. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure human error prevention techniques, such as self- and peer-checking were used so that work activities are performed safely.
Inspection Report# : 2012003 *(pdf)*

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# Mitigating Systems

**Significance:** Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Preferrd Offsite Power System Design Control**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after plant engineers failed to adequately translate regulatory requirements and the design bases into the offsite power interface calculation on May 6, 2011. As a result, the licensee failed to demonstrate that the 230 kilo-Volt preferred offsite power source had adequate capacity and capability to supply the minimum required terminal voltage to plant engineering safety features following a limiting transmission system contingency. The licensee took corrective actions to limit the plant load that would automatically transfer to the preferred power source following a unit trip and entered the condition into the corrective action program as Notification 50492766.
The failure to ensure that the 230 kV power system had adequate capability and capability as defined in the current licensing basis requirements was a performance deficiency. This performance deficiency was more than minor because it was associated with the modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded thisfinding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

**Significance:** Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a 50.59 Evaluation**
The inspectors identified a non-cited violation of 10 CFR 50.59, “Changes, Tests, and Experiments,” because the licensee failed to document an evaluation providing a basis that changes made to the facility and associated changes to Procedure OP J-2:VIII, “Guidelines for Reliable Transmission Service for DCPP,” did not require prior NRC approval. When a 50.59 review was performed, the licensee incorrectly concluded that only a screening was needed. Plant operators use Procedure OP J-2:VIII to determine the operability of the preferred offsite power system for various transmission system configurations. This change accepted a reduction in the preferred offsite power capacity and capability, below the minimum specified by the current licensing basis, due to local service area load growth. This condition would have likely required prior NRC approval had a 50.59 evaluation been performed. The licensee

entered this finding into the corrective action program as Notification 50492767.
The failure to perform a 50.59 evaluation was also a performance deficiency. The inspectors concluded that this issue involved traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This performance deficiency is more than minor because it was associated with modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions to adopt the licensing basis requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate OperabilityDetermination**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," after operations personnel declared diesel generator 2-3 operable after failing to meet all surveillance test acceptance criterion. On December 22, 2011, diesel generator 2-3 did not meet frequency acceptance criteria during technical specification surveillance testing. Plant operators declared the diesel operable after applying an engineering evaluation. The inspectors identified that the evaluation was not appropriate to the conditions of the failed test. The licensee's corrective actions included corrective maintenance, re-performance of the surveillance test, and entering the condition into the corrective action program as Notifications 50449027 and 50449504.

The failure of operations personnel to recognize that diesel generator surveillance results indicated that the system was not fully operable was a performance deficiency. This finding was more than minor because the licensee's engineering evaluation created a reasonable doubt that the system was operable, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues." The inspectors concluded that the finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in the loss of operability or functionality of a single train for greater than the technical specification outage time, did not represent an actual loss of safety function, and was not potentially risk significant due to a seismic, flooding, or severe weather event. The most significant contributor to this performance deficiency was that operators did not review and understand the diesel generator surveillance results sufficiently to recognize that the condition did not match the previously-evaluated condition that was used to conclude the diesel generator remained operable. Therefore, this finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component [P.1(c)].
Inspection Report# : 2012002 *(pdf)*

**Significance:** G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform an Operability Determination for New Seismic Information**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to evaluate the affect of new seismic information on the operability of plant structures, systems and components. On January 7, 2011, the licensee completed and submitted to the NRC a report to the detailing the results of a deterministic reevaluation of the local seismology. This report concluded that an earthquake on three local faults could produce greater vibratory ground motion than bound by the

safe shutdown earthquake as described in the Final Safety Evaluation Report Update. Quality Procedure OM7.ID12, “Operability Determinations,” required plant operators to assess the impact of nonconforming conditions for the affect on plant structures, systems and components without delay. On June 22, 2011, the licensee entered the condition into the corrective action program as Notification 50410266 and completed an operability determination on June 24, 2011.

The inspectors determined that the licensee’s failure to evaluate the new seismic information against the plant design and licensing bases was a performance deficiency. The finding was more than minor because the performance deficiency was associated with the Mitigating Systems Cornerstone initial design control attribute and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The senior reactor analyst evaluated the significance of the finding using a Phase 3 analysis because the inspectors were unable to confirm that the operability of plant systems was not impacted. The senior reactor analyst concluded that the finding was of very low risk significance (Green) because no significant change in overall core damage frequency resulted from the new seismic hazards. This finding had a crosscutting aspect in the area of human performance associated with the decision-making component because the licensee used non-conservative assumptions in deciding not to evaluate the new seismic information against the current plant design and licensing bases (H.1.b).

Inspection Report# : 2011005 *(pdf)*

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# Barrier Integrity

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: VIO Violation
**Incomplete and Inaccurate Information Provided to the NRC in Response to Generic Letter 2003-01, "Control Room Habitability"**
The inspectors identified a Green finding and Severity Level III violation of 10 CFR 50.9, “Completeness and Accuracy of Information,” after Pacific Gas and Electric failed to submitted complete and accurate information in response to Generic Letter 2003-01, “Control Room Habitability.” Generic Letter 2003-01 requested that the licensee submit information demonstrating that the control room habitability system was in compliance with the current licensing and design bases. The licensee was specifically requested to verify that the most limiting unfiltered in-leakage into the control room envelope was no more than the value assumed in the design basis radiological analyses for control room habitability. On April 22, 2005, the licensee reported to the NRC that testing performed in the most limiting configuration for operator dose demonstrated that there was no unfiltered in-leakage into the control room envelope. This was material because the NRC used this information to close out Generic Letter 2003-01. In September 2011, the inspectors identified that the control room test results were greater than the value assumed in the design basis radiological analysis and that the licensee’s testing was not performed in the most limiting configuration for operator dose. Using the actual control room in-leakage rates, the inspectors concluded that the resultant operator dose could have exceeded the limit established by current licensing and design bases during an accident.

The inspectors concluded that the failure of Pacific Gas and Electric to provide complete and accurate information in response to Generic Letter 2003-01 was a performance deficiency. The finding was more than minor because the information was material to the NRC’s decision making processes. The inspectors screened the issue through the Reactor Oversight Process because the finding included a performance deficiency that was reasonably within the licensee’s ability to control. The inspectors concluded that the finding was of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The inspectors also screened the issue through the traditional enforcement process because the violation impacted the regulatory process. The inspectors

concluded that the violation was a Severity Level III because had the licensee provided complete and accurate information in their letter dated April 22, 2005, the NRC would have likely reconsidered a regulatory position or undertaken a substantial further inquiry. The inspectors did not identify a cross-cutting aspect because the performance deficiency was not reflective of present performance.
Inspection Report# : 2012002 *(pdf)*

**Significance:** G Dec 31, 2011
Identified By: NRC
Item Type: NCV NonCited Violation
**Less than Adequate Evaluation of a Nonconforming Control Room Habitability Train**
The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after operations personnel failed to adequately evaluate the operability and extent of condition of a nonconforming control room habitability train. Beginning on August 30, 2011, the inspectors identified several nonconforming conditions associated with the habitability system, including disconnected ductwork, two 12 inch diameter openings in the envelope boundary, and less than adequate control room envelope pressurization and tracer gas surveillance tests. On November 7, 2011, the licensee re-performed the tracer gas test and observed gross unfiltered in-leakage into the control room envelope. Plant operators declared the habitability system inoperable. The licensee restored system operability after implementing a series of compensatory measures. The licensee entered the finding into the corrective action program as Notification 50425114 and plans to restore the system to the current licensing basis condition.

The inspectors concluded that the failure of plant operators to adequately evaluate the operability and extent of a nonconforming condition was a performance deficiency. This finding was more than minor because the licensee's operability evaluation created a reasonable doubt that the system was capable of performing the specified safety function, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues." The inspectors concluded that the finding was of very low safety significance because only the radiological barrier function of the control room was affected. This finding had a crosscutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the degraded control room ventilation train for operability and extent of condition [P.1(c)].

Inspection Report# : 2011005 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

**Significance: N/A** Mar 24, 2012
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**
The inspection team concluded that the implementation of the corrective action program and overall performance related to identifying, evaluating, and resolving problems at Diablo Canyon was generally effective. Licensee identified problems were entered into the corrective action program at an appropriately low threshold. Problems were effectively prioritized and evaluated commensurate with the safety significance of the problems. Corrective actions were effectively implemented in a timely manner commensurate with their importance to safety and addressed the identified causes of problems. Lessons learned from industry operating experience were effectively reviewed and applied when appropriate. Audits and self-assessments were effectively used to identify problems and appropriate actions. Finally, Diablo Canyon effectively established and maintained a Safety Conscious Work Environment.
Inspection Report# : 2012007 *(pdf)*

Last modified : November 30, 2012

# Diablo Canyon 2
# 4Q/2012 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Emergency Operating Procedures**
The inspectors identified a self-revealing non-cited violation of Technical Specification 5.4.1(b) for failure to maintain emergency operating procedures after personnel reviewing a temporary modification failed to identify and change affected emergency operating procedures. Specifically, the emergency operating procedure EOP E-0.1, "Reactor Trip Response," Revision 28, was not updated to be consistent with a temporary modification of steam generator water level low-low bistable setpoints. The licensee entered the condition into the corrective action program as Notifications 50517883, 50520697, and 50518355.

The failure to update emergency operating procedure E-0.1 "Reactor Trip Response," Revision 28, to account for higher low-low water level bistable reset setpoints introduced by Temporary Modification 60044709 was a performance deficiency. The finding was more than minor because it was associated with the procedure quality attribute of the Initiating Events cornerstone. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance (Green) because the finding does not represent a loss of system and/or function and does not represent an actual loss of function of at least a single train for greater than its Technical Specification allowed outage time, or two separate safety systems out-of-service for greater than its Technical Specification allowed outage time. This finding had a crosscutting aspect in the area of human performance, associated with the resources component, because the licensee did not ensure complete, accurate and up-to-date procedures were available and adequate to ensure nuclear safety
Inspection Report# : 2012005 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure for the Control of Tools for Use on Stainless Steel**
Inspectors identified a non-cited violation of Technical Specification 5.4.1.e, for the failure to follow procedures that ensured hand files and wire brushes designated for stainless steel weld preparation were stored and maintained separately from hand files and wire brushes used on carbon steel. Specifically, the inspectors determined that the licensee was not segregating tools as required by Procedure MA1.ID12, "Control of Tools for Use on Stainless Steel," Revision 1, because inspectors observed rust deposits on stainless steel components in the plant. This indicated that carbon steel contaminated tools may have been used on these systems. The licensee took corrective actions to segregate the stainless steel tools that were mixed with tools used on carbon steel. The licensee established segregated locations in tool rooms for the separation of abrasive tools, trained tool room attendants to properly store and mark abrasive tools designated for use on stainless steel and evaluated the systems with indications of rust deposits. This issue was entered into the licensee's corrective action program as Notifications 50475217 and 50475779.
Failure to assure that hand files and wire brushes designated for exclusive use on stainless steel were stored separately from tools used on other materials was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and, if left uncorrected, could become a more significant safety concern. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure supervisory and management

oversight of work activities, including contractors, such that nuclear safety is supported.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: FIN Finding
**Feedwater System Weld Flaw**
The inspectors identified a finding for failure to follow applicable ASME Code requirements prior to returning the feedwater system to service after code repairs for flow accelerated corrosion. The licensee failed to recognize a rejectable indication in feedwater piping weld 2K16-550-30 FW 33 observable in the original acceptance radiography film. The licensee entered the issue into their corrective action program as Notifications 50473769 and 50475897 and re-examined the radiographic films for welds performed during Refueling Outage 2R16. A random re-examination of other radiographic films will be completed at a later date.

This finding was more than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and directly affected the cornerstone objective of limiting events that challenge plant stability. Based on the results of the engineering evaluation that was performed when the flaw was recognized, the inspectors determined that the structural integrity of the feedwater piping was not affected. Based on the results of a significance determination process Phase 1 evaluation, the finding was determined to be of very low safety significance (Green) because it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or an internal/external flood. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure human error prevention techniques, such as self- and peer-checking were used so that work activities are performed safely.
Inspection Report# : 2012003 *(pdf)*

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# Mitigating Systems

**Significance:** G Dec 20, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Maintain Required Firewater System Configuration**
The team reviewed a self-revealing non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, “Fire Protection Program,” due to the licensee inadvertently isolating the firewater yard loop for approximately three days, reducing the plant’s fire protection capability without compensatory actions. The licensee entered this issue in their corrective action program as Notification 50513006.

The failure to maintain the fire water system configuration as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The performance deficiency affected the fire protection defense-in depth strategies involving post-fire safe shutdown systems. The major fire loading in the yard area resulted from the 12 large transformers. The senior reactor analyst made the bounding assumption that any transformer fire without suppression would result in an unrecoverable loss of offsite power. A bounding value was calculated by multiplying the fire ignition frequency by the conditional core damage probability. This resulted in a change to core damage frequency of 1.2 x 10-7. Therefore, the subject finding was of very low safety significance (Green).

This performance deficiency had a cross-cutting aspect in the area of resources associated with providing complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. Specifically, the licensee did not provide sufficient details in procedures for operators to successfully align an infrequently operated valve with no position indication. [H.2(c)]
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Dec 20, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Compensatory Measures for Fire Protection Program Deficiencies**
The team identified a non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, "Fire Protection Program," due to the licensee's failure to establish or adequately implement compensatory measures for non-compliances with the licensee's approved fire protection program. These non-compliances were identified during the licensee's ongoing transition to a new fire protection program in compliance with National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," (NFPA 805). The licensee entered this issue in their corrective action program as Notifications 50521360 and 50531363.

The failure to establish or maintain appropriate compensatory measures for identified deficiencies in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A senior reactor analyst evaluated the significance of this performance deficiency.

A fire that results in the loss of switchgear room ventilation would cause a loss of all ac and dc power if operators did not take action to recover cooling. The analyst determined that the licensed operators would have at least two clear annunciators indicating that ventilation had been lost and that room temperatures were increasing. Additionally, Procedure CP-M10, "Fire Protection of Safe Shutdown Equipment", was available to assist in providing portable fan cooling to the rooms.

For a fire to result in an intersystem loss of coolant accident, it would have to cause a 3 phase hot short on both of two shutdown cooling suction valves. Given that each valve is on a different electrical train, the analyst determined that the conditional probabilities of the hot shorts involved would best be modeled as independent. Accounting for the risk associated with both issues evaluated, the analyst estimated the change to core damage probability to be 1.5 x 10-7 per unit. Therefore, the performance deficiency was considered to be of very low safety significance (Green).

This finding did not have a cross-cutting aspect because it was not indicative of the licensee's present performance.
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Sep 30, 2012
Identified By: NRC
Item Type: VIO Violation
**Inadequate Corrective Actions to Update the Final Safety Analysis Report Update (FSARU) with Required Information**
The inspectors identified a cited violation of 10 CFR Part 50.71(e), "Maintenance of Records, Making of Reports," for failing to update the Final Safety Analysis Report. Specifically, the licensee failed to update the Final Safety Analysis Report to include the information describing the extent to which plant structures, systems, and components met 10 CFR 50, Appendix A, or describing and justifying exceptions to those General Design Criteria. This failure to update the Final Safety Analysis Report was previously identified as a non-cited violation in NRC's "Diablo Canyon Power Plant Integrated Inspection Report 05000275/2009003 and 05000323/2009003." The licensee entered the condition into the corrective action program as Notification 50513243.

The failure to correct missing information that was required to be in the Final Safety Analysis Report Update was a performance deficiency. The inspectors concluded that the finding is more than minor because, if left uncorrected, this could lead to a more significant safety concern because future changes to the facility, procedures, and programs would not be able to consider the licensing basis information that was removed or never inserted. The finding was screened using Manual Chapter 0609, "Significance Determination Process." The inspectors concluded that the finding was of very low safety significance (Green) because while the finding was a deficiency affecting design or qualification of a mitigating system, it did not result in the loss of operability or functionality of a system. The finding also affected the

NRC's ability to perform its regulatory function and was evaluated using the traditional enforcement process. The finding was determined to be Severity Level IV because the required information was not used to make an unacceptable change to the facility or procedures, which was consistent with the determination that the issue had very low safety significance. The inspectors concluded that this finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions in decision making and did not adopt a requirement to demonstrate that the proposed action is safe in order to proceed.
Inspection Report# : 2012004 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Preferrd Offsite Power System Design Control**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after plant engineers failed to adequately translate regulatory requirements and the design bases into the offsite power interface calculation on May 6, 2011. As a result, the licensee failed to demonstrate that the 230 kilo-Volt preferred offsite power source had adequate capacity and capability to supply the minimum required terminal voltage to plant engineering safety features following a limiting transmission system contingency. The licensee took corrective actions to limit the plant load that would automatically transfer to the preferred power source following a unit trip and entered the condition into the corrective action program as Notification 50492766.
The failure to ensure that the 230 kV power system had adequate capability and capability as defined in the current licensing basis requirements was a performance deficiency. This performance deficiency was more than minor because it was associated with the modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded thisfinding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a 50.59 Evaluation**
The inspectors identified a non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," because the licensee failed to document an evaluation providing a basis that changes made to the facility and associated changes to Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," did not require prior NRC approval. When a 50.59 review was performed, the licensee incorrectly concluded that only a screening was needed. Plant operators use Procedure OP J-2:VIII to determine the operability of the preferred offsite power system for various transmission system configurations. This change accepted a reduction in the preferred offsite power capacity and capability, below the minimum specified by the current licensing basis, due to local service area load growth. This condition would have likely required prior NRC approval had a 50.59 evaluation been performed. The licensee entered this finding into the corrective action program as Notification 50492767.
The failure to perform a 50.59 evaluation was also a performance deficiency. The inspectors concluded that this issue involved traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory function. This performance deficiency is more than minor because it was associated with modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen

as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions to adopt the licensing basis requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate OperabilityDetermination**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after operations personnel declared diesel generator 2-3 operable after failing to meet all surveillance test acceptance criterion. On December 22, 2011, diesel generator 2-3 did not meet frequency acceptance criteria during technical specification surveillance testing. Plant operators declared the diesel operable after applying an engineering evaluation. The inspectors identified that the evaluation was not appropriate to the conditions of the failed test. The licensee’s corrective actions included corrective maintenance, re-performance of the surveillance test, and entering the condition into the corrective action program as Notifications 50449027 and 50449504.

The failure of operations personnel to recognize that diesel generator surveillance results indicated that the system was not fully operable was a performance deficiency. This finding was more than minor because the licensee’s engineering evaluation created a reasonable doubt that the system was operable, similar to Example 3.k in Inspection Manual Chapter 0612, Appendix E, “Examples of Minor Issues.” The inspectors concluded that the finding was of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not result in the loss of operability or functionality of a single train for greater than the technical specification outage time, did not represent an actual loss of safety function, and was not potentially risk significant due to a seismic, flooding, or severe weather event. The most significant contributor to this performance deficiency was that operators did not review and understand the diesel generator surveillance results sufficiently to recognize that the condition did not match the previously-evaluated condition that was used to conclude the diesel generator remained operable. Therefore, this finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component [P.1(c)].
Inspection Report# : 2012002 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Operability Evaluation**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after personnel failed to adequately assess the impact of an unanalyzed condition on control room envelope operability. Specifically, personnel performed a problem screening for a nonconforming condition that impacted operability of the control room ventilation system operability and determined that a review by the Shift Foreman, work control Shift Foreman, or Shift Manager was not required. The licensee entered the condition into the corrective action program as Notification 50497774.

The failure to adequately assess the impact of an unanalyzed, non-conservative condition on control room habitability system operability was a performance deficiency. This finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the impact of non-conservative control room atmospheric dispersion factor methodology on control room habitability system operability,

Inspection Report# : 2012005 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-conservative Decision Making Resulted in a Violation of Technical Specification**
The inspectors identified a non-cited violation of Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)," after the control room envelope boundary for both units was inoperable for a greater duration than permitted by the out-of-service time. Specifically, the licensee operated Units 1 and 2 without an operable control room envelope from between at least September 2011 and December 2012, which is greater than the 90 day allowed outage time. The licensee entered the condition into the corrective action program as Notifications 50483820, 50497328, and 50485800

The failure to comply with Technical Specification 3.7.10 was a performance deficiency. The finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance that the control room physical design would protect operators from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the for the control room. This finding had a crosscutting aspect in the area human performance associated with decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the control room envelope boundary,

Inspection Report# : 2012005 *(pdf)*

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: VIO Violation
**Incomplete and Inaccurate Information Provided to the NRC in Response to Generic Letter 2003-01, "Control Room Habitability"**
The inspectors identified a Green finding and Severity Level III violation of 10 CFR 50.9, “Completeness and Accuracy of Information,” after Pacific Gas and Electric failed to submitted complete and accurate information in response to Generic Letter 2003-01, “Control Room Habitability.” Generic Letter 2003-01 requested that the licensee submit information demonstrating that the control room habitability system was in compliance with the current licensing and design bases. The licensee was specifically requested to verify that the most limiting unfiltered in-leakage into the control room envelope was no more than the value assumed in the design basis radiological analyses for control room habitability. On April 22, 2005, the licensee reported to the NRC that testing performed in the most limiting configuration for operator dose demonstrated that there was no unfiltered in-leakage into the control room envelope. This was material because the NRC used this information to close out Generic Letter 2003-01. In September 2011, the inspectors identified that the control room test results were greater than the value assumed in the design basis radiological analysis and that the licensee’s testing was not performed in the most limiting configuration for operator dose. Using the actual control room in-leakage rates, the inspectors concluded that the resultant operator dose could have exceeded the limit established by current licensing and design bases during an accident.

The inspectors concluded that the failure of Pacific Gas and Electric to provide complete and accurate information in response to Generic Letter 2003-01 was a performance deficiency. The finding was more than minor because the information was material to the NRC’s decision making processes. The inspectors screened the issue through the Reactor Oversight Process because the finding included a performance deficiency that was reasonably within the licensee’s ability to control. The inspectors concluded that the finding was of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The inspectors also screened the issue through the traditional enforcement process because the violation impacted the regulatory process. The inspectors

concluded that the violation was a Severity Level III because had the licensee provided complete and accurate information in their letter dated April 22, 2005, the NRC would have likely reconsidered a regulatory position or undertaken a substantial further inquiry. The inspectors did not identify a cross-cutting aspect because the performance deficiency was not reflective of present performance.
Inspection Report# : 2012002 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

**Significance: N/A** Mar 24, 2012
Identified By: NRC
Item Type: FIN Finding
**Problem Identification and Resolution**
The inspection team concluded that the implementation of the corrective action program and overall performance related to identifying, evaluating, and resolving problems at Diablo Canyon was generally effective. Licensee identified problems were entered into the corrective action program at an appropriately low threshold. Problems were effectively prioritized and evaluated commensurate with the safety significance of the problems. Corrective actions were effectively implemented in a timely manner commensurate with their importance to safety and addressed the identified causes of problems. Lessons learned from industry operating experience were effectively reviewed and applied when appropriate. Audits and self-assessments were effectively used to identify problems and appropriate actions. Finally, Diablo Canyon effectively established and maintained a Safety Conscious Work Environment.
Inspection Report# : 2012007 *(pdf)*

Last modified : February 28, 2013

# Diablo Canyon 2
# 1Q/2013 Plant Inspection Findings

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## Initiating Events

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Provide Adequate Guidance To Address General Welding Standard Requirements**
On February 14, 2013, the inspectors observed field welders add a partial circumferential weld on one side of the pipe in efforts to repair the pipe misalignment prior to the completion of the final visual inspection. This action represents a violation of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," because the licensee's procedure established special controls for critical distortions but failed to adequately define what situations fit that category. The licensee reviewed the stress calculation for the piping in question and concluded that the addition of the weld filler material did not affect the fatigue resistance of the weld, but acknowledged that a definition and additional guidance for the term "critical" was missing in the procedure and could have adverse effects on future final welds. The licensee entered the finding into their corrective action program as Notification 50542347.

The inspectors determined that the failure of the site's welding standard to provide adequate guidance to identify what constitutes a weld distortion during welding activities is a performance deficiency. The finding is more than minor because if left uncorrected, it has the potential to lead to a more significant safety concern. Specifically, Procedure GSW ASME did not provide the necessary guidance for welders and quality assurance personnel to identify and understand what constitutes critical distortion of a weld. The welding process can introduce effects of weld shrinkage (stresses) and distortion that could adversely affect the final condition of the weld, potentially leading to a service induced failure. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined the finding had a cross cutting aspect in the human performance area associated with work practices, procedural compliance, because the licensee did not adequately define or train welders to know what constituted a critical distortion, and did not effectively communicate the expectation of questioning the procedure if the welding activity required skill of the craft. [H.4(b)]

Inspection Report# : 2013002 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Identify Existing Indications During Prior Ultrasonic Examinations Of Pressurizer Structural Weld Overlays**
The inspectors identified a Green non-cited violation of 10 CFR 50.55a(a)(3)(i), which requires that proposed alternatives to industry codes and standards provide an acceptable level of quality and safety. The NRC staff approved relief request REP 1 U2 dated March 28, 2007, for installing six structural weld overlays on the pressurizer safety, relief, spray and surge nozzles. The request established acceptance criteria of laminar flaws during weld acceptance examinations limited to only the third 10 year inservice inspection interval. Contrary to the above, the licensee failed

to identify unacceptable flaws as defined by the approved request following completion of these welds in 2008. The licensee entered the finding into their corrective action program as Notification 50540188.

The inspectors determined that the licensee’s failure to identify indications that exceeded the acceptable linear dimension of laminar flaws prior to placing the system in service is a performance deficiency. The performance deficiency is more than minor because it is associated with the initiating events cornerstone attribute of equipment performance, and adversely affects the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, during the months of February and March 2013, the licensee identified that three out of the six pressurizer structural weld overlays exhibited laminar flaws that exceeded the linear dimensions approved by the safety evaluation. Using Manual Chapter 0609, Attachment A, “The Significance Determination Process (SDP) for Findings At Power,” the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. This issue did not have a cross-cutting aspect associated with it because it is not indicative of current performance.
Inspection Report# : 2013002 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: Self-Revealing
Item Type: FIN Finding
**Failure to Effectively Evaluate Design Change for High Voltage Bushing**
The inspectors reviewed a self-revealing finding for failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications. Specifically, design engineering staff was not effective in using applicable station design documents, in conjunction with industry standards to determine minimum creepage distance for high voltage insulators when replacing ceramic bushings with polymer bushings on the main bank transformer. As a result, the licensee approved installation of an insulator stack that did not provide adequate ground protection, which caused a plant trip on October 11, 2012. The licensee entered the condition in their corrective action program as Notification 50518473.

Failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 1, “Initiating Events Screening Questions,” this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. This finding had a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions in decision making when considering the suitability of the design for the environment [H.1(b)].
Inspection Report# : 2013002 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Emergency Operating Procedures**
The inspectors identified a self-revealing non-cited violation of Technical Specification 5.4.1(b) for failure to maintain emergency operating procedures after personnel reviewing a temporary modification failed to identify and change affected emergency operating procedures. Specifically, the emergency operating procedure EOP E-0.1, “Reactor Trip

Response," Revision 28, was not updated to be consistent with a temporary modification of steam generator water level low-low bistable setpoints. The licensee entered the condition into the corrective action program as Notifications 50517883, 50520697, and 50518355.

The failure to update emergency operating procedure E-0.1 "Reactor Trip Response," Revision 28, to account for higher low-low water level bistable reset setpoints introduced by Temporary Modification 60044709 was a performance deficiency. The finding was more than minor because it was associated with the procedure quality attribute of the Initiating Events cornerstone. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance (Green) because the finding does not represent a loss of system and/or function and does not represent an actual loss of function of at least a single train for greater than its Technical Specification allowed outage time, or two separate safety systems out-of-service for greater than its Technical Specification allowed outage time. This finding had a crosscutting aspect in the area of human performance, associated with the resources component, because the licensee did not ensure complete, accurate and up-to-date procedures were available and adequate to ensure nuclear safety
Inspection Report# : 2012005 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure for the Control of Tools for Use on Stainless Steel**
Inspectors identified a non-cited violation of Technical Specification 5.4.1.e, for the failure to follow procedures that ensured hand files and wire brushes designated for stainless steel weld preparation were stored and maintained separately from hand files and wire brushes used on carbon steel. Specifically, the inspectors determined that the licensee was not segregating tools as required by Procedure MA1.ID12, "Control of Tools for Use on Stainless Steel," Revision 1, because inspectors observed rust deposits on stainless steel components in the plant. This indicated that carbon steel contaminated tools may have been used on these systems. The licensee took corrective actions to segregate the stainless steel tools that were mixed with tools used on carbon steel. The licensee established segregated locations in tool rooms for the separation of abrasive tools, trained tool room attendants to properly store and mark abrasive tools designated for use on stainless steel and evaluated the systems with indications of rust deposits. This issue was entered into the licensee's corrective action program as Notifications 50475217 and 50475779.
Failure to assure that hand files and wire brushes designated for exclusive use on stainless steel were stored separately from tools used on other materials was a performance deficiency. This finding is more than minor because it is associated with the equipment performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and, if left uncorrected, could become a more significant safety concern. Using Inspection Manual Chapter 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," this finding was determined to be of very low safety significance because the issue would not result in exceeding the technical specification limit for identified reactor coolant system leakage or affect other mitigating systems resulting in a total loss of their safety function. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: FIN Finding
**Feedwater System Weld Flaw**
The inspectors identified a finding for failure to follow applicable ASME Code requirements prior to returning the feedwater system to service after code repairs for flow accelerated corrosion. The licensee failed to recognize a rejectable indication in feedwater piping weld 2K16-550-30 FW 33 observable in the original acceptance radiography

film. The licensee entered the issue into their corrective action program as Notifications 50473769 and 50475897 and re-examined the radiographic films for welds performed during Refueling Outage 2R16. A random re-examination of other radiographic films will be completed at a later date.

This finding was more than minor because it is associated with the human performance attribute of the Initiating Events Cornerstone and directly affected the cornerstone objective of limiting events that challenge plant stability. Based on the results of the engineering evaluation that was performed when the flaw was recognized, the inspectors determined that the structural integrity of the feedwater piping was not affected. Based on the results of a significance determination process Phase 1 evaluation, the finding was determined to be of very low safety significance (Green) because it did not contribute to the likelihood of a loss of coolant accident, did not contribute to a loss of mitigation equipment, and did not increase the likelihood of a fire or an internal/external flood. This finding has a cross-cutting aspect in the area of human performance, work practices, in that the licensee failed to ensure human error prevention techniques, such as self- and peer-checking were used so that work activities are performed safely.
Inspection Report# : 2012003 *(pdf)*

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# Mitigating Systems

**Significance:** G Dec 20, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Maintain Required Firewater System Configuration**
The team reviewed a self-revealing non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, “Fire Protection Program,” due to the licensee inadvertently isolating the firewater yard loop for approximately three days, reducing the plant’s fire protection capability without compensatory actions. The licensee entered this issue in their corrective action program as Notification 50513006.

The failure to maintain the fire water system configuration as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The performance deficiency affected the fire protection defense-in depth strategies involving post-fire safe shutdown systems. The major fire loading in the yard area resulted from the 12 large transformers. The senior reactor analyst made the bounding assumption that any transformer fire without suppression would result in an unrecoverable loss of offsite power. A bounding value was calculated by multiplying the fire ignition frequency by the conditional core damage probability. This resulted in a change to core damage frequency of 1.2 x 10-7. Therefore, the subject finding was of very low safety significance (Green).

This performance deficiency had a cross-cutting aspect in the area of resources associated with providing complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. Specifically, the licensee did not provide sufficient details in procedures for operators to successfully align an infrequently operated valve with no position indication. [H.2(c)]
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Dec 20, 2012
Identified By: NRC
Item Type: NCV NonCited Violation

**Inadequate Compensatory Measures for Fire Protection Program Deficiencies**
The team identified a non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, "Fire Protection Program," due to the licensee's failure to establish or adequately implement compensatory measures for non-compliances with the licensee's approved fire protection program. These non-compliances were identified during the licensee's ongoing transition to a new fire protection program in compliance with National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," (NFPA 805). The licensee entered this issue in their corrective action program as Notifications 50521360 and 50531363.

The failure to establish or maintain appropriate compensatory measures for identified deficiencies in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A senior reactor analyst evaluated the significance of this performance deficiency.

A fire that results in the loss of switchgear room ventilation would cause a loss of all ac and dc power if operators did not take action to recover cooling. The analyst determined that the licensed operators would have at least two clear annunciators indicating that ventilation had been lost and that room temperatures were increasing. Additionally, Procedure CP-M10, "Fire Protection of Safe Shutdown Equipment", was available to assist in providing portable fan cooling to the rooms.

For a fire to result in an intersystem loss of coolant accident, it would have to cause a 3 phase hot short on both of two shutdown cooling suction valves. Given that each valve is on a different electrical train, the analyst determined that the conditional probabilities of the hot shorts involved would best be modeled as independent. Accounting for the risk associated with both issues evaluated, the analyst estimated the change to core damage probability to be 1.5 x 10-7 per unit. Therefore, the performance deficiency was considered to be of very low safety significance (Green).

This finding did not have a cross-cutting aspect because it was not indicative of the licensee's present performance.
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Sep 30, 2012
Identified By: NRC
Item Type: VIO Violation
**Inadequate Corrective Actions to Update the Final Safety Analysis Report Update (FSARU) with Required Information**
The inspectors identified a cited violation of 10 CFR Part 50.71(e), "Maintenance of Records, Making of Reports," for failing to update the Final Safety Analysis Report. Specifically, the licensee failed to update the Final Safety Analysis Report to include the information describing the extent to which plant structures, systems, and components met 10 CFR 50, Appendix A, or describing and justifying exceptions to those General Design Criteria. This failure to update the Final Safety Analysis Report was previously identified as a non-cited violation in NRC's "Diablo Canyon Power Plant Integrated Inspection Report 05000275/2009003 and 05000323/2009003." The licensee entered the condition into the corrective action program as Notification 50513243.

The failure to correct missing information that was required to be in the Final Safety Analysis Report Update was a performance deficiency. The inspectors concluded that the finding is more than minor because, if left uncorrected, this could lead to a more significant safety concern because future changes to the facility, procedures, and programs would not be able to consider the licensing basis information that was removed or never inserted. The finding was screened using Manual Chapter 0609, "Significance Determination Process." The inspectors concluded that the finding was of very low safety significance (Green) because while the finding was a deficiency affecting design or qualification of a

mitigating system, it did not result in the loss of operability or functionality of a system. The finding also affected the NRC's ability to perform its regulatory function and was evaluated using the traditional enforcement process. The finding was determined to be Severity Level IV because the required information was not used to make an unacceptable change to the facility or procedures, which was consistent with the determination that the issue had very low safety significance. The inspectors concluded that this finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions in decision making and did not adopt a requirement to demonstrate that the proposed action is safe in order to proceed.
Inspection Report# : 2012004 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Preferrd Offsite Power System Design Control**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," after plant engineers failed to adequately translate regulatory requirements and the design bases into the offsite power interface calculation on May 6, 2011. As a result, the licensee failed to demonstrate that the 230 kilo-Volt preferred offsite power source had adequate capacity and capability to supply the minimum required terminal voltage to plant engineering safety features following a limiting transmission system contingency. The licensee took corrective actions to limit the plant load that would automatically transfer to the preferred power source following a unit trip and entered the condition into the corrective action program as Notification 50492766.
The failure to ensure that the 230 kV power system had adequate capability and capability as defined in the current licensing basis requirements was a performance deficiency. This performance deficiency was more than minor because it was associated with the modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded thisfinding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not demonstrate that the proposed action was safe in order to proceed while assessing the CLB requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

**Significance:** G Jun 22, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform a 50.59 Evaluation**
The inspectors identified a non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," because the licensee failed to document an evaluation providing a basis that changes made to the facility and associated changes to Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," did not require prior NRC approval. When a 50.59 review was performed, the licensee incorrectly concluded that only a screening was needed. Plant operators use Procedure OP J-2:VIII to determine the operability of the preferred offsite power system for various transmission system configurations. This change accepted a reduction in the preferred offsite power capacity and capability, below the minimum specified by the current licensing basis, due to local service area load growth. This condition would have likely required prior NRC approval had a 50.59 evaluation been performed. The licensee entered this finding into the corrective action program as Notification 50492767.
The failure to perform a 50.59 evaluation was also a performance deficiency. The inspectors concluded that this issue involved traditional enforcement because it had the potential for impacting the NRC's ability to perform its regulatory

function. This performance deficiency is more than minor because it was associated with modification design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors concluded this finding was of very low safety significance because the duration of potential losses of a single offsite power source safety function was less than the technical specification allowed outage time, did not represent an actual loss of safety function of risk significant non-technical specification equipment, and did not screen as potentially risk significant due to seismic, flooding, or severe weather initiating events. This finding has a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions to adopt the licensing basis requirement during decision making.
Inspection Report# : 2012003 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Operability Evaluation**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after personnel failed to adequately assess the impact of an unanalyzed condition on control room envelope operability. Specifically, personnel performed a problem screening for a nonconforming condition that impacted operability of the control room ventilation system operability and determined that a review by the Shift Foreman, work control Shift Foreman, or Shift Manager was not required. The licensee entered the condition into the corrective action program as Notification 50497774.

The failure to adequately assess the impact of an unanalyzed, non-conservative condition on control room habitability system operability was a performance deficiency. This finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the impact of non-conservative control room atmospheric dispersion factor methodology on control room habitability system operability,

Inspection Report# : 2012005 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-conservative Decision Making Resulted in a Violation of Technical Specification**
The inspectors identified a non-cited violation of Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)," after the control room envelope boundary for both units was inoperable for a greater duration than permitted by the out-of-service time. Specifically, the licensee operated Units 1 and 2 without an operable control room envelope from between at least September 2011 and December 2012, which is greater than the 90 day allowed outage time. The licensee entered the condition into the corrective action program as Notifications 50483820, 50497328, and

50485800

The failure to comply with Technical Specification 3.7.10 was a performance deficiency. The finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance that the control room physical design would protect operators from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the for the control room. This finding had a crosscutting aspect in the area human performance associated with decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the control room envelope boundary,

Inspection Report# : 2012005 *(pdf)*

**Significance:** G Mar 23, 2012
Identified By: NRC
Item Type: VIO Violation
**Incomplete and Inaccurate Information Provided to the NRC in Response to Generic Letter 2003-01, "Control Room Habitability"**
The inspectors identified a Green finding and Severity Level III violation of 10 CFR 50.9, “Completeness and Accuracy of Information,” after Pacific Gas and Electric failed to submitted complete and accurate information in response to Generic Letter 2003-01, “Control Room Habitability.” Generic Letter 2003-01 requested that the licensee submit information demonstrating that the control room habitability system was in compliance with the current licensing and design bases. The licensee was specifically requested to verify that the most limiting unfiltered in-leakage into the control room envelope was no more than the value assumed in the design basis radiological analyses for control room habitability. On April 22, 2005, the licensee reported to the NRC that testing performed in the most limiting configuration for operator dose demonstrated that there was no unfiltered in-leakage into the control room envelope. This was material because the NRC used this information to close out Generic Letter 2003-01. In September 2011, the inspectors identified that the control room test results were greater than the value assumed in the design basis radiological analysis and that the licensee’s testing was not performed in the most limiting configuration for operator dose. Using the actual control room in-leakage rates, the inspectors concluded that the resultant operator dose could have exceeded the limit established by current licensing and design bases during an accident.

The inspectors concluded that the failure of Pacific Gas and Electric to provide complete and accurate information in response to Generic Letter 2003-01 was a performance deficiency. The finding was more than minor because the information was material to the NRC’s decision making processes. The inspectors screened the issue through the Reactor Oversight Process because the finding included a performance deficiency that was reasonably within the licensee’s ability to control. The inspectors concluded that the finding was of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The inspectors also screened the issue through the traditional enforcement process because the violation impacted the regulatory process. The inspectors concluded that the violation was a Severity Level III because had the licensee provided complete and accurate information in their letter dated April 22, 2005, the NRC would have likely reconsidered a regulatory position or undertaken a substantial further inquiry. The inspectors did not identify a cross-cutting aspect because the performance deficiency was not reflective of present performance.
Inspection Report# : 2012002 *(pdf)*

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## Emergency Preparedness

## Occupational Radiation Safety

## Public Radiation Safety

## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

## Miscellaneous

Last modified : June 04, 2013

# Diablo Canyon 2
# 2Q/2013 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jun 30, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Implement the Fire Protection Program Requirements for the Control of Transient Combustible Material**
Green. The inspectors identified a Green non-cited violation of the licensee's approved fire protection program as defined in Diablo Canyon Facility Operating License Conditions 2.C(5) for Unit 1 and 2.C(4) for Unit 2 involving the failure to effectively implement the fire protection program. Specifically, the inspectors identified multiple examples where the licensee failed to maintain control and tracking of combustible materials, welding equipment, and oxygen/acetylene rigs in the plant. The licensee entered the condition into the corrective action program as Notifications 50510062, 50511864, 50561959, and 50537650.
The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Initiating Events Cornerstone and it adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors concluded that the finding was of very low safety significance (Green) because each deficiency was rated as "Low" degradation because for the violations of the hot work permitting program, all normally required fire prevention measures remained in place and for the violations of the transient combustibles control program, the materials involved did not significantly increase the fire frequency. This finding had a cross-cutting aspect in the area of human performance associated with the work practices component, because the cause of the performance deficiency involved the licensee not ensuring supervisory and management oversight of work activities, such that nuclear safety was supported.
Inspection Report# : 2013003 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Provide Adequate Guidance To Address General Welding Standard Requirements**
On February 14, 2013, the inspectors observed field welders add a partial circumferential weld on one side of the pipe in efforts to repair the pipe misalignment prior to the completion of the final visual inspection. This action represents a violation of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," because the licensee's procedure established special controls for critical distortions but failed to adequately define what situations fit that category. The licensee reviewed the stress calculation for the piping in question and concluded that the addition of the weld filler material did not affect the fatigue resistance of the weld, but acknowledged that a definition and additional guidance for the term "critical" was missing in the procedure and could have adverse effects on future final welds. The licensee entered the finding into their corrective action program as Notification 50542347.

The inspectors determined that the failure of the site's welding standard to provide adequate guidance to identify what constitutes a weld distortion during welding activities is a performance deficiency. The finding is more than minor because if left uncorrected, it has the potential to lead to a more significant safety concern. Specifically, Procedure GSW ASME did not provide the necessary guidance for welders and quality assurance personnel to identify and understand what constitutes critical distortion of a weld. The welding process can introduce effects of weld shrinkage (stresses) and distortion that could adversely affect the final condition of the weld, potentially leading to a service induced failure. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined the finding had a cross cutting aspect in the human performance area associated with work practices, procedural compliance, because the licensee did not adequately define or train welders to know what constituted a critical distortion, and did not effectively communicate the expectation of questioning the procedure if the welding activity required skill of the craft. [H.4(b)]

Inspection Report# : 2013002 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Identify Existing Indications During Prior Ultrasonic Examinations Of Pressurizer Structural Weld Overlays**
The inspectors identified a Green non-cited violation of 10 CFR 50.55a(a)(3)(i), which requires that proposed alternatives to industry codes and standards provide an acceptable level of quality and safety. The NRC staff approved relief request REP 1 U2 dated March 28, 2007, for installing six structural weld overlays on the pressurizer safety, relief, spray and surge nozzles. The request established acceptance criteria of laminar flaws during weld acceptance examinations limited to only the third 10 year inservice inspection interval. Contrary to the above, the licensee failed to identify unacceptable flaws as defined by the approved request following completion of these welds in 2008. The licensee entered the finding into their corrective action program as Notification 50540188.

The inspectors determined that the licensee's failure to identify indications that exceeded the acceptable linear dimension of laminar flaws prior to placing the system in service is a performance deficiency. The performance deficiency is more than minor because it is associated with the initiating events cornerstone attribute of equipment performance, and adversely affects the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, during the months of February and March 2013, the licensee identified that three out of the six pressurizer structural weld overlays exhibited laminar flaws that exceeded the linear dimensions approved by the safety evaluation. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. This issue did not have a cross-cutting aspect associated with it because it is not indicative of current performance.
Inspection Report# : 2013002 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: Self-Revealing
Item Type: FIN Finding
**Failure to Effectively Evaluate Design Change for High Voltage Bushing**
The inspectors reviewed a self-revealing finding for failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications. Specifically, design engineering staff was not effective in using applicable station design documents, in conjunction with industry standards to determine minimum

creepage distance for high voltage insulators when replacing ceramic bushings with polymer bushings on the main bank transformer. As a result, the licensee approved installation of an insulator stack that did not provide adequate ground protection, which caused a plant trip on October 11, 2012. The licensee entered the condition in their corrective action program as Notification 50518473.

Failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 1, “Initiating Events Screening Questions,” this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. This finding had a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions in decision making when considering the suitability of the design for the environment [H.1(b)].
Inspection Report# : 2013002 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Emergency Operating Procedures**
The inspectors identified a self-revealing non-cited violation of Technical Specification 5.4.1(b) for failure to maintain emergency operating procedures after personnel reviewing a temporary modification failed to identify and change affected emergency operating procedures. Specifically, the emergency operating procedure EOP E-0.1, “Reactor Trip Response,” Revision 28, was not updated to be consistent with a temporary modification of steam generator water level low-low bistable setpoints. The licensee entered the condition into the corrective action program as Notifications 50517883, 50520697, and 50518355.

The failure to update emergency operating procedure E-0.1 “Reactor Trip Response,” Revision 28, to account for higher low-low water level bistable reset setpoints introduced by Temporary Modification 60044709 was a performance deficiency. The finding was more than minor because it was associated with the procedure quality attribute of the Initiating Events cornerstone. Using Inspection Manual Chapter 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” this finding was determined to be of very low safety significance (Green) because the finding does not represent a loss of system and/or function and does not represent an actual loss of function of at least a single train for greater than its Technical Specification allowed outage time, or two separate safety systems out-of-service for greater than its Technical Specification allowed outage time. This finding had a crosscutting aspect in the area of human performance, associated with the resources component, because the licensee did not ensure complete, accurate and up-to-date procedures were available and adequate to ensure nuclear safety
Inspection Report# : 2012005 *(pdf)*

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# Mitigating Systems

**Significance:** G Dec 20, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation

**Failure to Maintain Required Firewater System Configuration**
The team reviewed a self-revealing non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, “Fire Protection Program,” due to the licensee inadvertently isolating the firewater yard loop for approximately three days, reducing the plant’s fire protection capability without compensatory actions. The licensee entered this issue in their corrective action program as Notification 50513006.

The failure to maintain the fire water system configuration as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The performance deficiency affected the fire protection defense-in depth strategies involving post-fire safe shutdown systems. The major fire loading in the yard area resulted from the 12 large transformers. The senior reactor analyst made the bounding assumption that any transformer fire without suppression would result in an unrecoverable loss of offsite power. A bounding value was calculated by multiplying the fire ignition frequency by the conditional core damage probability. This resulted in a change to core damage frequency of 1.2 x 10-7. Therefore, the subject finding was of very low safety significance (Green).

This performance deficiency had a cross-cutting aspect in the area of resources associated with providing complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. Specifically, the licensee did not provide sufficient details in procedures for operators to successfully align an infrequently operated valve with no position indication. [H.2(c)]
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Dec 20, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Compensatory Measures for Fire Protection Program Deficiencies**
The team identified a non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, “Fire Protection Program,” due to the licensee’s failure to establish or adequately implement compensatory measures for non-compliances with the licensee’s approved fire protection program. These non-compliances were identified during the licensee’s ongoing transition to a new fire protection program in compliance with National Fire Protection Association Standard 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants,” (NFPA 805). The licensee entered this issue in their corrective action program as Notifications 50521360 and 50531363.

The failure to establish or maintain appropriate compensatory measures for identified deficiencies in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A senior reactor analyst evaluated the significance of this performance deficiency.

A fire that results in the loss of switchgear room ventilation would cause a loss of all ac and dc power if operators did not take action to recover cooling. The analyst determined that the licensed operators would have at least two clear annunciators indicating that ventilation had been lost and that room temperatures were increasing. Additionally, Procedure CP-M10, “Fire Protection of Safe Shutdown Equipment”, was available to assist in providing portable fan cooling to the rooms.

For a fire to result in an intersystem loss of coolant accident, it would have to cause a 3 phase hot short on both of two shutdown cooling suction valves. Given that each valve is on a different electrical train, the analyst determined that the conditional probabilities of the hot shorts involved would best be modeled as independent. Accounting for the risk associated with both issues evaluated, the analyst estimated the change to core damage probability to be 1.5 x 10-7 per unit. Therefore, the performance deficiency was considered to be of very low safety significance (Green).

This finding did not have a cross-cutting aspect because it was not indicative of the licensee’s present performance.
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Sep 30, 2012
Identified By: NRC
Item Type: VIO Violation
**Inadequate Corrective Actions to Update the Final Safety Analysis Report Update (FSARU) with Required Information**
The inspectors identified a cited violation of 10 CFR Part 50.71(e), “Maintenance of Records, Making of Reports,” for failing to update the Final Safety Analysis Report. Specifically, the licensee failed to update the Final Safety Analysis Report to include the information describing the extent to which plant structures, systems, and components met 10 CFR 50, Appendix A, or describing and justifying exceptions to those General Design Criteria. This failure to update the Final Safety Analysis Report was previously identified as a non-cited violation in NRC’s “Diablo Canyon Power Plant Integrated Inspection Report 05000275/2009003 and 05000323/2009003.” The licensee entered the condition into the corrective action program as Notification 50513243.

The failure to correct missing information that was required to be in the Final Safety Analysis Report Update was a performance deficiency. The inspectors concluded that the finding is more than minor because, if left uncorrected, this could lead to a more significant safety concern because future changes to the facility, procedures, and programs would not be able to consider the licensing basis information that was removed or never inserted. The finding was screened using Manual Chapter 0609, “Significance Determination Process.” The inspectors concluded that the finding was of very low safety significance (Green) because while the finding was a deficiency affecting design or qualification of a mitigating system, it did not result in the loss of operability or functionality of a system. The finding also affected the NRC’s ability to perform its regulatory function and was evaluated using the traditional enforcement process. The finding was determined to be Severity Level IV because the required information was not used to make an unacceptable change to the facility or procedures, which was consistent with the determination that the issue had very low safety significance. The inspectors concluded that this finding had a crosscutting aspect in the area of human performance associated with the decision making component because the licensee did not use conservative assumptions in decision making and did not adopt a requirement to demonstrate that the proposed action is safe in order to proceed.
Inspection Report# : 2012004 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Operability Evaluation**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after personnel failed to adequately assess the impact of an unanalyzed condition on control room envelope operability. Specifically, personnel performed a problem screening for a nonconforming condition that impacted operability of the control room ventilation system operability and determined that a review by

the Shift Foreman, work control Shift Foreman, or Shift Manager was not required. The licensee entered the condition into the corrective action program as Notification 50497774.

The failure to adequately assess the impact of an unanalyzed, non-conservative condition on control room habitability system operability was a performance deficiency. This finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the impact of non-conservative control room atmospheric dispersion factor methodology on control room habitability system operability,

Inspection Report# : 2012005 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-conservative Decision Making Resulted in a Violation of Technical Specification**
The inspectors identified a non-cited violation of Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)," after the control room envelope boundary for both units was inoperable for a greater duration than permitted by the out-of-service time. Specifically, the licensee operated Units 1 and 2 without an operable control room envelope from between at least September 2011 and December 2012, which is greater than the 90 day allowed outage time. The licensee entered the condition into the corrective action program as Notifications 50483820, 50497328, and 50485800

The failure to comply with Technical Specification 3.7.10 was a performance deficiency. The finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance that the control room physical design would protect operators from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the for the control room. This finding had a crosscutting aspect in the area human performance associated with decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the control room envelope boundary,

Inspection Report# : 2012005 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : September 03, 2013

# Diablo Canyon 2
# 3Q/2013 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jun 30, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Implement the Fire Protection Program Requirements for the Control of Transient Combustible Material**
Green. The inspectors identified a Green non-cited violation of the licensee's approved fire protection program as defined in Diablo Canyon Facility Operating License Conditions 2.C(5) for Unit 1 and 2.C(4) for Unit 2 involving the failure to effectively implement the fire protection program. Specifically, the inspectors identified multiple examples where the licensee failed to maintain control and tracking of combustible materials, welding equipment, and oxygen/acetylene rigs in the plant. The licensee entered the condition into the corrective action program as Notifications 50510062, 50511864, 50561959, and 50537650.
The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Initiating Events Cornerstone and it adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors concluded that the finding was of very low safety significance (Green) because each deficiency was rated as "Low" degradation because for the violations of the hot work permitting program, all normally required fire prevention measures remained in place and for the violations of the transient combustibles control program, the materials involved did not significantly increase the fire frequency. This finding had a cross-cutting aspect in the area of human performance associated with the work practices component, because the cause of the performance deficiency involved the licensee not ensuring supervisory and management oversight of work activities, such that nuclear safety was supported.
Inspection Report# : 2013003 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Provide Adequate Guidance To Address General Welding Standard Requirements**
On February 14, 2013, the inspectors observed field welders add a partial circumferential weld on one side of the pipe in efforts to repair the pipe misalignment prior to the completion of the final visual inspection. This action represents a violation of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," because the licensee's procedure established special controls for critical distortions but failed to adequately define what situations fit that category. The licensee reviewed the stress calculation for the piping in question and concluded that the addition of the weld filler material did not affect the fatigue resistance of the weld, but acknowledged that a definition and additional guidance for the term "critical" was missing in the procedure and could have adverse effects on future final welds. The licensee entered the finding into their corrective action program as Notification 50542347.

The inspectors determined that the failure of the site's welding standard to provide adequate guidance to identify what constitutes a weld distortion during welding activities is a performance deficiency. The finding is more than minor because if left uncorrected, it has the potential to lead to a more significant safety concern. Specifically, Procedure GSW ASME did not provide the necessary guidance for welders and quality assurance personnel to identify and understand what constitutes critical distortion of a weld. The welding process can introduce effects of weld shrinkage (stresses) and distortion that could adversely affect the final condition of the weld, potentially leading to a service induced failure. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined the finding had a cross cutting aspect in the human performance area associated with work practices, procedural compliance, because the licensee did not adequately define or train welders to know what constituted a critical distortion, and did not effectively communicate the expectation of questioning the procedure if the welding activity required skill of the craft. [H.4(b)]

Inspection Report# : 2013002 *(pdf)*

**Significance:** Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Identify Existing Indications During Prior Ultrasonic Examinations Of Pressurizer Structural Weld Overlays**
The inspectors identified a Green non-cited violation of 10 CFR 50.55a(a)(3)(i), which requires that proposed alternatives to industry codes and standards provide an acceptable level of quality and safety. The NRC staff approved relief request REP 1 U2 dated March 28, 2007, for installing six structural weld overlays on the pressurizer safety, relief, spray and surge nozzles. The request established acceptance criteria of laminar flaws during weld acceptance examinations limited to only the third 10 year inservice inspection interval. Contrary to the above, the licensee failed to identify unacceptable flaws as defined by the approved request following completion of these welds in 2008. The licensee entered the finding into their corrective action program as Notification 50540188.

The inspectors determined that the licensee's failure to identify indications that exceeded the acceptable linear dimension of laminar flaws prior to placing the system in service is a performance deficiency. The performance deficiency is more than minor because it is associated with the initiating events cornerstone attribute of equipment performance, and adversely affects the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, during the months of February and March 2013, the licensee identified that three out of the six pressurizer structural weld overlays exhibited laminar flaws that exceeded the linear dimensions approved by the safety evaluation. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. This issue did not have a cross-cutting aspect associated with it because it is not indicative of current performance.
Inspection Report# : 2013002 *(pdf)*

**Significance:** Mar 23, 2013
Identified By: Self-Revealing
Item Type: FIN Finding
**Failure to Effectively Evaluate Design Change for High Voltage Bushing**
The inspectors reviewed a self-revealing finding for failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications. Specifically, design engineering staff was not effective in using applicable station design documents, in conjunction with industry standards to determine minimum

creepage distance for high voltage insulators when replacing ceramic bushings with polymer bushings on the main bank transformer. As a result, the licensee approved installation of an insulator stack that did not provide adequate ground protection, which caused a plant trip on October 11, 2012. The licensee entered the condition in their corrective action program as Notification 50518473.

Failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 1, “Initiating Events Screening Questions,” this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. This finding had a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions in decision making when considering the suitability of the design for the environment [H.1(b)].
Inspection Report# : 2013002 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Update Emergency Operating Procedures**
The inspectors identified a self-revealing non-cited violation of Technical Specification 5.4.1(b) for failure to maintain emergency operating procedures after personnel reviewing a temporary modification failed to identify and change affected emergency operating procedures. Specifically, the emergency operating procedure EOP E-0.1, “Reactor Trip Response,” Revision 28, was not updated to be consistent with a temporary modification of steam generator water level low-low bistable setpoints. The licensee entered the condition into the corrective action program as Notifications 50517883, 50520697, and 50518355.

The failure to update emergency operating procedure E-0.1 “Reactor Trip Response,” Revision 28, to account for higher low-low water level bistable reset setpoints introduced by Temporary Modification 60044709 was a performance deficiency. The finding was more than minor because it was associated with the procedure quality attribute of the Initiating Events cornerstone. Using Inspection Manual Chapter 0609.04, “Phase 1 – Initial Screening and Characterization of Findings,” this finding was determined to be of very low safety significance (Green) because the finding does not represent a loss of system and/or function and does not represent an actual loss of function of at least a single train for greater than its Technical Specification allowed outage time, or two separate safety systems out-of-service for greater than its Technical Specification allowed outage time. This finding had a crosscutting aspect in the area of human performance, associated with the resources component, because the licensee did not ensure complete, accurate and up-to-date procedures were available and adequate to ensure nuclear safety
Inspection Report# : 2012005 *(pdf)*

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# Mitigating Systems

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation

**Failure to Evaluate the Effects on the Emergency Diesel Generator Load Capability for Maximum Combustion Air Temperature Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Specifically, as of July 11, 2013, the licensee failed to evaluate the impact of the site combustion air temperature and the vendor specified diesel generator rating for combustion air temperature in the emergency diesel generator loading analysis. In addition, the licensee failed to evaluate the available combustion air temperature for the maximum site outside air conditions could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notifications DN-50573049 and DN-50570764

The failure to properly evaluate the vendor stated effects of combustion air temperature on the diesel generator capability and to determine and evaluate the expected maximum value for diesel generator combustion air temperature, based on site-specific conditions, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using actual data, the licensee found that derating of 1.5 percent was necessary under limiting air temperature conditions. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding had a problem identification and resolution cross-cutting aspect associated with thoroughly evaluating problems such that the resolution addresses cause and extent of condition.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Auxiliary Feedwater Pump Motor Capability for the Effects of Pump Maximum Breakhorsepower Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Specifically, as of July 11, 2013, the licensee failed to evaluate the effects of pump load on the auxiliary feedwater pump motor for the design basis maximum flow conditions that could occur during a postulated steam line break coincident with maximum diesel generator frequency which could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notification DN-50572850.

The failure to evaluate the capability of auxiliary feedwater pump motors for the design basis accident maximum pump brake horsepower condition coincident with the maximum diesel generator frequency, which could result in a motor overload, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, there was no analysis or test that demonstrated the motors would be capable of operating for the required mission time during a high energy line break, which resulted in maximum pump brake horsepower conditions that could occur coincident with maximum diesel engine frequency. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding did not have a cross-cutting aspect because the most significant contributor did

not reflect current licensee performance.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Procedures for Establishing Temporary Ventilation**
The team identified a Green non-cited violation associated with Technical Specification 5.4.1(a), "Procedures," which requires that written procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, "Quality Assurance Program," Appendix A, Section 5, requires procedures for Abnormal, Offnormal, or Alarm Conditions. Specifically, as of July 11, 2013, Procedure CP M-10, "Fire Protection of Safe Shutdown Equipment," Revision 27, Attachment 7.8, "Temporary Ventilation for the Control Room, Inverter/Charger Rooms, and 480V Vital Switchgear Rooms and Charging Pump 1-3 Room," Section 4a, requires the use of two 24-inch diameter fans, which, if connected as directed, would not perform the function as prescribed by the procedure as the fans require more current than can be supplied from either the equipment room receptacles or from the alternate power source (the temporary generator and distribution panel). This finding was entered into the corrective action program as Notifications DN-50570838 and DN-50572295.

The failure to provide an adequate procedure for establishing temporary ventilation was a performance deficiency. The finding was more than minor because it affected the equipment performance attribute associated with the Mitigating Systems Cornerstone as related to the availability, reliability, and capability of the 480V Vital Switchgear Rooms. The team reviewed this finding using Inspection Manual Chapter 0609 Attachment 0609.04; 0609 Appendix A, Exhibit 2; and Inspection Manual 0609 Appendix A, Exhibit 4, because it affected the External Event Mitigation Systems (Seismic/Fire/Flood/Severe Weather Protection Degraded) while the plant was at power and involved the loss or degradation of equipment specifically designed to mitigate an external initiating event such as a fire. Inspection Manual Chapter 0609 Appendix A, Exhibit 4, led to a Detailed Risk Evaluation because the finding would degrade two or more trains of a multi-train system or function and would degrade one or more trains of a system that supports a risk significant system or function. The bounding change to the core damage frequency was 4E-7/year (Green). The finding was not a significant contributor to the large early release frequency. The most dominant sequences included fires in Fire Area 34, failure of the 480 Vac switchgear cooling, and the failure of the manual action to restore cooling. The low frequency of applicable fires combined with the relatively low failure probability for the alternate cooling helped to reduce the risk. This finding had a human performance cross-cutting aspect associated with resources, because the licensee did not have adequate procedures and available facilities and equipment, including physical improvements, simulator fidelity and emergency facilities and equipment.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Dec 20, 2012
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Failure to Maintain Required Firewater System Configuration**
The team reviewed a self-revealing non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, "Fire Protection Program," due to the licensee inadvertently isolating the firewater yard loop for approximately three days, reducing the plant's fire protection capability without compensatory actions. The licensee entered this issue in their corrective action program as Notification 50513006.

The failure to maintain the fire water system configuration as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The performance deficiency affected the fire protection defense-in depth strategies involving post-fire safe shutdown systems. The major fire loading in the yard area resulted from the 12 large transformers. The senior reactor analyst made the bounding assumption that any transformer fire without suppression would result in an unrecoverable loss of offsite power. A bounding value was calculated by multiplying the fire ignition frequency by the conditional core damage probability. This resulted in a change to core damage frequency of 1.2 x 10-7. Therefore, the subject finding was of very low safety significance (Green).

This performance deficiency had a cross-cutting aspect in the area of resources associated with providing complete, accurate and up-to-date design documentation, procedures, and work packages, and correct labeling of components. Specifically, the licensee did not provide sufficient details in procedures for operators to successfully align an infrequently operated valve with no position indication. [H.2(c)]
Inspection Report# : 2012008 *(pdf)*

**Significance:** G Dec 20, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Compensatory Measures for Fire Protection Program Deficiencies**
The team identified a non-cited violation of License Conditions 2.C(4) for Unit 1 and 2.C(5) for Unit 2, "Fire Protection Program," due to the licensee's failure to establish or adequately implement compensatory measures for non-compliances with the licensee's approved fire protection program. These non-compliances were identified during the licensee's ongoing transition to a new fire protection program in compliance with National Fire Protection Association Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," (NFPA 805). The licensee entered this issue in their corrective action program as Notifications 50521360 and 50531363.

The failure to establish or maintain appropriate compensatory measures for identified deficiencies in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. A senior reactor analyst evaluated the significance of this performance deficiency.

A fire that results in the loss of switchgear room ventilation would cause a loss of all ac and dc power if operators did not take action to recover cooling. The analyst determined that the licensed operators would have at least two clear annunciators indicating that ventilation had been lost and that room temperatures were increasing. Additionally, Procedure CP-M10, "Fire Protection of Safe Shutdown Equipment", was available to assist in providing portable fan cooling to the rooms.

For a fire to result in an intersystem loss of coolant accident, it would have to cause a 3 phase hot short on both of two shutdown cooling suction valves. Given that each valve is on a different electrical train, the analyst determined that the conditional probabilities of the hot shorts involved would best be modeled as independent. Accounting for the risk associated with both issues evaluated, the analyst estimated the change to core damage probability to be 1.5 x 10-7 per unit. Therefore, the performance deficiency was considered to be of very low safety significance (Green).

This finding did not have a cross-cutting aspect because it was not indicative of the licensee's present performance.
Inspection Report# : 2012008 *(pdf)*

# Barrier Integrity

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Operability Evaluation**
The inspectors identified a non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” after personnel failed to adequately assess the impact of an unanalyzed condition on control room envelope operability. Specifically, personnel performed a problem screening for a nonconforming condition that impacted operability of the control room ventilation system operability and determined that a review by the Shift Foreman, work control Shift Foreman, or Shift Manager was not required. The licensee entered the condition into the corrective action program as Notification 50497774.

The failure to adequately assess the impact of an unanalyzed, non-conservative condition on control room habitability system operability was a performance deficiency. This finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance for the control room physical design to protect from radionuclide releases caused by accidents or events. Using the Inspection Manual Chapter 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room. This finding had a cross-cutting aspect in the area of problem identification and resolution, associated with the corrective action program component, because the licensee did not thoroughly evaluate the impact of non-conservative control room atmospheric dispersion factor methodology on control room habitability system operability,

Inspection Report# : 2012005 *(pdf)*

**Significance:** G Dec 31, 2012
Identified By: NRC
Item Type: NCV NonCited Violation
**Non-conservative Decision Making Resulted in a Violation of Technical Specification**
The inspectors identified a non-cited violation of Technical Specification 3.7.10, "Control Room Ventilation System (CRVS)," after the control room envelope boundary for both units was inoperable for a greater duration than permitted by the out-of-service time. Specifically, the licensee operated Units 1 and 2 without an operable control room envelope from between at least September 2011 and December 2012, which is greater than the 90 day allowed outage time. The licensee entered the condition into the corrective action program as Notifications 50483820, 50497328, and 50485800

The failure to comply with Technical Specification 3.7.10 was a performance deficiency. The finding was more than minor because it was associated with the Barrier Integrity Cornerstone objective design control attribute to provide reasonable assurance that the control room physical design would protect operators from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors concluded that the finding was of very low safety significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the for the control room. This finding had a crosscutting aspect in the area human performance associated with decision-making component because the licensee did not use conservative assumptions in their decision to implement compensatory actions following the inoperability of the control room envelope boundary,

Inspection Report# : 2012005 *(pdf)*

## Emergency Preparedness

## Occupational Radiation Safety

## Public Radiation Safety

## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

## Miscellaneous

Last modified : December 03, 2013

# Diablo Canyon 2
# 4Q/2013 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jul 10, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Reactor Trip due to a Lightning Arrester Flashover**
The inspectors reviewed a Green self revealing non cited violation of 10 CFR 50.65(a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” for failure to implement adequate oversight controls and risk assessment while performing 500kV transmission line insulator maintenance on Unit 2. This caused an initiating event due to a flashover on the main transformer lightning arrester that resulted in a reactor trip.

The failure to effectively perform a risk assessment and properly control maintenance activities that resulted in a reactor trip was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 1, “Initiating Events Screening Questions,” this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. Additionally, using Inspection Manual Chapter 0612, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” this finding was determined to be of very low safety significance (Green). The licensee entered the condition into the corrective action program as Notification 50572800.

This finding had a cross-cutting aspect in the area of human performance, associated with the decision-making component, because the licensee did not demonstrate that nuclear safety was an overriding priority during this maintenance activity. Specifically, the licensee did not initially use conservative decision making in not properly categorizing the activity as a reactor trip risk (despite internal and external operating experience to the contrary), and again when the licensee did not terminate the hot washing activities when environmental conditions degraded resulting in excessive water dispersion [H.1(b)].

Inspection Report# : 2013005 *(pdf)*

**Significance:** G Jun 30, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Implement the Fire Protection Program Requirements for the Control of Transient Combustible Material**
Green. The inspectors identified a Green non-cited violation of the licensee’s approved fire protection program as defined in Diablo Canyon Facility Operating License Conditions 2.C(5) for Unit 1 and 2.C(4) for Unit 2 involving the failure to effectively implement the fire protection program. Specifically, the inspectors identified multiple examples where the licensee failed to maintain control and tracking of combustible materials, welding equipment, and oxygen/acetylene rigs in the plant. The licensee entered the condition into the corrective action program as

Notifications 50510062, 50511864, 50561959, and 50537650.
The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Initiating Events Cornerstone and it adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions. Using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," the inspectors concluded that the finding was of very low safety significance (Green) because each deficiency was rated as "Low" degradation because for the violations of the hot work permitting program, all normally required fire prevention measures remained in place and for the violations of the transient combustibles control program, the materials involved did not significantly increase the fire frequency. This finding had a cross-cutting aspect in the area of human performance associated with the work practices component, because the cause of the performance deficiency involved the licensee not ensuring supervisory and management oversight of work activities, such that nuclear safety was supported.
Inspection Report# : 2013003 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Provide Adequate Guidance To Address General Welding Standard Requirements**
On February 14, 2013, the inspectors observed field welders add a partial circumferential weld on one side of the pipe in efforts to repair the pipe misalignment prior to the completion of the final visual inspection. This action represents a violation of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," because the licensee's procedure established special controls for critical distortions but failed to adequately define what situations fit that category. The licensee reviewed the stress calculation for the piping in question and concluded that the addition of the weld filler material did not affect the fatigue resistance of the weld, but acknowledged that a definition and additional guidance for the term "critical" was missing in the procedure and could have adverse effects on future final welds. The licensee entered the finding into their corrective action program as Notification 50542347.

The inspectors determined that the failure of the site's welding standard to provide adequate guidance to identify what constitutes a weld distortion during welding activities is a performance deficiency. The finding is more than minor because if left uncorrected, it has the potential to lead to a more significant safety concern. Specifically, Procedure GSW ASME did not provide the necessary guidance for welders and quality assurance personnel to identify and understand what constitutes critical distortion of a weld. The welding process can introduce effects of weld shrinkage (stresses) and distortion that could adversely affect the final condition of the weld, potentially leading to a service induced failure. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. The inspectors determined the finding had a cross cutting aspect in the human performance area associated with work practices, procedural compliance, because the licensee did not adequately define or train welders to know what constituted a critical distortion, and did not effectively communicate the expectation of questioning the procedure if the welding activity required skill of the craft. [H.4(b)]

Inspection Report# : 2013002 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure To Identify Existing Indications During Prior Ultrasonic Examinations Of Pressurizer Structural Weld**

**Overlays**
The inspectors identified a Green non-cited violation of 10 CFR 50.55a(a)(3)(i), which requires that proposed alternatives to industry codes and standards provide an acceptable level of quality and safety. The NRC staff approved relief request REP 1 U2 dated March 28, 2007, for installing six structural weld overlays on the pressurizer safety, relief, spray and surge nozzles. The request established acceptance criteria of laminar flaws during weld acceptance examinations limited to only the third 10 year inservice inspection interval. Contrary to the above, the licensee failed to identify unacceptable flaws as defined by the approved request following completion of these welds in 2008. The licensee entered the finding into their corrective action program as Notification 50540188.

The inspectors determined that the licensee's failure to identify indications that exceeded the acceptable linear dimension of laminar flaws prior to placing the system in service is a performance deficiency. The performance deficiency is more than minor because it is associated with the initiating events cornerstone attribute of equipment performance, and adversely affects the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, during the months of February and March 2013, the licensee identified that three out of the six pressurizer structural weld overlays exhibited laminar flaws that exceeded the linear dimensions approved by the safety evaluation. Using Manual Chapter 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At Power," the finding was determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss-of-coolant accident and did not affect other systems used to mitigate a loss-of-coolant accident resulting in a total loss of their function. This issue did not have a cross-cutting aspect associated with it because it is not indicative of current performance.
Inspection Report# : 2013002 *(pdf)*

**Significance:** G Mar 23, 2013
Identified By: Self-Revealing
Item Type: FIN Finding
**Failure to Effectively Evaluate Design Change for High Voltage Bushing**
The inspectors reviewed a self-revealing finding for failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications. Specifically, design engineering staff was not effective in using applicable station design documents, in conjunction with industry standards to determine minimum creepage distance for high voltage insulators when replacing ceramic bushings with polymer bushings on the main bank transformer. As a result, the licensee approved installation of an insulator stack that did not provide adequate ground protection, which caused a plant trip on October 11, 2012. The licensee entered the condition in their corrective action program as Notification 50518473.

Failure to effectively and accurately evaluate all available resources to procure appropriate equipment for plant modifications was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 1, "Initiating Events Screening Questions," this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. This finding had a cross-cutting aspect in the area of human performance, associated with the decision making component, because the licensee did not use conservative assumptions in decision making when considering the suitability of the design for the environment [H.1(b)].
Inspection Report# : 2013002 *(pdf)*

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# Mitigating Systems

**Significance:** G Sep 20, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Valid EDG 2-1 Start Signal Caused by a Loss of 4 kV Class 1E Bus G**
The inspectors reviewed a self-revealing non-cited violation 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” associated with troubleshooting of the Unit 2, 4kV bus G that resulted in an unplanned de-energization. This caused an unplanned entry into a 72 hour shutdown technical specification due to diesel fuel oil transfer pump 0 2 becoming unavailable. The licensee entered the condition into the corrective action program as Notification 50544198.

The failure to plan and coordinate emergent maintenance such that it would not impact other mitigating systems was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. This finding was evaluated for each unit separately. For Unit 1, which was at power, using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. For Unit 2 this finding did not require evaluation using Inspection Manual Chapter 0609, and Appendix G because the unit was defueled. The finding had a cross-cutting aspect in the area of human performance, work practices component, because workers failed to use multiple human error prevention techniques.
Inspection Report# : 2013004 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Effects on the Emergency Diesel Generator Load Capability for Maximum Combustion Air Temperature Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Specifically, as of July 11, 2013, the licensee failed to evaluate the impact of the site combustion air temperature and the vendor specified diesel generator rating for combustion air temperature in the emergency diesel generator loading analysis. In addition, the licensee failed to evaluate the available combustion air temperature for the maximum site outside air conditions could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notifications DN-50573049 and DN-50570764

The failure to properly evaluate the vendor stated effects of combustion air temperature on the diesel generator capability and to determine and evaluate the expected maximum value for diesel generator combustion air temperature, based on site-specific conditions, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using actual data, the licensee found that derating of 1.5 percent was necessary under limiting air temperature conditions. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because

the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding had a problem identification and resolution cross-cutting aspect associated with thoroughly evaluating problems such that the resolution addresses cause and extent of condition.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Auxiliary Feedwater Pump Motor Capability for the Effects of Pump Maximum Breakhorsepower Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions." Specifically, as of July 11, 2013, the licensee failed to evaluate the effects of pump load on the auxiliary feedwater pump motor for the design basis maximum flow conditions that could occur during a postulated steam line break coincident with maximum diesel generator frequency which could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notification DN-50572850.

The failure to evaluate the capability of auxiliary feedwater pump motors for the design basis accident maximum pump brake horsepower condition coincident with the maximum diesel generator frequency, which could result in a motor overload, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, there was no analysis or test that demonstrated the motors would be capable of operating for the required mission time during a high energy line break, which resulted in maximum pump brake horsepower conditions that could occur coincident with maximum diesel engine frequency. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Procedures for Establishing Temporary Ventilation**
The team identified a Green non-cited violation associated with Technical Specification 5.4.1(a), "Procedures," which requires that written procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, "Quality Assurance Program," Appendix A, Section 5, requires procedures for Abnormal, Offnormal, or Alarm Conditions. Specifically, as of July 11, 2013, Procedure CP M-10, "Fire Protection of Safe Shutdown Equipment," Revision 27, Attachment 7.8, "Temporary Ventilation for the Control Room, Inverter/Charger Rooms, and 480V Vital Switchgear Rooms and Charging Pump 1-3 Room," Section 4a, requires the use of two 24-inch diameter fans, which, if connected as directed, would not perform the function as prescribed by the procedure as the fans require more current than can be supplied from either the equipment room receptacles or from the alternate power source (the temporary generator and distribution panel). This finding was entered into the corrective action program as Notifications DN-50570838 and DN-50572295.

The failure to provide an adequate procedure for establishing temporary ventilation was a performance deficiency. The finding was more than minor because it affected the equipment performance attribute associated with the Mitigating Systems Cornerstone as related to the availability, reliability, and capability of the 480V Vital Switchgear Rooms. The team reviewed this finding using Inspection Manual Chapter 0609 Attachment 0609.04; 0609 Appendix A, Exhibit 2; and Inspection Manual 0609 Appendix A, Exhibit 4, because it affected the External Event Mitigation Systems (Seismic/Fire/Flood/Severe Weather Protection Degraded) while the plant was at power and involved the loss or degradation of equipment specifically designed to mitigate an external initiating event such as a fire. Inspection Manual Chapter 0609 Appendix A, Exhibit 4, led to a Detailed Risk Evaluation because the finding would degrade two or more trains of a multi-train system or function and would degrade one or more trains of a system that supports a risk significant system or function. The bounding change to the core damage frequency was 4E-7/year (Green). The finding was not a significant contributor to the large early release frequency. The most dominant sequences included fires in Fire Area 34, failure of the 480 Vac switchgear cooling, and the failure of the manual action to restore cooling. The low frequency of applicable fires combined with the relatively low failure probability for the alternate cooling helped to reduce the risk. This finding had a human performance cross-cutting aspect associated with resources, because the licensee did not have adequate procedures and available facilities and equipment, including physical improvements, simulator fidelity and emergency facilities and equipment.
Inspection Report# : 2013007 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Loss of Control Room Ventilation System due to Inadequate Design Control**
The inspectors reviewed a Green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after the licensee performed a design change to the control room ventilation system (CRVS) that resulted in none of the four CRVS pressurization fans being able to continuously operate if they started in response to a Phase A containment isolation or control room radiation atmosphere intake actuation signal. This resulted in declaring the Units 1 and 2 CRVS actuation instrumentation and CRVS inoperable and an unplanned entry into Technical Specifications (TS) 3.3.7, "Control Room Ventilation System Actuation Instrumentation," and TS 3.7.10, "Control Room Ventilation System," respectively.

The failure to use proper design control during the CRVS modification was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone, and it adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radiological releases caused by accidents or events, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 3, “Barrier Integrity Screening Questions,” this finding was determined to be of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The licensee entered the condition into the corrective action program as Notification 50525605.

The finding had a cross cutting aspect in the area of human performance resources component because licensee staff did not maintain complete, accurate, and up to date design documentation – specifically, because the functions of the pressure switches and CRVS interlocks had never been adequately described in design control documents [H.2(c)].
Inspection Report# : 2013005 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : February 24, 2014

# Diablo Canyon 2
# 1Q/2014 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jul 10, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Reactor Trip due to a Lightning Arrester Flashover**
The inspectors reviewed a Green self revealing non cited violation of 10 CFR 50.65(a)(4), “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” for failure to implement adequate oversight controls and risk assessment while performing 500kV transmission line insulator maintenance on Unit 2. This caused an initiating event due to a flashover on the main transformer lightning arrester that resulted in a reactor trip.

The failure to effectively perform a risk assessment and properly control maintenance activities that resulted in a reactor trip was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 1, “Initiating Events Screening Questions,” this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. Additionally, using Inspection Manual Chapter 0612, Appendix K, “Maintenance Risk Assessment and Risk Management Significance Determination Process,” this finding was determined to be of very low safety significance (Green). The licensee entered the condition into the corrective action program as Notification 50572800.

This finding had a cross-cutting aspect in the area of human performance, associated with the decision-making component, because the licensee did not demonstrate that nuclear safety was an overriding priority during this maintenance activity. Specifically, the licensee did not initially use conservative decision making in not properly categorizing the activity as a reactor trip risk (despite internal and external operating experience to the contrary), and again when the licensee did not terminate the hot washing activities when environmental conditions degraded resulting in excessive water dispersion [H.1(b)].

Inspection Report# : 2013005 *(pdf)*

**Significance:** G Jun 30, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Implement the Fire Protection Program Requirements for the Control of Transient Combustible Material**
Green. The inspectors identified a Green non-cited violation of the licensee’s approved fire protection program as defined in Diablo Canyon Facility Operating License Conditions 2.C(5) for Unit 1 and 2.C(4) for Unit 2 involving the failure to effectively implement the fire protection program. Specifically, the inspectors identified multiple examples where the licensee failed to maintain control and tracking of combustible materials, welding equipment, and oxygen/acetylene rigs in the plant. The licensee entered the condition into the corrective action program as

Notifications 50510062, 50511864, 50561959, and 50537650.
The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Initiating Events Cornerstone and it adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions. Using Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process,” the inspectors concluded that the finding was of very low safety significance (Green) because each deficiency was rated as “Low” degradation because for the violations of the hot work permitting program, all normally required fire prevention measures remained in place and for the violations of the transient combustibles control program, the materials involved did not significantly increase the fire frequency. This finding had a cross-cutting aspect in the area of human performance associated with the work practices component, because the cause of the performance deficiency involved the licensee not ensuring supervisory and management oversight of work activities, such that nuclear safety was supported.
Inspection Report# : 2013003 *(pdf)*

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# Mitigating Systems

**Significance:** Mar 21, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Results in Inadequate Operability Assessment**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to follow the operability assessment procedure in considering the tornado atmospheric effects and tornado missile impactive force effects on the emergency diesel generator radiator ventilation plenum and engine exhaust pipes. The licensee took immediate corrective actions to remove potential tornado missiles that may affect the operability of the emergency diesel generators.
The licensee’s failure to account for tornado atmospheric pressure change effects and tornado-generated missile impactive loads is a performance deficiency. Specifically, the operability assessment did not account for the pressure change or impactive loads as described by the Standard Review Plan methodology. This performance deficiency was more than minor because it is associated with the protection against external factors attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process (SDP) For Findings At-Power”, dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk analysis determined that the calculated tornado missile strike frequency at Diablo Canyon is lower than the 1 x 10-6 threshold in the significance determination process, and therefore, the finding was determined to be of very low safety significance (Green).
This finding has a problem identification and resolution cross-cutting aspect associated with evaluation; specifically in that the licensee did not thoroughly evaluate the problem to ensure that resolutions addressed the cause(s) and extent of conditions, commensurate with their safety significance [P.2].
Inspection Report# : 2014002 *(pdf)*

**Significance:** Sep 20, 2013
Identified By: NRC
Item Type: NCV NonCited Violation

**Valid EDG 2-1 Start Signal Caused by a Loss of 4 kV Class 1E Bus G**
The inspectors reviewed a self-revealing non-cited violation 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” associated with troubleshooting of the Unit 2, 4kV bus G that resulted in an unplanned de-energization. This caused an unplanned entry into a 72 hour shutdown technical specification due to diesel fuel oil transfer pump 0 2 becoming unavailable. The licensee entered the condition into the corrective action program as Notification 50544198.

The failure to plan and coordinate emergent maintenance such that it would not impact other mitigating systems was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. This finding was evaluated for each unit separately. For Unit 1, which was at power, using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. For Unit 2 this finding did not require evaluation using Inspection Manual Chapter 0609, and Appendix G because the unit was defueled. The finding had a cross-cutting aspect in the area of human performance, work practices component, because workers failed to use multiple human error prevention techniques.
Inspection Report# : 2013004 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Effects on the Emergency Diesel Generator Load Capability for Maximum Combustion Air Temperature Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Specifically, as of July 11, 2013, the licensee failed to evaluate the impact of the site combustion air temperature and the vendor specified diesel generator rating for combustion air temperature in the emergency diesel generator loading analysis. In addition, the licensee failed to evaluate the available combustion air temperature for the maximum site outside air conditions could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notifications DN-50573049 and DN-50570764

The failure to properly evaluate the vendor stated effects of combustion air temperature on the diesel generator capability and to determine and evaluate the expected maximum value for diesel generator combustion air temperature, based on site-specific conditions, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using actual data, the licensee found that derating of 1.5 percent was necessary under limiting air temperature conditions. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding had a problem identification and resolution cross-cutting aspect associated with thoroughly evaluating problems such that the resolution addresses cause and extent of condition.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Auxiliary Feedwater Pump Motor Capability for the Effects of Pump Maximum Breakhorsepower Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions.” Specifically, as of July 11, 2013, the licensee failed to evaluate the effects of pump load on the auxiliary feedwater pump motor for the design basis maximum flow conditions that could occur during a postulated steam line break coincident with maximum diesel generator frequency which could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notification DN-50572850.

The failure to evaluate the capability of auxiliary feedwater pump motors for the design basis accident maximum pump brake horsepower condition coincident with the maximum diesel generator frequency, which could result in a motor overload, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, there was no analysis or test that demonstrated the motors would be capable of operating for the required mission time during a high energy line break, which resulted in maximum pump brake horsepower conditions that could occur coincident with maximum diesel engine frequency. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Procedures for Establishing Temporary Ventilation**
The team identified a Green non-cited violation associated with Technical Specification 5.4.1(a), “Procedures,” which requires that written procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, “Quality Assurance Program,” Appendix A, Section 5, requires procedures for Abnormal, Offnormal, or Alarm Conditions. Specifically, as of July 11, 2013, Procedure CP M-10, “Fire Protection of Safe Shutdown Equipment,” Revision 27, Attachment 7.8, “Temporary Ventilation for the Control Room, Inverter/Charger Rooms, and 480V Vital Switchgear Rooms and Charging Pump 1-3 Room,” Section 4a, requires the use of two 24-inch diameter fans, which, if connected as directed, would not perform the function as prescribed by the procedure as the fans require more current than can be supplied from either the equipment room receptacles or from the alternate power source (the temporary generator and distribution panel). This finding was entered into the corrective action program as Notifications DN-50570838 and DN-50572295.

The failure to provide an adequate procedure for establishing temporary ventilation was a performance deficiency. The finding was more than minor because it affected the equipment performance attribute associated with the Mitigating Systems Cornerstone as related to the availability, reliability, and capability of the 480V Vital Switchgear Rooms. The team reviewed this finding using Inspection Manual Chapter 0609 Attachment 0609.04; 0609 Appendix A, Exhibit 2; and Inspection Manual 0609 Appendix A, Exhibit 4, because it affected the External Event Mitigation Systems (Seismic/Fire/Flood/Severe Weather Protection Degraded) while the plant was at power and involved the loss

or degradation of equipment specifically designed to mitigate an external initiating event such as a fire. Inspection Manual Chapter 0609 Appendix A, Exhibit 4, led to a Detailed Risk Evaluation because the finding would degrade two or more trains of a multi-train system or function and would degrade one or more trains of a system that supports a risk significant system or function. The bounding change to the core damage frequency was 4E-7/year (Green). The finding was not a significant contributor to the large early release frequency. The most dominant sequences included fires in Fire Area 34, failure of the 480 Vac switchgear cooling, and the failure of the manual action to restore cooling. The low frequency of applicable fires combined with the relatively low failure probability for the alternate cooling helped to reduce the risk. This finding had a human performance cross-cutting aspect associated with resources, because the licensee did not have adequate procedures and available facilities and equipment, including physical improvements, simulator fidelity and emergency facilities and equipment.
Inspection Report# : 2013007 *(pdf)*

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## Barrier Integrity

**Significance:** G Dec 31, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Loss of Control Room Ventilation System due to Inadequate Design Control**
The inspectors reviewed a Green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after the licensee performed a design change to the control room ventilation system (CRVS) that resulted in none of the four CRVS pressurization fans being able to continuously operate if they started in response to a Phase A containment isolation or control room radiation atmosphere intake actuation signal. This resulted in declaring the Units 1 and 2 CRVS actuation instrumentation and CRVS inoperable and an unplanned entry into Technical Specifications (TS) 3.3.7, "Control Room Ventilation System Actuation Instrumentation," and TS 3.7.10, "Control Room Ventilation System," respectively.

The failure to use proper design control during the CRVS modification was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone, and it adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radiological releases caused by accidents or events, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 3, “Barrier Integrity Screening Questions,” this finding was determined to be of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The licensee entered the condition into the corrective action program as Notification 50525605.

The finding had a cross cutting aspect in the area of human performance resources component because licensee staff did not maintain complete, accurate, and up to date design documentation – specifically, because the functions of the pressure switches and CRVS interlocks had never been adequately described in design control documents [H.2(c)].
Inspection Report# : 2013005 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : May 30, 2014

# Diablo Canyon 2
# 2Q/2014 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jul 10, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Reactor Trip due to a Lightning Arrester Flashover**
The inspectors reviewed a Green self revealing non cited violation of 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for failure to implement adequate oversight controls and risk assessment while performing 500kV transmission line insulator maintenance on Unit 2. This caused an initiating event due to a flashover on the main transformer lightning arrester that resulted in a reactor trip.

The failure to effectively perform a risk assessment and properly control maintenance activities that resulted in a reactor trip was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenged critical safety functions during power operations, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 1, "Initiating Events Screening Questions," this finding was determined to be of very low safety significance (Green) because, although it resulted in a reactor trip, it did not result in the loss of mitigating equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. Additionally, using Inspection Manual Chapter 0612, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process," this finding was determined to be of very low safety significance (Green). The licensee entered the condition into the corrective action program as Notification 50572800.

This finding had a cross-cutting aspect in the area of human performance, associated with the decision-making component, because the licensee did not demonstrate that nuclear safety was an overriding priority during this maintenance activity. Specifically, the licensee did not initially use conservative decision making in not properly categorizing the activity as a reactor trip risk (despite internal and external operating experience to the contrary), and again when the licensee did not terminate the hot washing activities when environmental conditions degraded resulting in excessive water dispersion [H.1(b)].

Inspection Report# : 2013005 *(pdf)*

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## Mitigating Systems

**Significance:** G Jun 30, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Associated with Seismic Induced Structural Interactions**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly identify and evaluate system interactions as required by the

licensee's Seismically-Induced Systems Interaction Program (SISIP) Procedure AD4.ID3, "SISIP Housekeeping Activities." Specifically, the inspectors identified multiple instances of components or sources capable of producing a potential threat related to seismic induced structural interactions of safety related equipment or components.

The failure of plant personnel to follow procedure requirements to properly identify and evaluate for impact equipment near sensitive or safety related equipment was a performance deficiency. This performance deficiency was more than minor and is therefore a finding because it was associated with the protection against external factors (seismic) attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, because Diablo Canyon staff did not fix or perform evaluations of seismic induced system interactions on safety-related or accident mitigating systems, this had the potential to challenge the availability, reliability, and capability of various systems required to function following or during earthquakes to prevent undesirable consequence.

Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating System Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding was associated with seismic design or qualification of systems, structures, and components but did not result in the loss of a system operability or functionality.

The inspectors determined this finding has a problem identification and resolution cross cutting aspect associated with the Identification attribute; specifically in that PG&E personnel failed to implement the SISIP with a low enough threshold for identifying and assessing seismic induced system interactions in accordance with the SISI program and procedures [P.1].
Inspection Report# : 2014003 *(pdf)*

**Significance:** G Mar 21, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Results in Inadequate Operability Assessment**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow the operability assessment procedure in considering the tornado atmospheric effects and tornado missile impactive force effects on the emergency diesel generator radiator ventilation plenum and engine exhaust pipes. The licensee took immediate corrective actions to remove potential tornado missiles that may affect the operability of the emergency diesel generators.
The licensee's failure to account for tornado atmospheric pressure change effects and tornado-generated missile impactive loads is a performance deficiency. Specifically, the operability assessment did not account for the pressure change or impactive loads as described by the Standard Review Plan methodology. This performance deficiency was more than minor because it is associated with the protection against external factors attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process (SDP) For Findings At-Power", dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk analysis determined that the calculated tornado missile strike frequency at Diablo Canyon is lower than the 1 x 10-6 threshold in the significance determination process, and therefore, the finding was determined to be of very low safety significance (Green).
This finding has a problem identification and resolution cross-cutting aspect associated with evaluation; specifically in that the licensee did not thoroughly evaluate the problem to ensure that resolutions addressed the cause(s) and extent of conditions, commensurate with their safety significance [P.2].
Inspection Report# : 2014002 *(pdf)*

**Significance:** G Sep 20, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Valid EDG 2-1 Start Signal Caused by a Loss of 4 kV Class 1E Bus G**
The inspectors reviewed a self-revealing non-cited violation 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," associated with troubleshooting of the Unit 2, 4kV bus G that resulted in an unplanned de-energization. This caused an unplanned entry into a 72 hour shutdown technical specification due to diesel fuel oil transfer pump 0 2 becoming unavailable. The licensee entered the condition into the corrective action program as Notification 50544198.

The failure to plan and coordinate emergent maintenance such that it would not impact other mitigating systems was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. This finding was evaluated for each unit separately. For Unit 1, which was at power, using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. For Unit 2 this finding did not require evaluation using Inspection Manual Chapter 0609, and Appendix G because the unit was defueled. The finding had a cross-cutting aspect in the area of human performance, work practices component, because workers failed to use multiple human error prevention techniques [H.4(a)].
Inspection Report# : 2013004 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Effects on the Emergency Diesel Generator Load Capability for Maximum Combustion Air Temperature Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions." Specifically, as of July 11, 2013, the licensee failed to evaluate the impact of the site combustion air temperature and the vendor specified diesel generator rating for combustion air temperature in the emergency diesel generator loading analysis. In addition, the licensee failed to evaluate the available combustion air temperature for the maximum site outside air conditions could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notifications DN-50573049 and DN-50570764.

The team determined that the failure to properly evaluate the vendor stated effects of combustion air temperature on the diesel generator capability and to determine and evaluate the expected maximum value for diesel generator combustion air temperature, based on site-specific conditions, was a performance deficiency. The finding was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, using actual data, the licensee found that derating of 1.5 percent was necessary under limiting air temperature conditions. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding had a problem identification and resolution cross-cutting aspect associated with

thoroughly evaluating problems such that the resolution addresses cause and extent of condition [P.1(c)].
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Evaluate the Auxiliary Feedwater Pump Motor Capability for the Effects of Pump Maximum Breakhorsepower Conditions**
The team identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions." Specifically, as of July 11, 2013, the licensee failed to evaluate the effects of pump load on the auxiliary feedwater pump motor for the design basis maximum flow conditions that could occur during a postulated steam line break coincident with maximum diesel generator frequency which could have affected the capability of safety-related equipment to respond to initiating events. This finding was entered into the corrective action program as Notification DN-50572850.

The team determined that the failure to evaluate the capability of auxiliary feedwater pump motors for the design basis accident maximum pump brake horsepower condition coincident with the maximum diesel generator frequency, which could result in a motor overload, was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, there was no analysis or test that demonstrated the motors would be capable of operating for the required mission time during a high energy line break, which resulted in maximum pump brake horsepower conditions that could occur coincident with maximum diesel engine frequency. Using Inspection Manual Chapter 0609, Significance Determination Process, Appendix A, the finding was determined to have very low safety significance (Green) because the finding was a design or qualification deficiency that did not result in the loss of operability or functionality, did not result in a loss of safety function, and did not screen as potentially risk significant due to external events. This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2013007 *(pdf)*

**Significance:** G Jul 11, 2013
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Procedures for Establishing Temporary Ventilation**
The team identified a Green non-cited violation associated with Technical Specification 5.4.1(a), "Procedures," which requires that written procedures be established, implemented, and maintained covering the applicable procedures in Regulatory Guide 1.33, Revision 2, Appendix A. Regulatory Guide 1.33, "Quality Assurance Program," Appendix A, Section 5, requires procedures for Abnormal, Offnormal, or Alarm Conditions. Specifically, as of July 11, 2013, Procedure CP M-10, "Fire Protection of Safe Shutdown Equipment," Revision 27, Attachment 7.8, "Temporary Ventilation for the Control Room, Inverter/Charger Rooms, and 480V Vital Switchgear Rooms and Charging Pump 1-3 Room," Section 4a, requires the use of two 24-inch diameter fans, which, if connected as directed, would not perform the function as prescribed by the procedure as the fans require more current than can be supplied from either the equipment room receptacles or from the alternate power source (the temporary generator and distribution panel). This finding was entered into the corrective action program as Notifications DN-50570838 and DN-50572295.

The team determined that the failure to provide an adequate procedure for establishing temporary ventilation was a performance deficiency. The finding was more than minor because it affected the equipment performance attribute associated with the Mitigating Systems Cornerstone as related to the availability, reliability, and capability of the

480V Vital Switchgear Rooms. The team reviewed this finding using Inspection Manual Chapter 0609 Attachment 0609.04; 0609 Appendix A, Exhibit 2; and Inspection Manual 0609 Appendix A, Exhibit 4, because it affected the External Event Mitigation Systems (Seismic/Fire/Flood/Severe Weather Protection Degraded) while the plant was at power and involved the loss or degradation of equipment specifically designed to mitigate an external initiating event such as a fire. Inspection Manual Chapter 0609 Appendix A, Exhibit 4, led to a Detailed Risk Evaluation because the finding would degrade two or more trains of a multi-train system or function and would degrade one or more trains of a system that supports a risk significant system or function. The bounding change to the core damage frequency was 4E-7/year (Green). The finding was not a significant contributor to the large early release frequency. The most dominant sequences included fires in Fire Area 34, failure of the 480 Vac switchgear cooling, and the failure of the manual action to restore cooling. The low frequency of applicable fires combined with the relatively low failure probability for the alternate cooling helped to reduce the risk. This finding had a human performance cross-cutting aspect associated with resources, because the licensee did not have adequate procedures and available facilities and equipment, including physical improvements, simulator fidelity and emergency facilities and equipment [H.2(d)]..
Inspection Report# : 2013007 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Loss of Control Room Ventilation System due to Inadequate Design Control**
The inspectors reviewed a Green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after the licensee performed a design change to the control room ventilation system (CRVS) that resulted in none of the four CRVS pressurization fans being able to continuously operate if they started in response to a Phase A containment isolation or control room radiation atmosphere intake actuation signal. This resulted in declaring the Units 1 and 2 CRVS actuation instrumentation and CRVS inoperable and an unplanned entry into Technical Specifications (TS) 3.3.7, "Control Room Ventilation System Actuation Instrumentation," and TS 3.7.10, "Control Room Ventilation System," respectively.

The failure to use proper design control during the CRVS modification was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone, and it adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radiological releases caused by accidents or events, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 3, “Barrier Integrity Screening Questions,” this finding was determined to be of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The licensee entered the condition into the corrective action program as Notification 50525605.

The finding had a cross cutting aspect in the area of human performance resources component because licensee staff did not maintain complete, accurate, and up to date design documentation – specifically, because the functions of the pressure switches and CRVS interlocks had never been adequately described in design control documents [H.2(c)].
Inspection Report# : 2013005 *(pdf)*

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# Emergency Preparedness

## Occupational Radiation Safety

## Public Radiation Safety

## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

## Miscellaneous

Last modified : August 29, 2014

# Diablo Canyon 2
# 3Q/2014 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Degraded Conditions in the Corrective Action Process**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and drawings," in that the licensee did not enter degraded conditions into the corrective action process. The inspectors identified two examples. Specifically, on May 12-13, 2014, the licensee experienced high temperatures in the 480 volt vital bus rooms and did not initiate a notification to document the unexpected condition. Second, on May 20, 2014, the licensee failed to document that a 480 volt vital bus room ventilation system register louvers was found closed.

The failure to enter problems into the corrective action process on the 480 volt busses was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. The inspectors determined this finding has a human performance cross-cutting aspect associated with challenging the unknown attribute, specifically in that licensee personnel did not maintain a questioning attitude to resolve unexpected conditions [H.11]. (Section 1R15)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors reviewed a Green self-revealing, non-cited violation of Technical Specification 5.4.1.a, "Procedures," for failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment. Specifically, inspectors reviewed the licensee performance associated with surveillance and maintenance activities and identified two examples of improper configuration of safety-related equipment returned to service, because of inadequate preplanned maintenance procedures.

The failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment is a performance deficiency. The inspectors determined that the finding was more than minor because it is associated with the procedure quality attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, the restriction of airflow caused by inadvertent closure of ventilation registers following the damper inspection resulted in the undesired consequences of higher ambient 480 volt switchgear room temperatures. In addition, the misconfiguration of the source range N-32 nuclear instrumentation impacted the functioning of the P-6 permissive and prevented it from performing properly during Unit 2 reactor startup such that operator action was necessary to prevent damage to the detector. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of mitigating structures, systems, and components that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with documentation, because the licensee did not ensure plant activities are governed with comprehensive maintenance procedures which are complete, accurate, and up to date to ensure work processes did not affect the performance of safety-related equipment [H.7]. (Section 4OA2.2)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Procedure Results in Unnecessary Main Steam Safety Valve Lift**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee failure to prescribe a procedure appropriate to the circumstances with respect to safety-related atmospheric dump valves and main steam safety valves. Specifically, control of atmospheric steam dump valves was not appropriate for a rapid plant shutdown resulting in unnecessary lifting of a spring-loaded main steam safety valve.

The inspectors determined that the licensee’s failure to ensure appropriate procedures to properly control steam generator pressure and prevent unnecessary lifting of main steam safety valves was a performance deficiency. This performance deficiency was determined to be more than minor because it affected the Mitigating Systems cornerstone attribute of procedural quality and the objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with avoiding complacency, because the licensee failed to recognize during rapid load reductions the inherent risk of lifting a main steam safety valve and did not recognize or plan with adequate procedures, for a condition with a potential latent problem [H.12]. (Section 4OA3.3)

Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Degraded Emergency Diesel Generator Fuel Injector Nozzles in the Corrective Action Program**

The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," in that the licensee did not enter degraded conditions into the corrective action process. The first example of this violation occurred in ten identified instances from 2009 to 2012 when technicians failed to document degraded emergency diesel generator fuel injector nozzles in the corrective action program. The second example occurred in July and August 2014 when engineering personnel failed to appropriately document loose bolts on 4.16kV breaker panels in the corrective action program. The licensee documented this issue in the corrective action program as SAPNs 50641514 and 50656750 and issued a communication to the station reminding personnel of the requirement to initiate notifications even when problems are immediately resolved.

The failure to document unsatisfactory emergency diesel generator fuel injection nozzles and loose 4.16kV switchgear bolts in the corrective action program as required by procedure was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is therefore a finding. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. The inspectors determined this finding has an identification cross-cutting aspect in the problem identification and resolution cross-cutting area because the organization failed to implement a corrective action program with a low threshold for identification (P.1). Specifically, personnel failed to recognize that identified deficiencies were deviations from standards and that degraded conditions were promptly documented in the corrective action program.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Technical Specification Surveillance Requirement for Emergency Diesel Generators**
The team identified a Green non-cited violation of 10 CFR 50.36 for the licensee's failure to establish an appropriate surveillance test to demonstrate operability of its emergency diesel generators. After revising its emergency diesel generator loading analysis, the licensee failed to adjust the parameters for the full-load-reject surveillance to ensure the test was performed with the maximum anticipated electrical loading. After the team identified this violation, the licensee entered Surveillance Requirement 3.0.3 and documented the condition in its corrective action program as SAPNs 50657635 and 50657637.

The licensee's failure to specify the "lowest functional capability or performance level of equipment required for safe operation of the facility" as required by 10 CFR 50.36 was a performance deficiency. This performance deficiency was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a resolution cross-cutting aspect in the problem identification and resolution cross-cutting area because the licensee failed to take effective corrective actions to address the nonconservative surveillance parameters in a timely manner (P.3). Specifically, the licensee did not take appropriate interim corrective actions to mitigate the issue while more fundamental causes were being assessed.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Longstanding Uncompensated Nonconforming Condition**

The team identified a Green non-cited violation of 10 CFR Part 50 Appendix B Criterion XVI for the licensee's failure to take timely corrective actions. In 2011, the licensee identified a potential path for gas intrusion into the containment spray system, contrary to design basis requirements. The licensee took no interim or compensatory actions while planning its final corrective actions. The licensee documented this condition in its corrective action program as SAPN 50657636.

The failure to take timely corrective actions as required by 10 CFR 50 Appendix B Criterion XVI was a performance deficiency. This performance deficiency was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609 Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a conservative bias cross-cutting aspect in the human performance cross-cutting area because licensee personnel failed to use decision-making practices that emphasized prudent choices over those that were simply allowable (H.14). Specifically, licensee managers failed to take timely action to address degraded conditions commensurate with their safety significance.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Jun 30, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Associated with Seismic Induced Structural Interactions**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly identify and evaluate system interactions as required by the licensee's Seismically-Induced Systems Interaction Program (SISIP) Procedure AD4.ID3, "SISIP Housekeeping Activities." Specifically, the inspectors identified multiple instances of components or sources capable of producing a potential threat related to seismic induced structural interactions of safety related equipment or components.

The failure of plant personnel to follow procedure requirements to properly identify and evaluate for impact equipment near sensitive or safety related equipment was a performance deficiency. This performance deficiency was more than minor and is therefore a finding because it was associated with the protection against external factors (seismic) attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, because Diablo Canyon staff did not fix or perform evaluations of seismic induced system interactions on safety-related or accident mitigating systems, this had the potential to challenge the availability, reliability, and capability of various systems required to function following or during earthquakes to prevent undesirable consequence.

Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating System Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding was associated with seismic design or qualification of systems, structures, and components but did not result in the loss of a system operability or functionality.

The inspectors determined this finding has a problem identification and resolution cross cutting aspect associated with the Identification attribute; specifically in that PG&E personnel failed to implement the SISIP with a low enough threshold for identifying and assessing seismic induced system interactions in accordance with the SISI program and procedures [P.1].
Inspection Report# : 2014003 *(pdf)*

**Significance:** G Mar 21, 2014

Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Results in Inadequate Operability Assessment**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to follow the operability assessment procedure in considering the tornado atmospheric effects and tornado missile impactive force effects on the emergency diesel generator radiator ventilation plenum and engine exhaust pipes. The licensee took immediate corrective actions to remove potential tornado missiles that may affect the operability of the emergency diesel generators.
The licensee’s failure to account for tornado atmospheric pressure change effects and tornado-generated missile impactive loads is a performance deficiency. Specifically, the operability assessment did not account for the pressure change or impactive loads as described by the Standard Review Plan methodology. This performance deficiency was more than minor because it is associated with the protection against external factors attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process (SDP) For Findings At-Power”, dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk analysis determined that the calculated tornado missile strike frequency at Diablo Canyon is lower than the 1 x 10-6 threshold in the significance determination process, and therefore, the finding was determined to be of very low safety significance (Green).
This finding has a problem identification and resolution cross-cutting aspect associated with evaluation; specifically in that the licensee did not thoroughly evaluate the problem to ensure that resolutions addressed the cause(s) and extent of conditions, commensurate with their safety significance [P.2].
Inspection Report# : 2014002 *(pdf)*

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# Barrier Integrity

**Significance:** G Dec 31, 2013
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Loss of Control Room Ventilation System due to Inadequate Design Control**
The inspectors reviewed a Green self-revealing non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” after the licensee performed a design change to the control room ventilation system (CRVS) that resulted in none of the four CRVS pressurization fans being able to continuously operate if they started in response to a Phase A containment isolation or control room radiation atmosphere intake actuation signal. This resulted in declaring the Units 1 and 2 CRVS actuation instrumentation and CRVS inoperable and an unplanned entry into Technical Specifications (TS) 3.3.7, "Control Room Ventilation System Actuation Instrumentation," and TS 3.7.10, "Control Room Ventilation System," respectively.

The failure to use proper design control during the CRVS modification was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Barrier Integrity Cornerstone, and it adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radiological releases caused by accidents or events, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 3, “Barrier Integrity Screening Questions,” this finding was determined to be of very low safety significance (Green) because only the radiological barrier function of the control room was affected. The licensee entered the condition into the corrective action program as Notification 50525605.

The finding had a cross cutting aspect in the area of human performance resources component because licensee staff did not maintain complete, accurate, and up to date design documentation – specifically, because the functions of the pressure switches and CRVS interlocks had never been adequately described in design control documents [H.2(c)].
Inspection Report# : 2013005 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : November 26, 2014

# Diablo Canyon 2
# 4Q/2014 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Implement Fire Protection Program**
The inspectors identified a non-cited violation of the licensee's approved fire protection program as defined in Diablo Canyon Unit 2 Facility Operating License Condition 2.C(4) for failure to effectively implement the fire protection program. Specifically, the inspectors identified that maintenance personnel inappropriately disabled a fire hose reel credited for fire protection of the mechanical penetration area. The licensee entered the condition into the corrective action program as Notifications 50663810 and 50663589.

The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The inspectors evaluated this finding using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process." The finding affected fixed fire suppression systems. Using Inspection Manual Chapter 0609, Appendix F, Attachment 1 "Fire Protection SDP Phase 1 Worksheet," the deficiency affected a fixed fire suppression system and the finding affects only a manually actuated suppression system for an area which is accessible by the fire brigade; therefore the finding was of very low safety significance (Green). This finding had a cross cutting aspect in the area of human performance associated with the work management component, because the organization did not implement a process of planning, controlling and executing work activities such that nuclear safety is the overriding priority [H.5]. (Section 1R05)

Inspection Report# : 2014005 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Degraded Conditions in the Corrective Action Process**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and drawings," in that the licensee did not enter degraded conditions into the corrective action process. The inspectors identified two examples. Specifically, on May 12-13, 2014, the licensee experienced high temperatures in the 480 volt vital bus rooms and did not initiate a notification to document the unexpected condition. Second, on May 20, 2014, the licensee failed to document that a 480 volt vital bus room ventilation system register louvers was found closed.

The failure to enter problems into the corrective action process on the 480 volt busses was a performance deficiency.

The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. The inspectors determined this finding has a human performance cross-cutting aspect associated with challenging the unknown attribute, specifically in that licensee personnel did not maintain a questioning attitude to resolve unexpected conditions [H.11]. (Section 1R15)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: Self-Revealing
Item Type: NCV NonCited Violation
**Inadequate Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors reviewed a Green self-revealing, non-cited violation of Technical Specification 5.4.1.a, “Procedures,” for failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment. Specifically, inspectors reviewed the licensee performance associated with surveillance and maintenance activities and identified two examples of improper configuration of safety-related equipment returned to service, because of inadequate preplanned maintenance procedures.

The failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment is a performance deficiency. The inspectors determined that the finding was more than minor because it is associated with the procedure quality attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, the restriction of airflow caused by inadvertent closure of ventilation registers following the damper inspection resulted in the undesired consequences of higher ambient 480 volt switchgear room temperatures. In addition, the misconfiguration of the source range N-32 nuclear instrumentation impacted the functioning of the P-6 permissive and prevented it from performing properly during Unit 2 reactor startup such that operator action was necessary to prevent damage to the detector. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of mitigating structures, systems, and components that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with documentation, because the licensee did not ensure plant activities are governed with comprehensive maintenance procedures which are complete, accurate, and up to date to ensure work processes did not affect the performance of safety-related equipment [H.7]. (Section 4OA2.2)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Inadequate Procedure Results in Unnecessary Main Steam Safety Valve Lift**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee failure to prescribe a procedure appropriate to the circumstances with respect to safety-related atmospheric dump valves and main steam safety valves. Specifically, control of atmospheric

steam dump valves was not appropriate for a rapid plant shutdown resulting in unnecessary lifting of a spring-loaded main steam safety valve.

The inspectors determined that the licensee's failure to ensure appropriate procedures to properly control steam generator pressure and prevent unnecessary lifting of main steam safety valves was a performance deficiency. This performance deficiency was determined to be more than minor because it affected the Mitigating Systems cornerstone attribute of procedural quality and the objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with avoiding complacency, because the licensee failed to recognize during rapid load reductions the inherent risk of lifting a main steam safety valve and did not recognize or plan with adequate procedures, for a condition with a potential latent problem [H.12]. (Section 4OA3.3)

Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Document Degraded Emergency Diesel Generator Fuel Injector Nozzles in the Corrective Action Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," in that the licensee did not enter degraded conditions into the corrective action process. The first example of this violation occurred in ten identified instances from 2009 to 2012 when technicians failed to document degraded emergency diesel generator fuel injector nozzles in the corrective action program. The second example occurred in July and August 2014 when engineering personnel failed to appropriately document loose bolts on 4.16kV breaker panels in the corrective action program. The licensee documented this issue in the corrective action program as SAPNs 50641514 and 50656750 and issued a communication to the station reminding personnel of the requirement to initiate notifications even when problems are immediately resolved.

The failure to document unsatisfactory emergency diesel generator fuel injection nozzles and loose 4.16kV switchgear bolts in the corrective action program as required by procedure was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is therefore a finding. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. The inspectors determined this finding has an identification cross-cutting aspect in the problem identification and resolution cross-cutting area because the organization failed to implement a corrective action program with a low threshold for identification (P.1). Specifically, personnel failed to recognize that identified deficiencies were deviations from standards and that degraded conditions were promptly documented in the corrective action program.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV NonCited Violation

**Inadequate Technical Specification Surveillance Requirement for Emergency Diesel Generators**
The team identified a Green non-cited violation of 10 CFR 50.36 for the licensee's failure to establish an appropriate surveillance test to demonstrate operability of its emergency diesel generators. After revising its emergency diesel generator loading analysis, the licensee failed to adjust the parameters for the full-load-reject surveillance to ensure the test was performed with the maximum anticipated electrical loading. After the team identified this violation, the licensee entered Surveillance Requirement 3.0.3 and documented the condition in its corrective action program as SAPNs 50657635 and 50657637.

The licensee's failure to specify the "lowest functional capability or performance level of equipment required for safe operation of the facility" as required by 10 CFR 50.36 was a performance deficiency. This performance deficiency was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a resolution cross-cutting aspect in the problem identification and resolution cross-cutting area because the licensee failed to take effective corrective actions to address the nonconservative surveillance parameters in a timely manner (P.3). Specifically, the licensee did not take appropriate interim corrective actions to mitigate the issue while more fundamental causes were being assessed.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Longstanding Uncompensated Nonconforming Condition**
The team identified a Green non-cited violation of 10 CFR Part 50 Appendix B Criterion XVI for the licensee's failure to take timely corrective actions. In 2011, the licensee identified a potential path for gas intrusion into the containment spray system, contrary to design basis requirements. The licensee took no interim or compensatory actions while planning its final corrective actions. The licensee documented this condition in its corrective action program as SAPN 50657636.

The failure to take timely corrective actions as required by 10 CFR 50 Appendix B Criterion XVI was a performance deficiency. This performance deficiency was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609 Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a conservative bias cross-cutting aspect in the human performance cross-cutting area because licensee personnel failed to use decision-making practices that emphasized prudent choices over those that were simply allowable (H.14). Specifically, licensee managers failed to take timely action to address degraded conditions commensurate with their safety significance.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Jun 30, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Associated with Seismic Induced Structural Interactions**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly identify and evaluate system interactions as required by the licensee's Seismically-Induced Systems Interaction Program (SISIP) Procedure AD4.ID3, "SISIP Housekeeping Activities." Specifically, the inspectors identified multiple instances of components or sources capable of producing a

potential threat related to seismic induced structural interactions of safety related equipment or components.

The failure of plant personnel to follow procedure requirements to properly identify and evaluate for impact equipment near sensitive or safety related equipment was a performance deficiency. This performance deficiency was more than minor and is therefore a finding because it was associated with the protection against external factors (seismic) attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, because Diablo Canyon staff did not fix or perform evaluations of seismic induced system interactions on safety-related or accident mitigating systems, this had the potential to challenge the availability, reliability, and capability of various systems required to function following or during earthquakes to prevent undesirable consequence.

Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating System Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding was associated with seismic design or qualification of systems, structures, and components but did not result in the loss of a system operability or functionality.

The inspectors determined this finding has a problem identification and resolution cross cutting aspect associated with the Identification attribute; specifically in that PG&E personnel failed to implement the SISIP with a low enough threshold for identifying and assessing seismic induced system interactions in accordance with the SISI program and procedures [P.1].
Inspection Report# : 2014003 *(pdf)*

**Significance:** G Mar 21, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Follow Procedure Results in Inadequate Operability Assessment**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to follow the operability assessment procedure in considering the tornado atmospheric effects and tornado missile impactive force effects on the emergency diesel generator radiator ventilation plenum and engine exhaust pipes. The licensee took immediate corrective actions to remove potential tornado missiles that may affect the operability of the emergency diesel generators.
The licensee's failure to account for tornado atmospheric pressure change effects and tornado-generated missile impactive loads is a performance deficiency. Specifically, the operability assessment did not account for the pressure change or impactive loads as described by the Standard Review Plan methodology. This performance deficiency was more than minor because it is associated with the protection against external factors attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process (SDP) For Findings At-Power", dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk analysis determined that the calculated tornado missile strike frequency at Diablo Canyon is lower than the 1 x 10-6 threshold in the significance determination process, and therefore, the finding was determined to be of very low safety significance (Green).
This finding has a problem identification and resolution cross-cutting aspect associated with evaluation; specifically in that the licensee did not thoroughly evaluate the problem to ensure that resolutions addressed the cause(s) and extent of conditions, commensurate with their safety significance [P.2].
Inspection Report# : 2014002 *(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

**Significance: TBD** Oct 17, 2014
Identified By: NRC
Item Type: AV Apparent Violation
**Failure to Obtain Prior Approval for a Change Which Decreased the Effectiveness of the Emergency Plan**
The inspectors identified an apparent Severity Level III violation of 10 CFR 50.54(q) and an associated preliminary finding of low to moderate significance (White) for failing to obtain prior approval for an emergency plan change which decreased the effectiveness of the emergency plan. Specifically, on November 4, 2005, without approval from the NRC, the licensee removed instructions in emergency plan implementing procedures for making protective action recommendations for members of the public on the ocean within the 10-mile emergency planning zone, decreasing the plan's effectiveness.

The plan change, as implemented, resulted in a failure to meet the planning standard requirement of 10 CFR 50.47(b)(10) to develop and have in place procedures for the issuance of protective action recommendations (PARs) for the plume exposure pathway emergency planning zone, specifically, for areas of the ocean. This change constituted a decrease in effectiveness of the plan and, therefore, implementing the change without prior approval from the NRC is a performance deficiency. This performance deficiency is more than minor because it impacts the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Using the examples in Table 5.10-1, "Significance Examples § 50.47(b)(10)," of Appendix B to Inspection Manual Chapter 0609, "Emergency Preparedness Significance Determination Process," the inspectors concluded that this finding represents a degradation of the licensee's risk-significant planning standard function. The required planning standard function was degraded because the licensee's procedures did not direct the licensee to issue appropriate protective action recommendations to cover affected areas over the ocean within 5 to 10 miles of the site. The planning standard function was degraded, rather than lost, because default procedural actions of local governments would have resulted in effective protective actions for affected areas within 5 miles of the site. The finding does not present an immediate safety concern because, even without appropriate protective action recommendations from the licensee, the local governments would have ordered adequate protective actions for members of the public in the affected areas. No cross-cutting aspect is proposed as this performance deficiency occurred in 2005 and is not indicative of current licensee performance.
Inspection Report# : 2014502 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Control Access to a High Radiation Area With Dose Rates Greater Than 1 Rem/Hour**
The inspectors reviewed a self-revealing non-cited violation of Technical Specification 5.7.2 because the licensee failed to control access to a high radiation area with dose rates greater than 1 rem/hour. A radiation protection technician assumed responsibility for guarding the area and reestablished compliance with technical specification

requirements. Licensee representatives documented the occurrence in the corrective action program and performed an apparent cause evaluation.

The failure to control access to a high radiation area with dose rates greater than 1 rem/hour is a performance deficiency. The requirement not met was Technical Specification 5.7.2. The significance of the performance deficiency was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern if workers had entered an uncontrolled high radiation area and received unintended radiation dose. The Occupational Radiation Safety Cornerstone was affected; therefore, the inspectors used Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," August 19, 2008, to determine the significance of the violation. The violation had very low safety significance because: (1) It was not an as low as is reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This violation has a cross cutting aspect in the human performance area, associated with avoiding complacency, because individuals did not recognize and plan for the possibility of mistakes, latent issues, and inherent risk and did not implement appropriate error reduction tools [H.12]. (Section 2RS1)

Inspection Report# : 2014005 *(pdf)*

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : February 26, 2015

# Diablo Canyon 2
# 1Q/2015 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Provide Adequate Design Review of Emergency Diesel Generator 2-3**
The inspectors documented a self-revealing violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to ensure that design control measures shall provide for verifying or checking the adequacy of design by the performance of design reviews and design control measures shall be applied to items such as maintenance, and repair; and delineation of acceptance criteria for inspections and tests. Specifically, the licensee failed to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1-2 and 1-1 trips and incorporate this modification into the design and installation of emergency diesel generator 2-3.

The licensee’s failure to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1 2 and 1 1 trips, and incorporate this modification into the design and installation of emergency diesel generator 2 3, was a performance deficiency. This performance deficiency was more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the performance deficiency adversely affected the diesel generator’s capability to operate loaded for the technical specification required time. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process For Findings At-Power,” dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance due to loss of a function of a single train for greater than its technical specification outage time. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk evaluation resulted determined that the findings was Green, or very low safety significance.

This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2015001 *(pdf)*

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Implement Fire Protection Program**
The inspectors identified a non-cited violation of the licensee’s approved fire protection program as defined in Diablo Canyon Unit 2 Facility Operating License Condition 2.C(4) for failure to effectively implement the fire protection

program. Specifically, the inspectors identified that maintenance personnel inappropriately disabled a fire hose reel credited for fire protection of the mechanical penetration area. The licensee entered the condition into the corrective action program as Notifications 50663810 and 50663589.

The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The inspectors evaluated this finding using Inspection Manual Chapter 0609, Appendix F, “Fire Protection Significance Determination Process.” The finding affected fixed fire suppression systems. Using Inspection Manual Chapter 0609, Appendix F, Attachment 1 “Fire Protection SDP Phase 1 Worksheet,” the deficiency affected a fixed fire suppression system and the finding affects only a manually actuated suppression system for an area which is accessible by the fire brigade; therefore the finding was of very low safety significance (Green). This finding had a cross cutting aspect in the area of human performance associated with the work management component, because the organization did not implement a process of planning, controlling and executing work activities such that nuclear safety is the overriding priority [H.5]. (Section 1R05)

Inspection Report# : 2014005 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document Degraded Conditions in the Corrective Action Process**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and drawings,” in that the licensee did not enter degraded conditions into the corrective action process. The inspectors identified two examples. Specifically, on May 12-13, 2014, the licensee experienced high temperatures in the 480 volt vital bus rooms and did not initiate a notification to document the unexpected condition. Second, on May 20, 2014, the licensee failed to document that a 480 volt vital bus room ventilation system register louvers was found closed.

The failure to enter problems into the corrective action process on the 480 volt busses was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. The inspectors determined this finding has a human performance cross-cutting aspect associated with challenging the unknown attribute, specifically in that licensee personnel did not maintain a questioning attitude to resolve unexpected conditions [H.11]. (Section 1R15)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Inadequate Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors reviewed a Green self-revealing, non-cited violation of Technical Specification 5.4.1.a, “Procedures,” for failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment. Specifically, inspectors reviewed the licensee performance associated with surveillance and maintenance

activities and identified two examples of improper configuration of safety-related equipment returned to service, because of inadequate preplanned maintenance procedures.

The failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment is a performance deficiency. The inspectors determined that the finding was more than minor because it is associated with the procedure quality attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, the restriction of airflow caused by inadvertent closure of ventilation registers following the damper inspection resulted in the undesired consequences of higher ambient 480 volt switchgear room temperatures. In addition, the misconfiguration of the source range N-32 nuclear instrumentation impacted the functioning of the P-6 permissive and prevented it from performing properly during Unit 2 reactor startup such that operator action was necessary to prevent damage to the detector. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of mitigating structures, systems, and components that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with documentation, because the licensee did not ensure plant activities are governed with comprehensive maintenance procedures which are complete, accurate, and up to date to ensure work processes did not affect the performance of safety-related equipment [H.7]. (Section 4OA2.2)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Procedure Results in Unnecessary Main Steam Safety Valve Lift**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee failure to prescribe a procedure appropriate to the circumstances with respect to safety-related atmospheric dump valves and main steam safety valves. Specifically, control of atmospheric steam dump valves was not appropriate for a rapid plant shutdown resulting in unnecessary lifting of a spring-loaded main steam safety valve.

The inspectors determined that the licensee’s failure to ensure appropriate procedures to properly control steam generator pressure and prevent unnecessary lifting of main steam safety valves was a performance deficiency. This performance deficiency was determined to be more than minor because it affected the Mitigating Systems cornerstone attribute of procedural quality and the objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with avoiding complacency, because the licensee failed to recognize during rapid load reductions the inherent risk of lifting a main steam safety valve and did not recognize or plan with adequate procedures, for a condition with a potential latent problem [H.12]. (Section 4OA3.3)

Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document Degraded Emergency Diesel Generator Fuel Injector Nozzles in the Corrective Action Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," in that the licensee did not enter degraded conditions into the corrective action process. The first example of this violation occurred in ten identified instances from 2009 to 2012 when technicians failed to document degraded emergency diesel generator fuel injector nozzles in the corrective action program. The second example occurred in July and August 2014 when engineering personnel failed to appropriately document loose bolts on 4.16kV breaker panels in the corrective action program. The licensee documented this issue in the corrective action program as SAPNs 50641514 and 50656750 and issued a communication to the station reminding personnel of the requirement to initiate notifications even when problems are immediately resolved.

The failure to document unsatisfactory emergency diesel generator fuel injection nozzles and loose 4.16kV switchgear bolts in the corrective action program as required by procedure was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is therefore a finding. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. The inspectors determined this finding has an identification cross-cutting aspect in the problem identification and resolution cross-cutting area because the organization failed to implement a corrective action program with a low threshold for identification (P.1). Specifically, personnel failed to recognize that identified deficiencies were deviations from standards and that degraded conditions were promptly documented in the corrective action program.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Technical Specification Surveillance Requirement for Emergency Diesel Generators**
The team identified a Green non-cited violation of 10 CFR 50.36 for the licensee's failure to establish an appropriate surveillance test to demonstrate operability of its emergency diesel generators. After revising its emergency diesel generator loading analysis, the licensee failed to adjust the parameters for the full-load-reject surveillance to ensure the test was performed with the maximum anticipated electrical loading. After the team identified this violation, the licensee entered Surveillance Requirement 3.0.3 and documented the condition in its corrective action program as SAPNs 50657635 and 50657637.

The licensee's failure to specify the "lowest functional capability or performance level of equipment required for safe operation of the facility" as required by 10 CFR 50.36 was a performance deficiency. This performance deficiency was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a resolution cross-cutting aspect in the problem identification and resolution cross-cutting area because the licensee failed to take effective corrective actions to address the nonconservative surveillance parameters in a timely manner (P.3). Specifically, the licensee did not take appropriate interim corrective actions to mitigate the issue while more fundamental causes were being assessed.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Longstanding Uncompensated Nonconforming Condition**
The team identified a Green non-cited violation of 10 CFR Part 50 Appendix B Criterion XVI for the licensee's failure to take timely corrective actions. In 2011, the licensee identified a potential path for gas intrusion into the containment spray system, contrary to design basis requirements. The licensee took no interim or compensatory actions while planning its final corrective actions. The licensee documented this condition in its corrective action program as SAPN 50657636.

The failure to take timely corrective actions as required by 10 CFR 50 Appendix B Criterion XVI was a performance deficiency. This performance deficiency was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609 Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a conservative bias cross-cutting aspect in the human performance cross-cutting area because licensee personnel failed to use decision-making practices that emphasized prudent choices over those that were simply allowable (H.14). Specifically, licensee managers failed to take timely action to address degraded conditions commensurate with their safety significance.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Jun 30, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Follow Procedure Associated with Seismic Induced Structural Interactions**
The inspectors identified a Green non-cited violation of 10 CFR, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly identify and evaluate system interactions as required by the licensee's Seismically-Induced Systems Interaction Program (SISIP) Procedure AD4.ID3, "SISIP Housekeeping Activities." Specifically, the inspectors identified multiple instances of components or sources capable of producing a potential threat related to seismic induced structural interactions of safety related equipment or components.

The failure of plant personnel to follow procedure requirements to properly identify and evaluate for impact equipment near sensitive or safety related equipment was a performance deficiency. This performance deficiency was more than minor and is therefore a finding because it was associated with the protection against external factors (seismic) attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, because Diablo Canyon staff did not fix or perform evaluations of seismic induced system interactions on safety-related or accident mitigating systems, this had the potential to challenge the availability, reliability, and capability of various systems required to function following or during earthquakes to prevent undesirable consequence.

Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating System Screening Questions," the finding was determined to be of very low safety significance (Green) because the finding was associated with seismic design or qualification of systems, structures, and components but did not result in the loss of a system operability or functionality.

The inspectors determined this finding has a problem identification and resolution cross cutting aspect associated with the Identification attribute; specifically in that PG&E personnel failed to implement the SISIP with a low enough threshold for identifying and assessing seismic induced system interactions in accordance with the SISI program and

procedures [P.1].
Inspection Report# : 2014003 *(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

**Significance: TBD** Oct 17, 2014
Identified By: NRC
Item Type: AV Apparent Violation
**Failure to Obtain Prior Approval for a Change Which Decreased the Effectiveness of the Emergency Plan**
The inspectors identified an apparent Severity Level III violation of 10 CFR 50.54(q) and an associated preliminary finding of low to moderate significance (White) for failing to obtain prior approval for an emergency plan change which decreased the effectiveness of the emergency plan. Specifically, on November 4, 2005, without approval from the NRC, the licensee removed instructions in emergency plan implementing procedures for making protective action recommendations for members of the public on the ocean within the 10-mile emergency planning zone, decreasing the plan’s effectiveness.

The plan change, as implemented, resulted in a failure to meet the planning standard requirement of 10 CFR 50.47(b)(10) to develop and have in place procedures for the issuance of protective action recommendations (PARs) for the plume exposure pathway emergency planning zone, specifically, for areas of the ocean. This change constituted a decrease in effectiveness of the plan and, therefore, implementing the change without prior approval from the NRC is a performance deficiency. This performance deficiency is more than minor because it impacts the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Using the examples in Table 5.10-1, “Significance Examples § 50.47(b)(10),” of Appendix B to Inspection Manual Chapter 0609, "Emergency Preparedness Significance Determination Process," the inspectors concluded that this finding represents a degradation of the licensee’s risk-significant planning standard function. The required planning standard function was degraded because the licensee’s procedures did not direct the licensee to issue appropriate protective action recommendations to cover affected areas over the ocean within 5 to 10 miles of the site. The planning standard function was degraded, rather than lost, because default procedural actions of local governments would have resulted in effective protective actions for affected areas within 5 miles of the site. The finding does not present an immediate safety concern because, even without appropriate protective action recommendations from the licensee, the local governments would have ordered adequate protective actions for members of the public in the affected areas. No cross-cutting aspect is proposed as this performance deficiency occurred in 2005 and is not indicative of current licensee performance.
Inspection Report# : 2014502 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2014
Identified By: NRC

Item Type: NCV Non-Cited Violation
**Failure to Control Access to a High Radiation Area With Dose Rates Greater Than 1 Rem/Hour**
The inspectors reviewed a self-revealing non-cited violation of Technical Specification 5.7.2 because the licensee failed to control access to a high radiation area with dose rates greater than 1 rem/hour. A radiation protection technician assumed responsibility for guarding the area and reestablished compliance with technical specification requirements. Licensee representatives documented the occurrence in the corrective action program and performed an apparent cause evaluation.

The failure to control access to a high radiation area with dose rates greater than 1 rem/hour is a performance deficiency. The requirement not met was Technical Specification 5.7.2. The significance of the performance deficiency was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern if workers had entered an uncontrolled high radiation area and received unintended radiation dose. The Occupational Radiation Safety Cornerstone was affected; therefore, the inspectors used Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," August 19, 2008, to determine the significance of the violation. The violation had very low safety significance because: (1) It was not an as low as is reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This violation has a cross cutting aspect in the human performance area, associated with avoiding complacency, because individuals did not recognize and plan for the possibility of mistakes, latent issues, and inherent risk and did not implement appropriate error reduction tools [H.12]. (Section 2RS1)

Inspection Report# : 2014005 *(pdf)*

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Miscellaneous

Last modified : June 16, 2015

# Diablo Canyon 2
# 2Q/2015 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Provide Adequate Design Review of Emergency Diesel Generator 2-3**
The inspectors documented a self-revealing violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to ensure that design control measures shall provide for verifying or checking the adequacy of design by the performance of design reviews and design control measures shall be applied to items such as maintenance, and repair; and delineation of acceptance criteria for inspections and tests. Specifically, the licensee failed to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1-2 and 1-1 trips and incorporate this modification into the design and installation of emergency diesel generator 2-3.

The licensee’s failure to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1 2 and 1 1 trips, and incorporate this modification into the design and installation of emergency diesel generator 2 3, was a performance deficiency. This performance deficiency was more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the performance deficiency adversely affected the diesel generator’s capability to operate loaded for the technical specification required time. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process For Findings At-Power,” dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance due to loss of a function of a single train for greater than its technical specification outage time. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk evaluation resulted determined that the findings was Green, or very low safety significance.

This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2015001 *(pdf)*

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Implement Fire Protection Program**
The inspectors identified a non-cited violation of the licensee’s approved fire protection program as defined in Diablo Canyon Unit 2 Facility Operating License Condition 2.C(4) for failure to effectively implement the fire protection

program. Specifically, the inspectors identified that maintenance personnel inappropriately disabled a fire hose reel credited for fire protection of the mechanical penetration area. The licensee entered the condition into the corrective action program as Notifications 50663810 and 50663589.

The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The inspectors evaluated this finding using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process." The finding affected fixed fire suppression systems. Using Inspection Manual Chapter 0609, Appendix F, Attachment 1 "Fire Protection SDP Phase 1 Worksheet," the deficiency affected a fixed fire suppression system and the finding affects only a manually actuated suppression system for an area which is accessible by the fire brigade; therefore the finding was of very low safety significance (Green). This finding had a cross cutting aspect in the area of human performance associated with the work management component, because the organization did not implement a process of planning, controlling and executing work activities such that nuclear safety is the overriding priority [H.5]. (Section 1R05)

Inspection Report# : 2014005 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document Degraded Conditions in the Corrective Action Process**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and drawings," in that the licensee did not enter degraded conditions into the corrective action process. The inspectors identified two examples. Specifically, on May 12-13, 2014, the licensee experienced high temperatures in the 480 volt vital bus rooms and did not initiate a notification to document the unexpected condition. Second, on May 20, 2014, the licensee failed to document that a 480 volt vital bus room ventilation system register louvers was found closed.

The failure to enter problems into the corrective action process on the 480 volt busses was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences, and is therefore a finding. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," this finding was determined to be of very low safety significance (Green) because, it was not a design or qualification deficiency, was not a loss of the system or function, and did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time. The inspectors determined this finding has a human performance cross-cutting aspect associated with challenging the unknown attribute, specifically in that licensee personnel did not maintain a questioning attitude to resolve unexpected conditions [H.11]. (Section 1R15)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Inadequate Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors reviewed a Green self-revealing, non-cited violation of Technical Specification 5.4.1.a, "Procedures," for failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment. Specifically, inspectors reviewed the licensee performance associated with surveillance and maintenance

activities and identified two examples of improper configuration of safety-related equipment returned to service, because of inadequate preplanned maintenance procedures.

The failure to implement properly preplanned maintenance procedures affecting the performance of safety-related equipment is a performance deficiency. The inspectors determined that the finding was more than minor because it is associated with the procedure quality attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, the restriction of airflow caused by inadvertent closure of ventilation registers following the damper inspection resulted in the undesired consequences of higher ambient 480 volt switchgear room temperatures. In addition, the misconfiguration of the source range N-32 nuclear instrumentation impacted the functioning of the P-6 permissive and prevented it from performing properly during Unit 2 reactor startup such that operator action was necessary to prevent damage to the detector. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” this finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of mitigating structures, systems, and components that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with documentation, because the licensee did not ensure plant activities are governed with comprehensive maintenance procedures which are complete, accurate, and up to date to ensure work processes did not affect the performance of safety-related equipment [H.7]. (Section 4OA2.2)
Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 19, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Procedure Results in Unnecessary Main Steam Safety Valve Lift**
The inspectors identified a Green non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee failure to prescribe a procedure appropriate to the circumstances with respect to safety-related atmospheric dump valves and main steam safety valves. Specifically, control of atmospheric steam dump valves was not appropriate for a rapid plant shutdown resulting in unnecessary lifting of a spring-loaded main steam safety valve.

The inspectors determined that the licensee’s failure to ensure appropriate procedures to properly control steam generator pressure and prevent unnecessary lifting of main steam safety valves was a performance deficiency. This performance deficiency was determined to be more than minor because it affected the Mitigating Systems cornerstone attribute of procedural quality and the objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. Using Inspection Manual Chapter 0609, Attachment 04, “Initial Characterization of Findings,” and Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component that did not affect operability or functionality.

The inspectors concluded that this finding affected the cross-cutting aspect of human performance associated with avoiding complacency, because the licensee failed to recognize during rapid load reductions the inherent risk of lifting a main steam safety valve and did not recognize or plan with adequate procedures, for a condition with a potential latent problem [H.12]. (Section 4OA3.3)

Inspection Report# : 2014004 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document Degraded Emergency Diesel Generator Fuel Injector Nozzles in the Corrective Action Program**
The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," in that the licensee did not enter degraded conditions into the corrective action process. The first example of this violation occurred in ten identified instances from 2009 to 2012 when technicians failed to document degraded emergency diesel generator fuel injector nozzles in the corrective action program. The second example occurred in July and August 2014 when engineering personnel failed to appropriately document loose bolts on 4.16kV breaker panels in the corrective action program. The licensee documented this issue in the corrective action program as SAPNs 50641514 and 50656750 and issued a communication to the station reminding personnel of the requirement to initiate notifications even when problems are immediately resolved.

The failure to document unsatisfactory emergency diesel generator fuel injection nozzles and loose 4.16kV switchgear bolts in the corrective action program as required by procedure was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. It is therefore a finding. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. The inspectors determined this finding has an identification cross-cutting aspect in the problem identification and resolution cross-cutting area because the organization failed to implement a corrective action program with a low threshold for identification (P.1). Specifically, personnel failed to recognize that identified deficiencies were deviations from standards and that degraded conditions were promptly documented in the corrective action program.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Technical Specification Surveillance Requirement for Emergency Diesel Generators**
The team identified a Green non-cited violation of 10 CFR 50.36 for the licensee's failure to establish an appropriate surveillance test to demonstrate operability of its emergency diesel generators. After revising its emergency diesel generator loading analysis, the licensee failed to adjust the parameters for the full-load-reject surveillance to ensure the test was performed with the maximum anticipated electrical loading. After the team identified this violation, the licensee entered Surveillance Requirement 3.0.3 and documented the condition in its corrective action program as SAPNs 50657635 and 50657637.

The licensee's failure to specify the "lowest functional capability or performance level of equipment required for safe operation of the facility" as required by 10 CFR 50.36 was a performance deficiency. This performance deficiency was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a resolution cross-cutting aspect in the problem identification and resolution cross-cutting area because the licensee failed to take effective corrective actions to address the nonconservative surveillance parameters in a timely manner (P.3). Specifically, the licensee did not take appropriate interim corrective actions to mitigate the issue while more fundamental causes were being assessed.
Inspection Report# : 2014007 *(pdf)*

**Significance:** G Sep 12, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Longstanding Uncompensated Nonconforming Condition**
The team identified a Green non-cited violation of 10 CFR Part 50 Appendix B Criterion XVI for the licensee's failure to take timely corrective actions. In 2011, the licensee identified a potential path for gas intrusion into the containment spray system, contrary to design basis requirements. The licensee took no interim or compensatory actions while planning its final corrective actions. The licensee documented this condition in its corrective action program as SAPN 50657636.

The failure to take timely corrective actions as required by 10 CFR 50 Appendix B Criterion XVI was a performance deficiency. This performance deficiency was more than minor because it was associated with the design control attribute of the mitigating systems cornerstone and it adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609 Appendix A, the team determined that this finding was of very low safety significance (Green) because it did not result in the loss of operability or functionality of a system or train. This finding has a conservative bias cross-cutting aspect in the human performance cross-cutting area because licensee personnel failed to use decision-making practices that emphasized prudent choices over those that were simply allowable (H.14). Specifically, licensee managers failed to take timely action to address degraded conditions commensurate with their safety significance.
Inspection Report# : 2014007 *(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance: TBD** Oct 17, 2014
Identified By: NRC
Item Type: AV Apparent Violation
**Failure to Obtain Prior Approval for a Change Which Decreased the Effectiveness of the Emergency Plan**
The inspectors identified an apparent Severity Level III violation of 10 CFR 50.54(q) and an associated preliminary finding of low to moderate significance (White) for failing to obtain prior approval for an emergency plan change which decreased the effectiveness of the emergency plan. Specifically, on November 4, 2005, without approval from the NRC, the licensee removed instructions in emergency plan implementing procedures for making protective action recommendations for members of the public on the ocean within the 10-mile emergency planning zone, decreasing the plan's effectiveness.

The plan change, as implemented, resulted in a failure to meet the planning standard requirement of 10 CFR 50.47(b)(10) to develop and have in place procedures for the issuance of protective action recommendations (PARs) for the plume exposure pathway emergency planning zone, specifically, for areas of the ocean. This change constituted a decrease in effectiveness of the plan and, therefore, implementing the change without prior approval from the NRC is a performance deficiency. This performance deficiency is more than minor because it impacts the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Using the examples in Table 5.10-1, "Significance Examples § 50.47(b)(10)," of Appendix B to Inspection Manual Chapter 0609, "Emergency Preparedness Significance Determination Process," the inspectors

concluded that this finding represents a degradation of the licensee's risk-significant planning standard function. The required planning standard function was degraded because the licensee's procedures did not direct the licensee to issue appropriate protective action recommendations to cover affected areas over the ocean within 5 to 10 miles of the site. The planning standard function was degraded, rather than lost, because default procedural actions of local governments would have resulted in effective protective actions for affected areas within 5 miles of the site. The finding does not present an immediate safety concern because, even without appropriate protective action recommendations from the licensee, the local governments would have ordered adequate protective actions for members of the public in the affected areas. No cross-cutting aspect is proposed as this performance deficiency occurred in 2005 and is not indicative of current licensee performance.
Inspection Report# : 2014502 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Control Access to a High Radiation Area With Dose Rates Greater Than 1 Rem/Hour**
The inspectors reviewed a self-revealing non-cited violation of Technical Specification 5.7.2 because the licensee failed to control access to a high radiation area with dose rates greater than 1 rem/hour. A radiation protection technician assumed responsibility for guarding the area and reestablished compliance with technical specification requirements. Licensee representatives documented the occurrence in the corrective action program and performed an apparent cause evaluation.

The failure to control access to a high radiation area with dose rates greater than 1 rem/hour is a performance deficiency. The requirement not met was Technical Specification 5.7.2. The significance of the performance deficiency was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern if workers had entered an uncontrolled high radiation area and received unintended radiation dose. The Occupational Radiation Safety Cornerstone was affected; therefore, the inspectors used Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," August 19, 2008, to determine the significance of the violation. The violation had very low safety significance because: (1) It was not an as low as is reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This violation has a cross cutting aspect in the human performance area, associated with avoiding complacency, because individuals did not recognize and plan for the possibility of mistakes, latent issues, and inherent risk and did not implement appropriate error reduction tools [H.12]. (Section 2RS1)

Inspection Report# : 2014005 *(pdf)*

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# Public Radiation Safety

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# Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

Last modified : August 07, 2015

# Diablo Canyon 2
# 3Q/2015 Plant Inspection Findings

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## Initiating Events

**Significance:** G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Appropriately Scope 230 KV Switchyard into the Maintenance Rule Monitoring Program**
The inspectors identified a Green, non-cited violation of 10 CFR 50.65(b)(2) for the licensee's failure to appropriately scope the 230 kV switchyard in the Maintenance Rule monitoring program. Specifically, from the inception of the facilities' monitoring program through May 18, 2015, the licensee failed to properly scope or evaluate the 230 kV switchyard to include the entire switchyard up through the first inter-tie circuit breakers CB262 and CB282 into the Maintenance Rule program. Electrical faults within the 230 kV switchyard can cause loss of offsite power which is relied upon to mitigate accidents and cause an actuation of a safety-related systems, such as, emergency diesel generators, and should have been included into its Maintenance Rule program. This issue was entered into the licensee's corrective action program as Notifications 50702970 and 50703118.

The inspectors determined that the licensee's failure to scope the 230 kV offsite power source including the switchyard up through the first breakers from the transmission system into the Maintenance Rule program was contrary to the requirements of 10 CFR 50.65 and therefore a performance deficiency. The performance deficiency was determined to be more than minor because it is associated with the initiating events attribute of protections against external factors and adversely affected the cornerstone objective, in that, a 230 kV switchyard failure can upset plant stability and challenge critical safety functions during shutdown as well as power operations. Failure to monitor the performance or condition of 230 kV offsite power source (including the switchyard up through the first breakers from the transmission system) in a manner sufficient to provide reasonable assurance the offsite power was capable of fulfilling the intended functions affected the reliability of the plant equipment to perform their safety function. The inspectors determined if the 230 kV switchyard was properly scoped into the Maintenance Rule program the loss of offsite power due to the flash over event may have been prevented. However the direct cause of the event has been identified as untimely corrective actions associated with an ineffective corrective action program. As such, improper Maintenance Rule scoping was not the direct cause. Therefore, the inspectors determined the finding could be evaluated using the significant determination process in accordance using IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions." The inspectors determined that the finding was of very low safety significance (Green) because the finding was determined not to be the cause of the actual 230 kV failure such that all of the screening questions in Exhibit 1 could be answered "no." The inspectors determined that since the scoping of the switchyard systems had occurred more than 3 years ago, and the opportunity to reevaluate system scoping had not recently occurred, the finding did not represent current licensee performance and therefore a cross-cutting aspect was not assigned.
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: Self-Revealing
Item Type: FIN Finding
**High Voltage Insulator Flashover Resulted in Loss of 230 kV Offsite Power and Start of Emergency Diesel Generators**

The inspectors reviewed a self-revealing, Green finding for the licensee’s failure to adequately implement procedure OM7.ID1, Problem Identification and Resolution, to prevent a high voltage insulator flashover event in the 230 kV switchyard that occurred on October 31, 2014. Specifically, corrective actions from three previous root cause evaluations were not effective to prevent a loss of the 230 kV start-up power and subsequent auto start of all of the safety standby emergency diesel generators (EDGs). This issue was entered into the licensee’s corrective action program as Notification 50699230.

The licensee’s failure to adequately implement procedure OM7.ID1, Problem Identification and Resolution was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, this failure resulted in another high-voltage insulator flashover, which resulted in loss of 230 kV offsite startup power and activation of all safety-related EDGs, on October 31, 2014. In accordance with IMC 0609.04, “Initial Characterization of Findings,” the inspectors determined that the impact of the finding on Unit 1 should be evaluated using Exhibit 1 of IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings at Power,” and further determined that this finding required a detailed risk evaluation by the regional senior risk analyst because the finding involved a partial loss of offsite power, a support system that contributes to the likelihood of an initiating event and affected mitigation equipment.

The risk analyst determined that, with the 230 kV system de-energized, any plant transient would result in a plant-centered loss of offsite power. Therefore, the risk analyst calculated the incremental conditional core damage probability for an exposure period of 9 hours to be 2.09 x 10-7, which is lower than the 1 x 10-6 threshold in the significance determination process; this finding is of very low safety significance (Green) for Unit 1. In accordance with IMC 0609.04, “Initial Characterization of Findings,” the inspectors determined that the impact of the finding on Unit 2 should be evaluated using IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” because the finding pertained to operations, an event, or a degraded condition while the plant was shut down. Unit 2 was shutdown in a refueling outage when the event occurred on October 31, 2014. Because of the shutdown configuration of Unit 2, the loss of 230 kV support system did not impact the ability to continue to provide decay heat removal for the unit. Therefore, the analyst determined qualitatively that this finding is also of very low safety significance (Green) for Unit 2. This finding has a cross-cutting aspect of work management, in the area of human performance, for failing to implement a process of planning, controlling, and executing work activities such that nuclear safety is an overriding priority. Specifically the licensee failed to effectively plan and coordinate preventative maintenance strategies associated with root causes from previous high-voltage insulators flashover or failures since 2008 to prevent the loss of offsite 230 kV and the transient on October 31, 2014 [H.5].
Inspection Report# : 2015002 *(pdf)*

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# Mitigating Systems

**Significance:** G Sep 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document an Adequate Evaluation for a Change in Seismic Load Combination Methodology**
The inspectors identified a Severity Level IV, Green, non cited violation of 10 CFR 50.59(d)(1) which requires, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2). Specifically, the licensee changed the method for combining earthquake loads and loss

of coolant accident loads from the absolute summation method to square root sum of the squares (SRSS) method without sufficient justification to demonstrate the change did not require prior NRC approval.

The licensee's failure to implement the requirements of 10 CFR 50.59 and adequately evaluate changes to determine if prior NRC approval is required was a performance deficiency. The licensee entered the issue into the corrective action program as Notification 50811191. In accordance with the licensee's corrective action program, this issue will be addressed by the licensee through a re-evaluation of the methodology change and the required actions that need to be taken by the licensee will be implemented. Additionally, the licensee performed an operability determination for the affected structures, systems, and components that established a reasonable expectation for operability pending final resolution of the issue.

This performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to determine that use of SRSS in the Watts Bar safety evaluation report cited in the PG&E evaluation represented a change in a method of evaluation, in that the Watts Bar safety evaluation report was very narrow in scope and not appropriate for the intended application at Diablo Canyon. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not result in the inoperability of the system. Because this performance deficiency had the potential to impact the NRC's ability to perform its regulatory function, the inspectors also evaluated the performance deficiency using traditional enforcement. Since the violation is associated with a Green finding having very low safety significance, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding had a cross cutting aspect in the area of human performance associated with design margins because individuals failed to ensure margins were carefully guarded and changed only through a systematic and rigorous process [H.6].
Inspection Report# : 2015003 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Maintain Operator Licensing Examination Integrity**
The inspectors reviewed a self-revealing, Severity Level IV non-cited violation of 10 CFR 55.49, "Integrity of Examinations and Tests," and an associated Green finding for the licensee's failure to provide adequate examination security measures during administration of the 2015 biennial requalification examination. On May 26, 2015, a licensed operator was able to obtain plant computer information that led to the discovery of specific plant events contained on the NRC-required annual operating test. The licensee entered this issue into the corrective action program as Notification 50704195 and retested the crew with a new scenario.

The failure of the licensee to provide adequate measures for examination security for the biennial requalification examinations was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the human performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 4, Tables 1 and 2 worksheets (issue date June 19, 2012); and the corresponding Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," Flowchart Block #10 (issue date December 6, 2011), the finding was determined to have very low safety significance (Green). Although the 2015 finding resulted in a compromise of the integrity of biennial dynamic simulator examinations had no compensatory actions been taken, the equitable and consistent administration of the biennial dynamic simulator examination was not actually affected

by this compromise. The traditional enforcement violation was determined to be a Severity Level IV violation consistent with Section 6.4.d of the Enforcement Policy. This finding has a cross-cutting aspect in the resources component of the human performance cross-cutting area because the licensee failed to ensure the procedures are adequate to ensure nuclear safety [H.1].
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Design Control for High-Energy Line Break Vent Flow Path**
The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the licensee’s failure to ensure credited design features, such as flow vent paths, protect safety-related systems, from temperature and pressure effects of a high-energy line break (HELB) in the auxiliary building. Specifically, the licensee allowed obstruction of a credited flow path with acrylic glass plates not qualified in the original design and not verified to function under a HELB scenario. The licensee entered this issue into the corrective action program as Notifications 50697910 and 50698102, and took immediate actions to remove the acrylic glass plates from the vent path doors in the auxiliary building.

The performance deficiency was determined to be more than minor because it affected the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not have adequate measures in place to ensure that qualified components were available to mitigate the consequences of a HELB in the auxiliary building. The finding screened as of very low safety significance (Green) because the finding did not affect the design or qualification of mitigating structures, systems, and components; the finding did not represent a loss of system and/or function; the finding did not represent an actual loss of a function of a single train for greater than the technical specification (TS) allowed outage time; the finding did not represent an actual loss of a function of one or more non-TS trains of equipment; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding was not assigned a cross-cutting aspect since the performance deficiency is not indicative of current plant performance.
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Provide Adequate Design Review of Emergency Diesel Generator 2-3**
The inspectors documented a self-revealing violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to ensure that design control measures shall provide for verifying or checking the adequacy of design by the performance of design reviews and design control measures shall be applied to items such as maintenance, and repair; and delineation of acceptance criteria for inspections and tests. Specifically, the licensee failed to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1-2 and 1-1 trips and incorporate this modification into the design and installation of emergency diesel generator 2-3.

The licensee’s failure to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1 2 and 1 1 trips, and incorporate this modification into the design and installation of emergency diesel generator 2 3, was a performance deficiency. This performance deficiency was more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the performance deficiency adversely affected the diesel generator's capability to operate loaded for the technical specification required time. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process For Findings At-Power," dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance due to loss of a function of a single train for greater than its technical specification outage time. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk evaluation resulted determined that the findings was Green, or very low safety significance.

This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2015001 *(pdf)*

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Implement Fire Protection Program**
The inspectors identified a non-cited violation of the licensee's approved fire protection program as defined in Diablo Canyon Unit 2 Facility Operating License Condition 2.C(4) for failure to effectively implement the fire protection program. Specifically, the inspectors identified that maintenance personnel inappropriately disabled a fire hose reel credited for fire protection of the mechanical penetration area. The licensee entered the condition into the corrective action program as Notifications 50663810 and 50663589.

The failure to effectively implement all fire prevention controls and processes as required in the approved fire protection program was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The inspectors evaluated this finding using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process." The finding affected fixed fire suppression systems. Using Inspection Manual Chapter 0609, Appendix F, Attachment 1 "Fire Protection SDP Phase 1 Worksheet," the deficiency affected a fixed fire suppression system and the finding affects only a manually actuated suppression system for an area which is accessible by the fire brigade; therefore the finding was of very low safety significance (Green). This finding had a cross cutting aspect in the area of human performance associated with the work management component, because the organization did not implement a process of planning, controlling and executing work activities such that nuclear safety is the overriding priority [H.5]. (Section 1R05)

Inspection Report# : 2014005 *(pdf)*

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## Barrier Integrity

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## Emergency Preparedness

**Significance:** W Oct 17, 2014
Identified By: NRC
Item Type: VIO Violation
**Failure to Obtain Prior Approval for a Change Which Decreased the Effectiveness of the Emergency Plan**

(Initial Entry)
The inspectors identified an apparent Severity Level III violation of 10 CFR 50.54(q) and an associated preliminary finding of low to moderate significance (White) for failing to obtain prior approval for an emergency plan change which decreased the effectiveness of the emergency plan. Specifically, on November 4, 2005, without approval from the NRC, the licensee removed instructions in emergency plan implementing procedures for making protective action recommendations for members of the public on the ocean within the 10-mile emergency planning zone, decreasing the plan’s effectiveness.

The plan change, as implemented, resulted in a failure to meet the planning standard requirement of 10 CFR 50.47(b)(10) to develop and have in place procedures for the issuance of protective action recommendations (PARs) for the plume exposure pathway emergency planning zone, specifically, for areas of the ocean. This change constituted a decrease in effectiveness of the plan and, therefore, implementing the change without prior approval from the NRC is a performance deficiency. This performance deficiency is more than minor because it impacts the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone objective to ensure that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Using the examples in Table 5.10-1, “Significance Examples § 50.47(b)(10),” of Appendix B to Inspection Manual Chapter 0609, "Emergency Preparedness Significance Determination Process," the inspectors concluded that this finding represents a degradation of the licensee’s risk-significant planning standard function. The required planning standard function was degraded because the licensee’s procedures did not direct the licensee to issue appropriate protective action recommendations to cover affected areas over the ocean within 5 to 10 miles of the site. The planning standard function was degraded, rather than lost, because default procedural actions of local governments would have resulted in effective protective actions for affected areas within 5 miles of the site. The finding does not present an immediate safety concern because, even without appropriate protective action recommendations from the licensee, the local governments would have ordered adequate protective actions for members of the public in the affected areas. No cross-cutting aspect is proposed as this performance deficiency occurred in 2005 and is not indicative of current licensee performance.

(First Update)
The NRC concluded that the violation is appropriately characterized as Severity Level III, and that the associated finding is appropriately characterized as White, meaning a finding of low to moderate safety significance.
Inspection Report# : 2014502 *(pdf)*
Inspection Report# : 2015502 *(pdf)*

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# Occupational Radiation Safety

**Significance:** G Sep 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Secure a Locked High Radiation Area**
The inspectors reviewed a self-revealing non-cited violation (NCV) of Technical Specification 5.4.1(a), “Procedures,” for failure to secure a locked high radiation area. Specifically, the padlock on the Letdown Filter 1-1 locking bar was found unlocked. Upon discovery, the licensee guarded the area until properly secured. This issue was entered into the licensee’s corrective action program as Notification 50710852.

The failure to secure a locked high radiation area was a performance deficiency. The performance deficiency was more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, failure to adequately secure the locked high radiation area could result in unintended exposure to high

levels of radiation. Using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, the inspectors determined the violation was of very low safety significance (Green) because: (1) it was not an as low as reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The finding had an avoid complacency cross-cutting aspect, in the area of human performance, because individuals failed to recognize and plan for the possibility of mistakes, even while expecting positive outcomes. Specifically, licensee personnel failed to ensure that the padlock was secured after completing the task [H.12].
Inspection Report# : 2015003 *(pdf)*

**Significance:** G Dec 31, 2014
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Control Access to a High Radiation Area With Dose Rates Greater Than 1 Rem/Hour**
The inspectors reviewed a self-revealing non-cited violation of Technical Specification 5.7.2 because the licensee failed to control access to a high radiation area with dose rates greater than 1 rem/hour. A radiation protection technician assumed responsibility for guarding the area and reestablished compliance with technical specification requirements. Licensee representatives documented the occurrence in the corrective action program and performed an apparent cause evaluation.

The failure to control access to a high radiation area with dose rates greater than 1 rem/hour is a performance deficiency. The requirement not met was Technical Specification 5.7.2. The significance of the performance deficiency was more than minor because, if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern if workers had entered an uncontrolled high radiation area and received unintended radiation dose. The Occupational Radiation Safety Cornerstone was affected; therefore, the inspectors used Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," August 19, 2008, to determine the significance of the violation. The violation had very low safety significance because: (1) It was not an as low as is reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. This violation has a cross cutting aspect in the human performance area, associated with avoiding complacency, because individuals did not recognize and plan for the possibility of mistakes, latent issues, and inherent risk and did not implement appropriate error reduction tools [H.12]. (Section 2RS1)

Inspection Report# : 2014005 *(pdf)*

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

# Miscellaneous

Last modified : December 30, 2015

# Diablo Canyon 2
# 4Q/2015 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Properly Evaluate for Aggregate Impact of Fire Impairments**
The inspectors identified a non-cited violation of Technical Specification 5.4.1.d, "Procedures," for the failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis. Specifically, from August 31 to September 2, 2015, the licensee failed to evaluate the aggregate impact of having three fire doors simultaneously blocked open in adjacent Unit 1 vital battery charger rooms. The licensee implemented immediate corrective actions by assigning a continuous fire watch to the area and documented the issue in the corrective action program as Notification 50826793.

The failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, the failure to evaluate the aggregate impact of multiple fire system impairments affected the licensee ability to limit the impact of a potential fire. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1–Initial Screening and Characterization of Findings." Because the finding involved fire protection, the inspectors transitioned to IMC 0609, Appendix F "Fire Protection Significance Determination Process." The inspectors characterized the finding using IMC 0609, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," dated September 20, 2013. The finding screened as very low safety significance (Green), per Attachment 1, Question 1.4.3-A since the fire finding category was determined to be fire confinement, due to the fire doors being propped open, and the combustion loading on both sides of the door was determined to be a duration of 30 minutes as documented in licensee calculation M-824, "Controlled Combustion Loading Tracking." In addition, the inspectors determined this finding had a cross-cutting aspect in human performance associated with the teamwork component because the licensee's work groups did not properly communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, the work planners did not properly communicate to the fire protection department that all three fire doors would be open at the same time during battery charger load testing. [H.4]
Inspection Report# : 2015004 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Appropriately Scope 230 KV Switchyard into the Maintenance Rule Monitoring Program**
The inspectors identified a Green, non-cited violation of 10 CFR 50.65(b)(2) for the licensee's failure to appropriately scope the 230 kV switchyard in the Maintenance Rule monitoring program. Specifically, from the inception of the facilities' monitoring program through May 18, 2015, the licensee failed to properly scope or evaluate the 230 kV switchyard to include the entire switchyard up through the first inter-tie circuit breakers CB262 and CB282 into the

Maintenance Rule program. Electrical faults within the 230 kV switchyard can cause loss of offsite power which is relied upon to mitigate accidents and cause an actuation of a safety-related systems, such as, emergency diesel generators, and should have been included into its Maintenance Rule program. This issue was entered into the licensee’s corrective action program as Notifications 50702970 and 50703118.

The inspectors determined that the licensee’s failure to scope the 230 kV offsite power source including the switchyard up through the first breakers from the transmission system into the Maintenance Rule program was contrary to the requirements of 10 CFR 50.65 and therefore a performance deficiency. The performance deficiency was determined to be more than minor because it is associated with the initiating events attribute of protections against external factors and adversely affected the cornerstone objective, in that, a 230 kV switchyard failure can upset plant stability and challenge critical safety functions during shutdown as well as power operations. Failure to monitor the performance or condition of 230 kV offsite power source (including the switchyard up through the first breakers from the transmission system) in a manner sufficient to provide reasonable assurance the offsite power was capable of fulfilling the intended functions affected the reliability of the plant equipment to perform their safety function. The inspectors determined if the 230 kV switchyard was properly scoped into the Maintenance Rule program the loss of offsite power due to the flash over event may have been prevented. However the direct cause of the event has been identified as untimely corrective actions associated with an ineffective corrective action program. As such, improper Maintenance Rule scoping was not the direct cause. Therefore, the inspectors determined the finding could be evaluated using the significant determination process in accordance using IMC 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” Exhibit 1, “Initiating Events Screening Questions.” The inspectors determined that the finding was of very low safety significance (Green) because the finding was determined not to be the cause of the actual 230 kV failure such that all of the screening questions in Exhibit 1 could be answered “no.” The inspectors determined that since the scoping of the switchyard systems had occurred more than 3 years ago, and the opportunity to reevaluate system scoping had not recently occurred, the finding did not represent current licensee performance and therefore a cross-cutting aspect was not assigned.
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: Self-Revealing
Item Type: FIN Finding
**High Voltage Insulator Flashover Resulted in Loss of 230 kV Offsite Power and Start of Emergency Diesel Generators**
The inspectors reviewed a self-revealing, Green finding for the licensee’s failure to adequately implement procedure OM7.ID1, Problem Identification and Resolution, to prevent a high voltage insulator flashover event in the 230 kV switchyard that occurred on October 31, 2014. Specifically, corrective actions from three previous root cause evaluations were not effective to prevent a loss of the 230 kV start-up power and subsequent auto start of all of the safety standby emergency diesel generators (EDGs). This issue was entered into the licensee’s corrective action program as Notification 50699230.

The licensee’s failure to adequately implement procedure OM7.ID1, Problem Identification and Resolution was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, this failure resulted in another high-voltage insulator flashover, which resulted in loss of 230 kV offsite startup power and activation of all safety-related EDGs, on October 31, 2014. In accordance with IMC 0609.04, “Initial Characterization of Findings,” the inspectors determined that the impact of the finding on Unit 1 should be evaluated using Exhibit 1 of IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings at Power,” and further determined that this finding required a detailed risk evaluation by the regional senior risk analyst because the finding involved a partial loss of offsite power, a support system that contributes to the likelihood of an initiating event and affected mitigation equipment.

The risk analyst determined that, with the 230 kV system de-energized, any plant transient would result in a plant-centered loss of offsite power. Therefore, the risk analyst calculated the incremental conditional core damage probability for an exposure period of 9 hours to be 2.09 x 10-7, which is lower than the 1 x 10-6 threshold in the significance determination process; this finding is of very low safety significance (Green) for Unit 1. In accordance with IMC 0609.04, “Initial Characterization of Findings,” the inspectors determined that the impact of the finding on Unit 2 should be evaluated using IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” because the finding pertained to operations, an event, or a degraded condition while the plant was shut down. Unit 2 was shutdown in a refueling outage when the event occurred on October 31, 2014. Because of the shutdown configuration of Unit 2, the loss of 230 kV support system did not impact the ability to continue to provide decay heat removal for the unit. Therefore, the analyst determined qualitatively that this finding is also of very low safety significance (Green) for Unit 2. This finding has a cross-cutting aspect of work management, in the area of human performance, for failing to implement a process of planning, controlling, and executing work activities such that nuclear safety is an overriding priority. Specifically the licensee failed to effectively plan and coordinate preventative maintenance strategies associated with root causes from previous high-voltage insulators flashover or failures since 2008 to prevent the loss of offsite 230 kV and the transient on October 31, 2014 [H.5].
Inspection Report# : 2015002 *(pdf)*

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# Mitigating Systems

**Significance:** G Sep 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document an Adequate Evaluation for a Change in Seismic Load Combination Methodology**
The inspectors identified a Severity Level IV, Green, non cited violation of 10 CFR 50.59(d)(1) which requires, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2). Specifically, the licensee changed the method for combining earthquake loads and loss of coolant accident loads from the absolute summation method to square root sum of the squares (SRSS) method without sufficient justification to demonstrate the change did not require prior NRC approval.

The licensee’s failure to implement the requirements of 10 CFR 50.59 and adequately evaluate changes to determine if prior NRC approval is required was a performance deficiency. The licensee entered the issue into the corrective action program as Notification 50811191. In accordance with the licensee’s corrective action program, this issue will be addressed by the licensee through a re-evaluation of the methodology change and the required actions that need to be taken by the licensee will be implemented. Additionally, the licensee performed an operability determination for the affected structures, systems, and components that established a reasonable expectation for operability pending final resolution of the issue.

This performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to determine that use of SRSS in the Watts Bar safety evaluation report cited in the PG&E evaluation represented a change in a method of evaluation, in that the Watts Bar safety evaluation report was very narrow in scope and not appropriate for the intended application at Diablo Canyon. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At

Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not result in the inoperability of the system. Because this performance deficiency had the potential to impact the NRC's ability to perform its regulatory function, the inspectors also evaluated the performance deficiency using traditional enforcement. Since the violation is associated with a Green finding having very low safety significance, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding had a cross cutting aspect in the area of human performance associated with design margins because individuals failed to ensure margins were carefully guarded and changed only through a systematic and rigorous process [H.6].
Inspection Report# : 2015003 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Maintain Operator Licensing Examination Integrity**
The inspectors reviewed a self-revealing, Severity Level IV non-cited violation of 10 CFR 55.49, "Integrity of Examinations and Tests," and an associated Green finding for the licensee's failure to provide adequate examination security measures during administration of the 2015 biennial requalification examination. On May 26, 2015, a licensed operator was able to obtain plant computer information that led to the discovery of specific plant events contained on the NRC-required annual operating test. The licensee entered this issue into the corrective action program as Notification 50704195 and retested the crew with a new scenario.

The failure of the licensee to provide adequate measures for examination security for the biennial requalification examinations was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the human performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 4, Tables 1 and 2 worksheets (issue date June 19, 2012); and the corresponding Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," Flowchart Block #10 (issue date December 6, 2011), the finding was determined to have very low safety significance (Green). Although the 2015 finding resulted in a compromise of the integrity of biennial dynamic simulator examinations had no compensatory actions been taken, the equitable and consistent administration of the biennial dynamic simulator examination was not actually affected by this compromise. The traditional enforcement violation was determined to be a Severity Level IV violation consistent with Section 6.4.d of the Enforcement Policy. This finding has a cross-cutting aspect in the resources component of the human performance cross-cutting area because the licensee failed to ensure the procedures are adequate to ensure nuclear safety [H.1].
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Design Control for High-Energy Line Break Vent Flow Path**
The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the licensee's failure to ensure credited design features, such as flow vent paths, protect safety-related systems, from temperature and pressure effects of a high-energy line break (HELB) in the auxiliary building. Specifically, the licensee allowed obstruction of a credited flow path with acrylic glass plates not qualified in the original design and not verified to function under a HELB scenario. The licensee entered this issue into the corrective action program as Notifications 50697910 and 50698102, and took immediate actions to remove the acrylic glass plates from the vent path doors in the auxiliary building.

The performance deficiency was determined to be more than minor because it affected the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not have adequate measures in place to ensure that qualified components were available to mitigate the consequences of a HELB in the auxiliary building. The finding screened as of very low safety significance (Green) because the finding did not affect the design or qualification of mitigating structures, systems, and components; the finding did not represent a loss of system and/or function; the finding did not represent an actual loss of a function of a single train for greater than the technical specification (TS) allowed outage time; the finding did not represent an actual loss of a function of one or more non-TS trains of equipment; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding was not assigned a cross-cutting aspect since the performance deficiency is not indicative of current plant performance.
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Mar 31, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Provide Adequate Design Review of Emergency Diesel Generator 2-3**
The inspectors documented a self-revealing violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure that design control measures shall provide for verifying or checking the adequacy of design by the performance of design reviews and design control measures shall be applied to items such as maintenance, and repair; and delineation of acceptance criteria for inspections and tests. Specifically, the licensee failed to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1-2 and 1-1 trips and incorporate this modification into the design and installation of emergency diesel generator 2-3.

The licensee's failure to identify that a terminal block cover was removed from the existing diesel generators as corrective actions following previous emergency diesel generator 1 2 and 1 1 trips, and incorporate this modification into the design and installation of emergency diesel generator 2 3, was a performance deficiency. This performance deficiency was more than minor because it is associated with the design control attribute of the Mitigating Systems cornerstone objective and adversely affected the objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the performance deficiency adversely affected the diesel generator's capability to operate loaded for the technical specification required time. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process For Findings At-Power," dated July 1, 2012, the inspectors determined that the finding could not be screened as Green, or very low safety significance due to loss of a function of a single train for greater than its technical specification outage time. As a result, a detailed risk evaluation was performed by a senior risk analyst. The detailed risk evaluation resulted determined that the findings was Green, or very low safety significance.

This finding did not have a cross-cutting aspect because the most significant contributor did not reflect current licensee performance.
Inspection Report# : 2015001 *(pdf)*

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# Barrier Integrity

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Sep 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Secure a Locked High Radiation Area**
The inspectors reviewed a self-revealing non-cited violation (NCV) of Technical Specification 5.4.1(a), “Procedures,” for failure to secure a locked high radiation area. Specifically, the padlock on the Letdown Filter 1-1 locking bar was found unlocked. Upon discovery, the licensee guarded the area until properly secured. This issue was entered into the licensee’s corrective action program as Notification 50710852.

The failure to secure a locked high radiation area was a performance deficiency. The performance deficiency was more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, failure to adequately secure the locked high radiation area could result in unintended exposure to high levels of radiation. Using Inspection Manual Chapter 0609, Appendix C, “Occupational Radiation Safety Significance Determination Process,” dated August 19, 2008, the inspectors determined the violation was of very low safety significance (Green) because: (1) it was not an as low as reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The finding had an avoid complacency cross-cutting aspect, in the area of human performance, because individuals failed to recognize and plan for the possibility of mistakes, even while expecting positive outcomes. Specifically, licensee personnel failed to ensure that the padlock was secured after completing the task [H.12].
Inspection Report# : 2015003 *(pdf)*

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# Public Radiation Safety

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# Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

Last modified : March 01, 2016

# Diablo Canyon 2
# 1Q/2016 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Properly Evaluate for Aggregate Impact of Fire Impairments**
The inspectors identified a non-cited violation of Technical Specification 5.4.1.d, "Procedures," for the failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis. Specifically, from August 31 to September 2, 2015, the licensee failed to evaluate the aggregate impact of having three fire doors simultaneously blocked open in adjacent Unit 1 vital battery charger rooms. The licensee implemented immediate corrective actions by assigning a continuous fire watch to the area and documented the issue in the corrective action program as Notification 50826793.

The failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, the failure to evaluate the aggregate impact of multiple fire system impairments affected the licensee ability to limit the impact of a potential fire. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1–Initial Screening and Characterization of Findings." Because the finding involved fire protection, the inspectors transitioned to IMC 0609, Appendix F "Fire Protection Significance Determination Process." The inspectors characterized the finding using IMC 0609, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," dated September 20, 2013. The finding screened as very low safety significance (Green), per Attachment 1, Question 1.4.3-A since the fire finding category was determined to be fire confinement, due to the fire doors being propped open, and the combustion loading on both sides of the door was determined to be a duration of 30 minutes as documented in licensee calculation M-824, "Controlled Combustion Loading Tracking." In addition, the inspectors determined this finding had a cross-cutting aspect in human performance associated with the teamwork component because the licensee's work groups did not properly communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, the work planners did not properly communicate to the fire protection department that all three fire doors would be open at the same time during battery charger load testing. [H.4]
Inspection Report# : 2015004 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Appropriately Scope 230 KV Switchyard into the Maintenance Rule Monitoring Program**
The inspectors identified a Green, non-cited violation of 10 CFR 50.65(b)(2) for the licensee's failure to appropriately scope the 230 kV switchyard in the Maintenance Rule monitoring program. Specifically, from the inception of the facilities' monitoring program through May 18, 2015, the licensee failed to properly scope or evaluate the 230 kV switchyard to include the entire switchyard up through the first inter-tie circuit breakers CB262 and CB282 into the

Maintenance Rule program. Electrical faults within the 230 kV switchyard can cause loss of offsite power which is relied upon to mitigate accidents and cause an actuation of a safety-related systems, such as, emergency diesel generators, and should have been included into its Maintenance Rule program. This issue was entered into the licensee's corrective action program as Notifications 50702970 and 50703118.

The inspectors determined that the licensee's failure to scope the 230 kV offsite power source including the switchyard up through the first breakers from the transmission system into the Maintenance Rule program was contrary to the requirements of 10 CFR 50.65 and therefore a performance deficiency. The performance deficiency was determined to be more than minor because it is associated with the initiating events attribute of protections against external factors and adversely affected the cornerstone objective, in that, a 230 kV switchyard failure can upset plant stability and challenge critical safety functions during shutdown as well as power operations. Failure to monitor the performance or condition of 230 kV offsite power source (including the switchyard up through the first breakers from the transmission system) in a manner sufficient to provide reasonable assurance the offsite power was capable of fulfilling the intended functions affected the reliability of the plant equipment to perform their safety function. The inspectors determined if the 230 kV switchyard was properly scoped into the Maintenance Rule program the loss of offsite power due to the flash over event may have been prevented. However the direct cause of the event has been identified as untimely corrective actions associated with an ineffective corrective action program. As such, improper Maintenance Rule scoping was not the direct cause. Therefore, the inspectors determined the finding could be evaluated using the significant determination process in accordance using IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions." The inspectors determined that the finding was of very low safety significance (Green) because the finding was determined not to be the cause of the actual 230 kV failure such that all of the screening questions in Exhibit 1 could be answered "no." The inspectors determined that since the scoping of the switchyard systems had occurred more than 3 years ago, and the opportunity to reevaluate system scoping had not recently occurred, the finding did not represent current licensee performance and therefore a cross-cutting aspect was not assigned.
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: Self-Revealing
Item Type: FIN Finding
**High Voltage Insulator Flashover Resulted in Loss of 230 kV Offsite Power and Start of Emergency Diesel Generators**
The inspectors reviewed a self-revealing, Green finding for the licensee's failure to adequately implement procedure OM7.ID1, Problem Identification and Resolution, to prevent a high voltage insulator flashover event in the 230 kV switchyard that occurred on October 31, 2014. Specifically, corrective actions from three previous root cause evaluations were not effective to prevent a loss of the 230 kV start-up power and subsequent auto start of all of the safety standby emergency diesel generators (EDGs). This issue was entered into the licensee's corrective action program as Notification 50699230.

The licensee's failure to adequately implement procedure OM7.ID1, Problem Identification and Resolution was a performance deficiency. The performance deficiency was more than minor because it was associated with the human performance attribute of the Initiating Events cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions. Specifically, this failure resulted in another high-voltage insulator flashover, which resulted in loss of 230 kV offsite startup power and activation of all safety-related EDGs, on October 31, 2014. In accordance with IMC 0609.04, "Initial Characterization of Findings," the inspectors determined that the impact of the finding on Unit 1 should be evaluated using Exhibit 1 of IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings at Power," and further determined that this finding required a detailed risk evaluation by the regional senior risk analyst because the finding involved a partial loss of offsite power, a support system that contributes to the likelihood of an initiating event and affected mitigation equipment.

The risk analyst determined that, with the 230 kV system de-energized, any plant transient would result in a plant-centered loss of offsite power. Therefore, the risk analyst calculated the incremental conditional core damage probability for an exposure period of 9 hours to be 2.09 x 10-7, which is lower than the 1 x 10-6 threshold in the significance determination process; this finding is of very low safety significance (Green) for Unit 1. In accordance with IMC 0609.04, “Initial Characterization of Findings,” the inspectors determined that the impact of the finding on Unit 2 should be evaluated using IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” because the finding pertained to operations, an event, or a degraded condition while the plant was shut down. Unit 2 was shutdown in a refueling outage when the event occurred on October 31, 2014. Because of the shutdown configuration of Unit 2, the loss of 230 kV support system did not impact the ability to continue to provide decay heat removal for the unit. Therefore, the analyst determined qualitatively that this finding is also of very low safety significance (Green) for Unit 2. This finding has a cross-cutting aspect of work management, in the area of human performance, for failing to implement a process of planning, controlling, and executing work activities such that nuclear safety is an overriding priority. Specifically the licensee failed to effectively plan and coordinate preventative maintenance strategies associated with root causes from previous high-voltage insulators flashover or failures since 2008 to prevent the loss of offsite 230 kV and the transient on October 31, 2014 [H.5].
Inspection Report# : 2015002 *(pdf)*

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## Mitigating Systems

**Significance:** G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Verify Adequate Design Airflow for 480 volt AC Switchgear and 125 volt DC Inverter Rooms**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to verify the design adequacy of the safety-related ventilation system for the 480-volt AC switchgear and 125-volt DC inverter rooms. Specifically, the licensee failed to verify sufficient ventilation system airflow to ensure the temperature in rooms housing safety-related electrical equipment remained below 104 degrees Fahrenheit. The licensee’s corrective actions were documented in Notification 50840266.

The failure to provide design control measures to verify the adequacy of the 480-volt AC switchgear and 125-volt DC inverter rooms ventilation system design was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the reduction in airflow to the rooms impacts the reliability of the safety-related equipment ventilation system to maintain the temperatures in these rooms below design limits for the duration of all accident scenarios. Using NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined the finding was of very low safety significance because (1) the finding was not a deficiency affecting the design or qualification of a mitigating system; (2) the finding did not represent a loss of system and/or function; (3) the finding did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time; and (4) the finding does not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program for greater than 24 hours.

The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor of this finding occurred more than three years ago, and is therefore, not representative of current licensee performance.

Inspection Report# : 2016001 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate 480 Vac Motor Starters with Circuit Breaker Trip Settings Higher than Manufacturers' Specifications**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to September 10, 2013, the licensee failed to verify the design of 480 Vac combination motor starter instantaneous magnetic circuit breakers settings, by the use of alternate or simplified calculational methods, for those breakers whose settings are higher than their manufacturers' specifications, as documented in calculation 195B-DC, "MCCB Settings for 460VAC Class 1E Motors," to provide the required level of protection and ensure that certain failures that could be caused by sustained fault currents below the circuit breaker trip setting would not occur. In response to this finding, the licensee conducted a preliminary evaluation of some of the affected equipment and concluded that sustained fault currents below the trip settings are unlikely. This finding was entered into the licensee's corrective action program as Notification 50838071.

The team determined the failure to evaluate 480 Vac combination motor starters with instantaneous magnetic circuit breaker trip current settings higher than their manufacturers' specifications was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper motor starter breaker trip settings could result in a fire in the motor control center cubicle, damage to motor starter components, spurious tripping of the entire motor control center, or lack of protection for downstream components during fault conditions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding did not have a cross-cutting aspect because the most significant causal factor of the performance deficiency did not reflect current licensee performance.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Promptly Correct the Lack of Design Verification of 460 Vac Motors at Maximum Allowable Frequency**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Specifically, prior to March 16, 2016, the licensee failed to assure that the lack of design verification of 460 Vac motors, which could be overloaded at the maximum allowable diesel generator frequency, was promptly corrected after having been identified in a 2013 apparent cause evaluation and again in a 2015 self-assessment as documented in Notifications 50572850 and 50826105, respectively. In response to this finding, the licensee performed a preliminary evaluation of the affected 460 Vac motors and concluded that operation at maximum

emergency diesel generator frequency would not cause them to overheat or trip on overcurrent. This finding was entered into the licensee's corrective action program as Notifications 50835699 and 50838988.

The team determined the failure to correct the lack of design verification of 460 Vac motors at maximum allowable frequency when powered from the emergency diesel generators was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of 460 Vac motors above their rated or analyzed maximum allowable frequencies could result in motor overheating or a trip of the thermal overload relays. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation because the licensee failed to ensure that the organization thoroughly evaluated issues to ensure that resolutions address causes and extent of conditions.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Functionality at Maximum Allowable Voltages**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to February 10, 2016, the licensee failed to verify the design of (1) equipment on the nominally 125 Vdc system at the maximum voltage specified in Procedure OP J-9:IV, “Performing a Battery Equalizing Charge,” and (2) equipment on 480 Vac and 120 Vac vital buses at maximum voltages specified in Procedure OP J-2:VIII, “Guidelines for Reliable Transmission Service for DCPP,” by the use of alternate or simplified calculational methods, to ensure equipment functionality. In response to this finding, the licensee conducted a preliminary evaluation of the affected equipment and concluded that any past exposure to voltages above their maximum rating would not have caused a loss of functionality. This finding was entered into the licensee's corrective action program as Notifications 50834558, 50835906, 50835394, 50835945, 50835949, 50836376, 50836439, 50836638, 50836872, and 50836995.

The team determined the failure to evaluate operation of 125 Vdc and 480 and 120 Vac equipment at maximum allowable voltages was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of equipment outside of its rated or analyzed maximum allowable voltages adversely affects the reliability and capability of that equipment required to perform safety-related functions. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Extent of Condition for a Degraded Condition on a Nonsafety-Related 4160 Vac Breaker**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” Specifically, in October of 2015, the licensee failed to evaluate the extent of condition of a cracked holding pawl on a nonsafety-related 4160 Vac SF6 breaker, which was procured as safety-related, in accordance with Procedure OM7.ID1, “Problem Identification and Resolution,” when the failure of the component could adversely impact safety-related breakers of the same make and model. In response to this finding, the licensee is performing a procedure review to include steps to perform an extent of condition analysis for unplanned nonsafety-related equipment issues that may also affect similar safety-related equipment. This finding was entered into the licensee's corrective action program as Notifications 50836859 and 50836689.

The team determined the failure to evaluate the impact of a cracked holding pawl identified on a nonsafety-related 4160 Vac SF6 breaker on additional safety-related 4160 Vac SF6 breakers was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 4160 Vac breaker with the cracked holding pawl was procured as safety-related; therefore, the condition extends to safety-related 4160 Vac breakers of the same make and model and potentially adversely affects the ability to perform their safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that individuals used decision-making practices that emphasized prudent choices.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Voltage Effects of Limiting Design Basis Events on the 230 kV Offsite Power Circuit**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to January 30, 2014, the licensee failed to verify the design of the 230 kV preferred offsite power source, such as by the performance of design reviews or use of alternate or simplified calculational methods, by assuming in calculation 359-DC, “Determination of 230 kV Grid Capability

Limits as DCPP Offsite Power Source," that the reactor trip and engineered safety features actuation system signals are coincident in time for all postulated design basis events. However, the plant is designed such that, during some events, the signals are separate in time and would result in a greater vital bus voltage depression than analyzed. In response to this finding, the licensee conducted a preliminary evaluation and concluded that the current transmission grid conditions were such that the calculation criteria would be met in the event of a design basis event involving non-coincident reactor trip and engineered safety features actuation system signals. This finding was entered into the licensee's corrective action program as Notification 50839137.

The team determined the failure to evaluate the voltage effects of a limiting design basis event with non-coincident reactor trip and engineered safety features actuation system signals on the 230 kV offsite power circuit was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure adequate bus voltages as a result of a design basis event with non-coincident reactor trip and engineered safety features actuation system signals would result in a trip of the undervoltage relays and the loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Translate Appropriate Load Tap Changer Timing Acceptance Criteria into Periodic Tests**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished." Specifically, prior to November 25, 2015, the licensee failed to include appropriate quantitative acceptance criteria in Procedure MP E-62.3, "Tap Changer Functional Test for Standby-Startup Transformer 11," to ensure that the load tap changer speed for standby-startup transformer 11 was adequate to restore vital bus voltages to the required level during design basis events. In response to this finding, the licensee performed a preliminary evaluation of the condition and concluded that the most recently measured speed of the load tap changer was adequate to ensure that it would restore vital bus voltage within the required time. This finding was entered into the licensee's corrective action program as Notification 50839333.

The team determined the failure to translate appropriate load tap changer timing acceptance criteria into functional tests to ensure that design assumptions were being maintained was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the load tap changer could meet its functional test acceptance criterion, but not operate fast enough to restore vital bus voltages within the required time during design basis events, which would result in an undervoltage trip and loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Sep 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document an Adequate Evaluation for a Change in Seismic Load Combination Methodology**
The inspectors identified a Severity Level IV, Green, non cited violation of 10 CFR 50.59(d)(1) which requires, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2). Specifically, the licensee changed the method for combining earthquake loads and loss of coolant accident loads from the absolute summation method to square root sum of the squares (SRSS) method without sufficient justification to demonstrate the change did not require prior NRC approval.

The licensee's failure to implement the requirements of 10 CFR 50.59 and adequately evaluate changes to determine if prior NRC approval is required was a performance deficiency. The licensee entered the issue into the corrective action program as Notification 50811191. In accordance with the licensee's corrective action program, this issue will be addressed by the licensee through a re-evaluation of the methodology change and the required actions that need to be taken by the licensee will be implemented. Additionally, the licensee performed an operability determination for the affected structures, systems, and components that established a reasonable expectation for operability pending final resolution of the issue.

This performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to determine that use of SRSS in the Watts Bar safety evaluation report cited in the PG&E evaluation represented a change in a method of evaluation, in that the Watts Bar safety evaluation report was very narrow in scope and not appropriate for the intended application at Diablo Canyon. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not result in the inoperability of the system. Because this performance deficiency had the potential to impact the NRC's ability to perform its regulatory function, the inspectors also evaluated the performance deficiency using traditional enforcement. Since the violation is associated with a Green finding having very low safety significance, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding had a cross cutting aspect in the area of human performance associated with design margins because individuals failed to ensure margins were carefully guarded and changed only through a systematic and rigorous process [H.6].
Inspection Report# : 2015003 *(pdf)*

**Significance:** G Jun 30, 2015

Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Maintain Operator Licensing Examination Integrity**
The inspectors reviewed a self-revealing, Severity Level IV non-cited violation of 10 CFR 55.49, “Integrity of Examinations and Tests,” and an associated Green finding for the licensee’s failure to provide adequate examination security measures during administration of the 2015 biennial requalification examination. On May 26, 2015, a licensed operator was able to obtain plant computer information that led to the discovery of specific plant events contained on the NRC-required annual operating test. The licensee entered this issue into the corrective action program as Notification 50704195 and retested the crew with a new scenario.

The failure of the licensee to provide adequate measures for examination security for the biennial requalification examinations was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it adversely affected the human performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using NRC Inspection Manual Chapter 0609, “Significance Determination Process,” Attachment 4, Tables 1 and 2 worksheets (issue date June 19, 2012); and the corresponding Appendix I, “Licensed Operator Requalification Significance Determination Process (SDP),” Flowchart Block #10 (issue date December 6, 2011), the finding was determined to have very low safety significance (Green). Although the 2015 finding resulted in a compromise of the integrity of biennial dynamic simulator examinations had no compensatory actions been taken, the equitable and consistent administration of the biennial dynamic simulator examination was not actually affected by this compromise. The traditional enforcement violation was determined to be a Severity Level IV violation consistent with Section 6.4.d of the Enforcement Policy. This finding has a cross-cutting aspect in the resources component of the human performance cross-cutting area because the licensee failed to ensure the procedures are adequate to ensure nuclear safety [H.1].
Inspection Report# : 2015002 *(pdf)*

**Significance:** G Jun 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Inadequate Design Control for High-Energy Line Break Vent Flow Path**
The inspectors identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving the licensee’s failure to ensure credited design features, such as flow vent paths, protect safety-related systems, from temperature and pressure effects of a high-energy line break (HELB) in the auxiliary building. Specifically, the licensee allowed obstruction of a credited flow path with acrylic glass plates not qualified in the original design and not verified to function under a HELB scenario. The licensee entered this issue into the corrective action program as Notifications 50697910 and 50698102, and took immediate actions to remove the acrylic glass plates from the vent path doors in the auxiliary building.

The performance deficiency was determined to be more than minor because it affected the Mitigating Systems Cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring the reliability, availability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not have adequate measures in place to ensure that qualified components were available to mitigate the consequences of a HELB in the auxiliary building. The finding screened as of very low safety significance (Green) because the finding did not affect the design or qualification of mitigating structures, systems, and components; the finding did not represent a loss of system and/or function; the finding did not represent an actual loss of a function of a single train for greater than the technical specification (TS) allowed outage time; the finding did not represent an actual loss of a function of one or more non-TS trains of equipment; and did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The finding was not assigned a cross-cutting aspect since the performance deficiency is not indicative of current plant performance.
Inspection Report# : 2015002 *(pdf)*

# Barrier Integrity

# Emergency Preparedness

# Occupational Radiation Safety

**Significance:** G Sep 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Secure a Locked High Radiation Area**
The inspectors reviewed a self-revealing non-cited violation (NCV) of Technical Specification 5.4.1(a), "Procedures," for failure to secure a locked high radiation area. Specifically, the padlock on the Letdown Filter 1-1 locking bar was found unlocked. Upon discovery, the licensee guarded the area until properly secured. This issue was entered into the licensee's corrective action program as Notification 50710852.

The failure to secure a locked high radiation area was a performance deficiency. The performance deficiency was more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, failure to adequately secure the locked high radiation area could result in unintended exposure to high levels of radiation. Using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, the inspectors determined the violation was of very low safety significance (Green) because: (1) it was not an as low as reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The finding had an avoid complacency cross-cutting aspect, in the area of human performance, because individuals failed to recognize and plan for the possibility of mistakes, even while expecting positive outcomes. Specifically, licensee personnel failed to ensure that the padlock was secured after completing the task [H.12].
Inspection Report# : 2015003 *(pdf)*

# Public Radiation Safety

# Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

# Miscellaneous

Last modified : July 11, 2016

# Diablo Canyon 2
# 2Q/2016 Plant Inspection Findings

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## Initiating Events

**Significance:** G Dec 31, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Properly Evaluate for Aggregate Impact of Fire Impairments**
The inspectors identified a non-cited violation of Technical Specification 5.4.1.d, "Procedures," for the failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis. Specifically, from August 31 to September 2, 2015, the licensee failed to evaluate the aggregate impact of having three fire doors simultaneously blocked open in adjacent Unit 1 vital battery charger rooms. The licensee implemented immediate corrective actions by assigning a continuous fire watch to the area and documented the issue in the corrective action program as Notification 50826793.

The failure to follow approved fire protection program procedures to review the fire impairments list to assess the aggregate impact on the fire protection design and safe shutdown analysis was a performance deficiency. This performance deficiency was more than minor, and therefore a finding, because it was associated with the Initiating Events cornerstone attribute of Protection against External Factors (Fire) and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during plant operations. Specifically, the failure to evaluate the aggregate impact of multiple fire system impairments affected the licensee ability to limit the impact of a potential fire. The inspectors evaluated the finding using IMC 0609, Attachment 4, "Phase 1–Initial Screening and Characterization of Findings." Because the finding involved fire protection, the inspectors transitioned to IMC 0609, Appendix F "Fire Protection Significance Determination Process." The inspectors characterized the finding using IMC 0609, Appendix F, Attachment 1, "Fire Protection SDP Phase 1 Worksheet," dated September 20, 2013. The finding screened as very low safety significance (Green), per Attachment 1, Question 1.4.3-A since the fire finding category was determined to be fire confinement, due to the fire doors being propped open, and the combustion loading on both sides of the door was determined to be a duration of 30 minutes as documented in licensee calculation M-824, "Controlled Combustion Loading Tracking." In addition, the inspectors determined this finding had a cross-cutting aspect in human performance associated with the teamwork component because the licensee's work groups did not properly communicate and coordinate their activities within and across organizational boundaries to ensure nuclear safety was maintained. Specifically, the work planners did not properly communicate to the fire protection department that all three fire doors would be open at the same time during battery charger load testing. [H.4]
Inspection Report# : 2015004 *(pdf)*

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## Mitigating Systems

**Significance:** G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Verify Adequate Design Airflow for 480 volt AC Switchgear and 125 volt DC Inverter Rooms**

The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to verify the design adequacy of the safety-related ventilation system for the 480-volt AC switchgear and 125-volt DC inverter rooms. Specifically, the licensee failed to verify sufficient ventilation system airflow to ensure the temperature in rooms housing safety-related electrical equipment remained below 104 degrees Fahrenheit. The licensee’s corrective actions were documented in Notification 50840266.

The failure to provide design control measures to verify the adequacy of the 480-volt AC switchgear and 125-volt DC inverter rooms ventilation system design was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the reduction in airflow to the rooms impacts the reliability of the safety-related equipment ventilation system to maintain the temperatures in these rooms below design limits for the duration of all accident scenarios. Using NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined the finding was of very low safety significance because (1) the finding was not a deficiency affecting the design or qualification of a mitigating system; (2) the finding did not represent a loss of system and/or function; (3) the finding did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time; and (4) the finding does not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program for greater than 24 hours.

The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor of this finding occurred more than three years ago, and is therefore, not representative of current licensee performance.
Inspection Report# : 2016001 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate 480 Vac Motor Starters with Circuit Breaker Trip Settings Higher than Manufacturers’ Specifications**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to September 10, 2013, the licensee failed to verify the design of 480 Vac combination motor starter instantaneous magnetic circuit breakers settings, by the use of alternate or simplified calculational methods, for those breakers whose settings are higher than their manufacturers' specifications, as documented in calculation 195B-DC, “MCCB Settings for 460VAC Class 1E Motors,” to provide the required level of protection and ensure that certain failures that could be caused by sustained fault currents below the circuit breaker trip setting would not occur. In response to this finding, the licensee conducted a preliminary evaluation of some of the affected equipment and concluded that sustained fault currents below the trip settings are unlikely. This finding was entered into the licensee's corrective action program as Notification 50838071.

The team determined the failure to evaluate 480 Vac combination motor starters with instantaneous magnetic circuit breaker trip current settings higher than their manufacturers' specifications was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper motor starter breaker trip settings could result in a fire in the motor control center cubicle, damage to motor starter components, spurious tripping of the entire motor control center, or lack of protection for downstream components during fault conditions. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding did not have a cross-cutting aspect because the most significant causal factor of the performance deficiency did not reflect current licensee performance.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Promptly Correct the Lack of Design Verification of 460 Vac Motors at Maximum Allowable Frequency**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," which states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." Specifically, prior to March 16, 2016, the licensee failed to assure that the lack of design verification of 460 Vac motors, which could be overloaded at the maximum allowable diesel generator frequency, was promptly corrected after having been identified in a 2013 apparent cause evaluation and again in a 2015 self-assessment as documented in Notifications 50572850 and 50826105, respectively. In response to this finding, the licensee performed a preliminary evaluation of the affected 460 Vac motors and concluded that operation at maximum emergency diesel generator frequency would not cause them to overheat or trip on overcurrent. This finding was entered into the licensee's corrective action program as Notifications 50835699 and 50838988.

The team determined the failure to correct the lack of design verification of 460 Vac motors at maximum allowable frequency when powered from the emergency diesel generators was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of 460 Vac motors above their rated or analyzed maximum allowable frequencies could result in motor overheating or a trip of the thermal overload relays. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation because the licensee failed to ensure that the organization thoroughly evaluated issues to ensure that resolutions address causes and extent of conditions.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Functionality at Maximum Allowable Voltages**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design,

such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to February 10, 2016, the licensee failed to verify the design of (1) equipment on the nominally 125 Vdc system at the maximum voltage specified in Procedure OP J-9:IV, "Performing a Battery Equalizing Charge," and (2) equipment on 480 Vac and 120 Vac vital buses at maximum voltages specified in Procedure OP J-2:VIII, "Guidelines for Reliable Transmission Service for DCPP," by the use of alternate or simplified calculational methods, to ensure equipment functionality. In response to this finding, the licensee conducted a preliminary evaluation of the affected equipment and concluded that any past exposure to voltages above their maximum rating would not have caused a loss of functionality. This finding was entered into the licensee's corrective action program as Notifications 50834558, 50835906, 50835394, 50835945, 50835949, 50836376, 50836439, 50836638, 50836872, and 50836995.

The team determined the failure to evaluate operation of 125 Vdc and 480 and 120 Vac equipment at maximum allowable voltages was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of equipment outside of its rated or analyzed maximum allowable voltages adversely affects the reliability and capability of that equipment required to perform safety-related functions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Extent of Condition for a Degraded Condition on a Nonsafety-Related 4160 Vac Breaker**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Specifically, in October of 2015, the licensee failed to evaluate the extent of condition of a cracked holding pawl on a nonsafety-related 4160 Vac SF6 breaker, which was procured as safety-related, in accordance with Procedure OM7.ID1, "Problem Identification and Resolution," when the failure of the component could adversely impact safety-related breakers of the same make and model. In response to this finding, the licensee is performing a procedure review to include steps to perform an extent of condition analysis for unplanned nonsafety-related equipment issues that may also affect similar safety-related equipment. This finding was entered into the licensee's corrective action program as Notifications 50836859 and 50836689.

The team determined the failure to evaluate the impact of a cracked holding pawl identified on a nonsafety-related 4160 Vac SF6 breaker on additional safety-related 4160 Vac SF6 breakers was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 4160 Vac breaker with the cracked holding pawl was procured as safety-related; therefore, the condition extends to safety-related 4160 Vac breakers of the same make and model and potentially adversely affects the ability to perform their safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that individuals used decision-making practices that emphasized prudent choices.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Voltage Effects of Limiting Design Basis Events on the 230 kV Offsite Power Circuit**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to January 30, 2014, the licensee failed to verify the design of the 230 kV preferred offsite power source, such as by the performance of design reviews or use of alternate or simplified calculational methods, by assuming in calculation 359-DC, "Determination of 230 kV Grid Capability Limits as DCPP Offsite Power Source," that the reactor trip and engineered safety features actuation system signals are coincident in time for all postulated design basis events. However, the plant is designed such that, during some events, the signals are separate in time and would result in a greater vital bus voltage depression than analyzed. In response to this finding, the licensee conducted a preliminary evaluation and concluded that the current transmission grid conditions were such that the calculation criteria would be met in the event of a design basis event involving non-coincident reactor trip and engineered safety features actuation system signals. This finding was entered into the licensee's corrective action program as Notification 50839137.

The team determined the failure to evaluate the voltage effects of a limiting design basis event with non-coincident reactor trip and engineered safety features actuation system signals on the 230 kV offsite power circuit was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure adequate bus voltages as a result of a design basis event with non-coincident reactor trip and engineered safety features actuation system signals would result in a trip of the undervoltage relays and the loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Translate Appropriate Load Tap Changer Timing Acceptance Criteria into Periodic Tests**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.” Specifically, prior to November 25, 2015, the licensee failed to include appropriate quantitative acceptance criteria in Procedure MP E-62.3, “Tap Changer Functional Test for Standby-Startup Transformer 11,” to ensure that the load tap changer speed for standby-startup transformer 11 was adequate to restore vital bus voltages to the required level during design basis events. In response to this finding, the licensee performed a preliminary evaluation of the condition and concluded that the most recently measured speed of the load tap changer was adequate to ensure that it would restore vital bus voltage within the required time. This finding was entered into the licensee's corrective action program as Notification 50839333.

The team determined the failure to translate appropriate load tap changer timing acceptance criteria into functional tests to ensure that design assumptions were being maintained was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the load tap changer could meet its functional test acceptance criterion, but not operate fast enough to restore vital bus voltages within the required time during design basis events, which would result in an undervoltage trip and loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Sep 30, 2015
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Document an Adequate Evaluation for a Change in Seismic Load Combination Methodology**
The inspectors identified a Severity Level IV, Green, non cited violation of 10 CFR 50.59(d)(1) which requires, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant to paragraph (c) of this section. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2). Specifically, the licensee changed the method for combining earthquake loads and loss of coolant accident loads from the absolute summation method to square root sum of the squares (SRSS) method without sufficient justification to demonstrate the change did not require prior NRC approval.

The licensee’s failure to implement the requirements of 10 CFR 50.59 and adequately evaluate changes to determine if prior NRC approval is required was a performance deficiency. The licensee entered the issue into the corrective action program as Notification 50811191. In accordance with the licensee’s corrective action program, this issue will be addressed by the licensee through a re-evaluation of the methodology change and the required actions that need to be taken by the licensee will be implemented. Additionally, the licensee performed an operability determination for the affected structures, systems, and components that established a reasonable expectation for operability pending final resolution of the issue.

This performance deficiency was more than minor, and therefore a finding, because it was associated with the design control attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to determine that use of SRSS in the Watts Bar safety evaluation report cited in the PG&E evaluation represented a change in a method of evaluation, in that the Watts Bar safety evaluation report was very narrow in scope and not appropriate for the intended application at Diablo Canyon. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At Power," dated June 19, 2012, Exhibit 2, "Mitigating Systems Screening Questions," the issue screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not result in the inoperability of the system. Because this performance deficiency had the potential to impact the NRC's ability to perform its regulatory function, the inspectors also evaluated the performance deficiency using traditional enforcement. Since the violation is associated with a Green finding having very low safety significance, the traditional enforcement violation was determined to be a Severity Level IV violation, consistent with the example in paragraph 6.1.d(2) of the NRC Enforcement Policy. This finding had a cross cutting aspect in the area of human performance associated with design margins because individuals failed to ensure margins were carefully guarded and changed only through a systematic and rigorous process [H.6].
Inspection Report# : 2015003 *(pdf)*

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# Barrier Integrity

**Significance:** G Jun 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Misplaced Spent Fuel Assembly in the Spent Fuel Pool**
The inspectors reviewed a self-revealed, non-cited violation of Technical Specification (TS) 5.4.1.a, "Procedures," for the licensee's failure to place a spent fuel assembly in its correct location in the spent fuel pool (SFP) in accordance with Procedure OP B-8H, "Spent Fuel Pool Work Instructions." Specifically, the fuel handling crew moved spent fuel assembly TT69 to location E-37 rather than its intended location E-27. In response to this error, reactor engineering performed a technical specification verification in order to ensure that fuel assembly TT69 could remain in Cell E-37. The licensee suspended further fuel movements pending corrective action and remediation of the operators. The licensee entered this into the corrective action program as Notifications 50846834 and 50847067.

The licensee's failure to place a spent fuel assembly in its correct location in the SFP was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the configuration control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because: (1) the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, (2) the finding did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, (3) the finding did not result in a loss

of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and (4) the finding did not affect the SFP neutron absorber, fuel bundle misplacement (i.e., fuel loading pattern error) or soluble Boron concentration. This finding had a cross-cutting aspect in the area of human performance associated with avoiding complacency. Specifically, individuals failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and individuals failed to implement appropriate error reduction tools. [H.12]
Inspection Report# : 2016002 *(pdf)*

**Significance:** G May 11, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Insufficient procedural direction contained within EOP E-2, Faulted Steam Generator Isolation**
The examiners identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Procedure EOP E-2, "Faulted Steam Generator Isolation," does not contain sufficient procedural direction for isolating auxiliary feedwater flow to a faulted steam generator in the event that auxiliary feedwater control valves cannot be closed from the control room. Procedure EOP E-2, Appendix HH, "Isolated Faulted Steam Generator," Step 1.d, and its associated column, Response Not Obtained, does not ensure that a faulted steam generator would remain isolated under all conditions. The Response Not Obtained column permits operators to either locally close auxiliary feedwater control valves OR secure the auxiliary feedwater pump feeding the faulted steam generator. However, due to the absence of pull-to-lock or hard stop switches for the auxiliary feedwater pumps, the possibility exists for an automatic restart of an auxiliary feedwater pump and a re-initiation of feedwater to a faulted steam generator.

The failure to ensure that Procedure EOP E-2 contained sufficient direction to isolate a faulted steam generator when auxiliary feedwater flow control valves cannot be closed from the control room was a performance deficiency. This performance deficiency was of more than minor safety significance because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone (reactor coolant system and containment) and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the re-initiation of feedwater to an isolated, faulted steam generator has the potential to adversely affect the reactor coolant system barrier by causing an additional unintended cooldown of the reactor coolant system, increased potential for pressurized thermal shock, and thermal stress to the steam generator u-tubes. Additionally, the containment barrier would be affected by the re-initiation of feedwater to a steam line break within containment. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the team determined that the finding required a detailed risk evaluation due to the potential to affect the reactor coolant system boundary. A senior reactor analyst performed a bounding detailed risk evaluation and estimated the maximum increase in core damage frequency to be 5.9E-8/year, and therefore the finding was determined to be of very low safety significance (Green). This increase in core damage frequency was mitigated by the low probability of multiple equipment failures in the auxiliary feedwater system when combined with the low initiating event frequency of a faulted steam generator. Because the violation was of very low safety significance (Green) and the issue was entered into the licensee's corrective action program as Notification 50847218, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000275/2016301; 05000323/2016301-01, "Insufficient Procedural Direction Contained Within E-2, Faulted Steam Generator Isolation." This finding has a cross-cutting aspect in the area of human performance associated with resources because the organization did not ensure procedures are available and adequate to support nuclear safety.
Inspection Report# : 2016301 *(pdf)*

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# Emergency Preparedness

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# Occupational Radiation Safety

**Significance:** G Sep 30, 2015
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Failure to Secure a Locked High Radiation Area**
The inspectors reviewed a self-revealing non-cited violation (NCV) of Technical Specification 5.4.1(a), "Procedures," for failure to secure a locked high radiation area. Specifically, the padlock on the Letdown Filter 1-1 locking bar was found unlocked. Upon discovery, the licensee guarded the area until properly secured. This issue was entered into the licensee's corrective action program as Notification 50710852.

The failure to secure a locked high radiation area was a performance deficiency. The performance deficiency was more than minor because, if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, failure to adequately secure the locked high radiation area could result in unintended exposure to high levels of radiation. Using Inspection Manual Chapter 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, the inspectors determined the violation was of very low safety significance (Green) because: (1) it was not an as low as reasonably achievable (ALARA) finding, (2) there was no overexposure, (3) there was no substantial potential for an overexposure, and (4) the ability to assess dose was not compromised. The finding had an avoid complacency cross-cutting aspect, in the area of human performance, because individuals failed to recognize and plan for the possibility of mistakes, even while expecting positive outcomes. Specifically, licensee personnel failed to ensure that the padlock was secured after completing the task [H.12].
Inspection Report# : 2015003 *(pdf)*

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# Public Radiation Safety

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# Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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# Miscellaneous

Last modified : August 29, 2016

# Diablo Canyon 2
# 4Q/2016 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** G Jul 14, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Inadequate Maintenance Procedure affected the Performance of Safety-Related Emergency Diesel Generator**
The inspectors assessed a self-revealed, non-cited violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to implement properly preplanned maintenance procedures that affected the performance of safety-related equipment. Specifically, two maintenance procedures associated with the emergency diesel generators' fuel injectors lacked adequate details on specific key mechanical parameters (capscrew bolt torque setup and fuel injection pump alignment) to ensure that maintenance activities were performed in a manner adequate to the circumstances. In both examples, the licensee entered the issues into the corrective action program and corrected the condition to restore the emergency diesel generators to an operable status.

This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At Power," issued June 19, 2012, the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent the loss of a system or function, the loss of a train of a technical specification safety system for greater than its allowed outage time, or the loss of a non-technical specification high-safety-significant system for greater than 24 hours. This finding had a cross-cutting aspect in the area of human performance associated with work management – "organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority." Specifically, work on the emergency diesel generators fuel oil system components was not effectively planned and executed by incorporating conditions to ensure a successful outcome [H.5].

Inspection Report# : 2016009 *(pdf)*

**Significance:** G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Verify Adequate Design Airflow for 480 volt AC Switchgear and 125 volt DC Inverter Rooms**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify the design adequacy of the safety-related ventilation system for the 480-volt AC switchgear and 125-volt DC inverter rooms. Specifically, the licensee failed to verify sufficient ventilation system airflow to ensure the temperature in rooms housing safety-related electrical equipment remained below 104 degrees Fahrenheit. The licensee's corrective actions were documented in Notification 50840266.

The failure to provide design control measures to verify the adequacy of the 480-volt AC switchgear and 125-volt DC inverter rooms ventilation system design was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the reduction in airflow to the rooms impacts the reliability of the safety-related equipment ventilation system to maintain the temperatures in these rooms below design limits for the duration of all accident scenarios. Using NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined the finding was of very low safety significance because (1) the finding was not a deficiency affecting the design or qualification of a mitigating system; (2) the finding did not represent a loss of system and/or function; (3) the finding did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time; and (4) the finding does not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program for greater than 24 hours.

The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor of this finding occurred more than three years ago, and is therefore, not representative of current licensee performance.
Inspection Report# : 2016001 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate 480 Vac Motor Starters with Circuit Breaker Trip Settings Higher than Manufacturers’ Specifications**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to September 10, 2013, the licensee failed to verify the design of 480 Vac combination motor starter instantaneous magnetic circuit breakers settings, by the use of alternate or simplified calculational methods, for those breakers whose settings are higher than their manufacturers' specifications, as documented in calculation 195B-DC, “MCCB Settings for 460VAC Class 1E Motors,” to provide the required level of protection and ensure that certain failures that could be caused by sustained fault currents below the circuit breaker trip setting would not occur. In response to this finding, the licensee conducted a preliminary evaluation of some of the affected equipment and concluded that sustained fault currents below the trip settings are unlikely. This finding was entered into the licensee's corrective action program as Notification 50838071.

The team determined the failure to evaluate 480 Vac combination motor starters with instantaneous magnetic circuit breaker trip current settings higher than their manufacturers' specifications was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper motor starter breaker trip settings could result in a fire in the motor control center cubicle, damage to motor starter components, spurious tripping of the entire motor control center, or lack of protection for downstream components during fault conditions. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding did not have a cross-cutting aspect because the most significant causal factor of the performance deficiency did not reflect current licensee performance.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Promptly Correct the Lack of Design Verification of 460 Vac Motors at Maximum Allowable Frequency**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” which states, in part, “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Specifically, prior to March 16, 2016, the licensee failed to assure that the lack of design verification of 460 Vac motors, which could be overloaded at the maximum allowable diesel generator frequency, was promptly corrected after having been identified in a 2013 apparent cause evaluation and again in a 2015 self-assessment as documented in Notifications 50572850 and 50826105, respectively. In response to this finding, the licensee performed a preliminary evaluation of the affected 460 Vac motors and concluded that operation at maximum emergency diesel generator frequency would not cause them to overheat or trip on overcurrent. This finding was entered into the licensee's corrective action program as Notifications 50835699 and 50838988.

The team determined the failure to correct the lack of design verification of 460 Vac motors at maximum allowable frequency when powered from the emergency diesel generators was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of 460 Vac motors above their rated or analyzed maximum allowable frequencies could result in motor overheating or a trip of the thermal overload relays. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation because the licensee failed to ensure that the organization thoroughly evaluated issues to ensure that resolutions address causes and extent of conditions.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Functionality at Maximum Allowable Voltages**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to February 10, 2016, the licensee failed to verify the design of (1) equipment on the nominally 125 Vdc system at the maximum voltage specified in Procedure OP J-9:IV, “Performing a Battery Equalizing Charge,” and (2) equipment on 480 Vac and 120 Vac vital buses at maximum voltages specified in Procedure OP J-2:VIII, “Guidelines for Reliable Transmission Service for DCPP,” by the use of alternate or simplified calculational methods, to ensure equipment functionality. In response to this finding, the licensee

conducted a preliminary evaluation of the affected equipment and concluded that any past exposure to voltages above their maximum rating would not have caused a loss of functionality. This finding was entered into the licensee's corrective action program as Notifications 50834558, 50835906, 50835394, 50835945, 50835949, 50836376, 50836439, 50836638, 50836872, and 50836995.

The team determined the failure to evaluate operation of 125 Vdc and 480 and 120 Vac equipment at maximum allowable voltages was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of equipment outside of its rated or analyzed maximum allowable voltages adversely affects the reliability and capability of that equipment required to perform safety-related functions. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Extent of Condition for a Degraded Condition on a Nonsafety-Related 4160 Vac Breaker**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings." Specifically, in October of 2015, the licensee failed to evaluate the extent of condition of a cracked holding pawl on a nonsafety-related 4160 Vac SF6 breaker, which was procured as safety-related, in accordance with Procedure OM7.ID1, "Problem Identification and Resolution," when the failure of the component could adversely impact safety-related breakers of the same make and model. In response to this finding, the licensee is performing a procedure review to include steps to perform an extent of condition analysis for unplanned nonsafety-related equipment issues that may also affect similar safety-related equipment. This finding was entered into the licensee's corrective action program as Notifications 50836859 and 50836689.

The team determined the failure to evaluate the impact of a cracked holding pawl identified on a nonsafety-related 4160 Vac SF6 breaker on additional safety-related 4160 Vac SF6 breakers was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 4160 Vac breaker with the cracked holding pawl was procured as safety-related; therefore, the condition extends to safety-related 4160 Vac breakers of the same make and model and potentially adversely affects the ability to perform their safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that individuals used decision-making practices that emphasized prudent choices.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Voltage Effects of Limiting Design Basis Events on the 230 kV Offsite Power Circuit**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to January 30, 2014, the licensee failed to verify the design of the 230 kV preferred offsite power source, such as by the performance of design reviews or use of alternate or simplified calculational methods, by assuming in calculation 359-DC, "Determination of 230 kV Grid Capability Limits as DCPP Offsite Power Source," that the reactor trip and engineered safety features actuation system signals are coincident in time for all postulated design basis events. However, the plant is designed such that, during some events, the signals are separate in time and would result in a greater vital bus voltage depression than analyzed. In response to this finding, the licensee conducted a preliminary evaluation and concluded that the current transmission grid conditions were such that the calculation criteria would be met in the event of a design basis event involving non-coincident reactor trip and engineered safety features actuation system signals. This finding was entered into the licensee's corrective action program as Notification 50839137.

The team determined the failure to evaluate the voltage effects of a limiting design basis event with non-coincident reactor trip and engineered safety features actuation system signals on the 230 kV offsite power circuit was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure adequate bus voltages as a result of a design basis event with non-coincident reactor trip and engineered safety features actuation system signals would result in a trip of the undervoltage relays and the loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Translate Appropriate Load Tap Changer Timing Acceptance Criteria into Periodic Tests**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which states, in part, "Instructions, procedures, or drawings shall include appropriate

quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.” Specifically, prior to November 25, 2015, the licensee failed to include appropriate quantitative acceptance criteria in Procedure MP E-62.3, “Tap Changer Functional Test for Standby-Startup Transformer 11,” to ensure that the load tap changer speed for standby-startup transformer 11 was adequate to restore vital bus voltages to the required level during design basis events. In response to this finding, the licensee performed a preliminary evaluation of the condition and concluded that the most recently measured speed of the load tap changer was adequate to ensure that it would restore vital bus voltage within the required time. This finding was entered into the licensee's corrective action program as Notification 50839333.

The team determined the failure to translate appropriate load tap changer timing acceptance criteria into functional tests to ensure that design assumptions were being maintained was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the load tap changer could meet its functional test acceptance criterion, but not operate fast enough to restore vital bus voltages within the required time during design basis events, which would result in an undervoltage trip and loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

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## Barrier Integrity

**Significance:** G Jun 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Misplaced Spent Fuel Assembly in the Spent Fuel Pool**
The inspectors reviewed a self-revealed, non-cited violation of Technical Specification (TS) 5.4.1.a, "Procedures," for the licensee’s failure to place a spent fuel assembly in its correct location in the spent fuel pool (SFP) in accordance with Procedure OP B-8H, "Spent Fuel Pool Work Instructions." Specifically, the fuel handling crew moved spent fuel assembly TT69 to location E-37 rather than its intended location E-27. In response to this error, reactor engineering performed a technical specification verification in order to ensure that fuel assembly TT69 could remain in Cell E-37. The licensee suspended further fuel movements pending corrective action and remediation of the operators. The licensee entered this into the corrective action program as Notifications 50846834 and 50847067.

The licensee’s failure to place a spent fuel assembly in its correct location in the SFP was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the configuration control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 3, "Barrier Integrity Screening

Questions," the inspectors determined that the finding was of very low safety significance (Green) because: (1) the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, (2) the finding did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, (3) the finding did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and (4) the finding did not affect the SFP neutron absorber, fuel bundle misplacement (i.e., fuel loading pattern error) or soluble Boron concentration. This finding had a cross-cutting aspect in the area of human performance associated with avoiding complacency. Specifically, individuals failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and individuals failed to implement appropriate error reduction tools. [H.12]
Inspection Report# : 2016002 *(pdf)*

**Significance:** G May 11, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Insufficient procedural direction contained within EOP E-2, Faulted Steam Generator Isolation**
The examiners identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Specifically, Procedure EOP E-2, "Faulted Steam Generator Isolation," does not contain sufficient procedural direction for isolating auxiliary feedwater flow to a faulted steam generator in the event that auxiliary feedwater control valves cannot be closed from the control room. Procedure EOP E-2, Appendix HH, "Isolated Faulted Steam Generator," Step 1.d, and its associated column, Response Not Obtained, does not ensure that a faulted steam generator would remain isolated under all conditions. The Response Not Obtained column permits operators to either locally close auxiliary feedwater control valves OR secure the auxiliary feedwater pump feeding the faulted steam generator. However, due to the absence of pull-to-lock or hard stop switches for the auxiliary feedwater pumps, the possibility exists for an automatic restart of an auxiliary feedwater pump and a re-initiation of feedwater to a faulted steam generator.

The failure to ensure that Procedure EOP E-2 contained sufficient direction to isolate a faulted steam generator when auxiliary feedwater flow control valves cannot be closed from the control room was a performance deficiency. This performance deficiency was of more than minor safety significance because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone (reactor coolant system and containment) and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the re-initiation of feedwater to an isolated, faulted steam generator has the potential to adversely affect the reactor coolant system barrier by causing an additional unintended cooldown of the reactor coolant system, increased potential for pressurized thermal shock, and thermal stress to the steam generator u-tubes. Additionally, the containment barrier would be affected by the re-initiation of feedwater to a steam line break within containment. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated June 19, 2012, the team determined that the finding required a detailed risk evaluation due to the potential to affect the reactor coolant system boundary. A senior reactor analyst performed a bounding detailed risk evaluation and estimated the maximum increase in core damage frequency to be 5.9E-8/year, and therefore the finding was determined to be of very low safety significance (Green). This increase in core damage frequency was mitigated by the low probability of multiple equipment failures in the auxiliary feedwater system when combined with the low initiating event frequency of a faulted steam generator. Because the violation was of very low safety significance (Green) and the issue was entered into the licensee's corrective action program as Notification 50847218, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000275/2016301; 05000323/2016301-01, "Insufficient Procedural Direction Contained Within E-2, Faulted Steam Generator Isolation." This finding has a cross-cutting aspect in the area of human performance associated with resources because the organization did not ensure procedures are available and adequate to support nuclear safety.

Inspection Report# : 2016301 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Security

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## Miscellaneous

Last modified : January 04, 2017

# Diablo Canyon 2
# 4Q/2016 Plant Inspection Findings

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## Initiating Events

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## Mitigating Systems

**Significance:** W Sep 12, 2016
Identified By: NRC
Item Type: VIO Violation
**Failure to Establish Adequate Work Instructions for Installation of Namco™ Snap Lock Limit Switches**
The inspectors identified a preliminary White finding associated with an apparent violation of Technical Specification 5.4.1.a, “Procedures,” for the licensee’s failure to develop adequate instructions for the installation, adjustment, and testing of Namco™ Model EA170 snap lock limit switches. Specifically, the licensee failed to provide site-specific instructions for limiting the travel of these external limit switches when installed in safety-related motor operated valves. Consequently, the lever switch actuator for valve RHR-2-8700B, residual heat removal pump 2-2 suction from the refueling water storage tank, was installed such that the limit switch was operated repeatedly in an over-travel condition resulting in a sheared internal roll pin that ultimately caused the limit switch to fail. Following identification of this issue, the licensee replaced the limit switch for valve RHR-2-8700B and implemented actions to modify maintenance procedures for installing, calibrating, and testing motor-operated valve external limit switches. The licensee entered this issue into their corrective action program as Notification 50852345.

The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, maintenance procedure MP E-53.10R, “Augmented Stem Lubrication for Limitorque Operated Valves,” used to perform limit switch adjustments on the Unit 2 valve RHR-2-8700B, did not provide adequate acceptance criteria to prevent overtravel of the limit switch actuating lever. This resulted in a subsequent failure of the limit switch, preventing the open permissive signal for valve SI-2-8982B, residual heat removal pump 2-2 suction from the containment recirculation sump, used during the emergency core cooling system (ECCS) recirculation mode. The inspectors evaluated the finding using the Attachment 0609.04, "Initial Characterization of Findings," worksheet to Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” issued June 19, 2012. The attachment instructs the inspectors to utilize IMC 0609, Appendix A, “Significance Determination Process (SDP) for Findings At-Power,” issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined that the finding required a detailed risk evaluation because it represented an actual loss of function of the train B ECCS for greater than its technical specification allowed outage time. A senior reactor analyst performed a detailed risk evaluation in accordance with IMC 0609, Appendix A, Section 6.0, “Detailed Risk Evaluation.” The calculated increase in core damage frequency was dominated by small and medium loss of coolant accident initiators with failures of the opposite train of ECCS or related support systems. The analyst did not evaluate the large early release frequency because this performance deficiency would not have challenged the containment. The NRC preliminarily determined that the increase in core damage frequency for internal and external initiators was 7.6E-06/year, a finding of low to moderate risk significance (White). The inspector did not identify a cross-cutting aspect with this finding because it was not reflective of current performance. The inadequate procedure was developed in 2011 and did not reflect the licensee’s

current performance related to procedure development.
(IR 05000275; 05000323/2016010, dated October 3, 2016, ML16277A340)

(FIRST UPDATE)
The finding was determined to be of low-to-moderate safety significance (White), because the NRC's calculated lower and upper estimations of the increase in core damage frequency of the performance deficiency were both greater than 1.0E-6 per year but less than 1.0E-5 per year. The NRC concluded that the preliminary significance determination change in core damage frequency result of 7.6E-6 per year represents the upper range of the increase in core damage frequency associated with the performance deficiency. Based on the information provided by the licensee at the November 15, 2016 regulatory conference, the NRC adjusted a number of assumptions used in the preliminary significance determination. Specifically, the NRC lowered the common cause alpha factors and adjusted several assumptions related to medium break loss-of-coolant accidents. The NRC also performed a variety of human error probability calculations to determine the likelihood of recovering the functionality of valve SI-2-8982B. The results of these calculations, which removed much of the conservativism from the assumptions used in the preliminary risk assessment, predicted a high likelihood of success (96.4 percent success) for recovering valve SI-2-8982B. Using these assumptions, the NRC concluded the lower range of increase in core damage frequency associated with the performance deficiency to be 1.3E-6 per year.
(Letter to E. Halpin from K. Kennedy, dated December 28, 2016, ML16363A429)
Inspection Report# : 2016010 *(pdf)*

**Significance:** G Jul 14, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Inadequate Maintenance Procedure affected the Performance of Safety-Related Emergency Diesel Generator**
The inspectors assessed a self-revealed, non-cited violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to implement properly preplanned maintenance procedures that affected the performance of safety-related equipment. Specifically, two maintenance procedures associated with the emergency diesel generators' fuel injectors lacked adequate details on specific key mechanical parameters (capscrew bolt torque setup and fuel injection pump alignment) to ensure that maintenance activities were performed in a manner adequate to the circumstances. In both examples, the licensee entered the issues into the corrective action program and corrected the condition to restore the emergency diesel generators to an operable status.

This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At Power," issued June 19, 2012, the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent the loss of a system or function, the loss of a train of a technical specification safety system for greater than its allowed outage time, or the loss of a non-technical specification high-safety-significant system for greater than 24 hours. This finding had a cross-cutting aspect in the area of human performance associated with work management – "organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority." Specifically, work on the emergency diesel generators fuel oil system components was not effectively planned and executed by incorporating conditions to ensure a successful outcome [H.5].

Inspection Report# : 2016009 *(pdf)*

**Significance:** G Mar 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation

**Failure to Verify Adequate Design Airflow for 480 volt AC Switchgear and 125 volt DC Inverter Rooms**
The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” for the failure to verify the design adequacy of the safety-related ventilation system for the 480-volt AC switchgear and 125-volt DC inverter rooms. Specifically, the licensee failed to verify sufficient ventilation system airflow to ensure the temperature in rooms housing safety-related electrical equipment remained below 104 degrees Fahrenheit. The licensee’s corrective actions were documented in Notification 50840266.

The failure to provide design control measures to verify the adequacy of the 480-volt AC switchgear and 125-volt DC inverter rooms ventilation system design was a performance deficiency. The performance deficiency was more than minor because it was associated with the design control attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the reduction in airflow to the rooms impacts the reliability of the safety-related equipment ventilation system to maintain the temperatures in these rooms below design limits for the duration of all accident scenarios. Using NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, “Mitigating Systems Screening Questions,” the inspectors determined the finding was of very low safety significance because (1) the finding was not a deficiency affecting the design or qualification of a mitigating system; (2) the finding did not represent a loss of system and/or function; (3) the finding did not represent an actual loss of function of a single train for greater than its technical specification allowed outage time; and (4) the finding does not represent an actual loss of function of one or more non-technical specification trains of equipment designated as high safety-significant in accordance with the licensee’s maintenance rule program for greater than 24 hours.

The inspectors determined that this finding did not have a cross-cutting aspect because the most significant contributor of this finding occurred more than three years ago, and is therefore, not representative of current licensee performance.
Inspection Report# : 2016001 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate 480 Vac Motor Starters with Circuit Breaker Trip Settings Higher than Manufacturers’ Specifications**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to September 10, 2013, the licensee failed to verify the design of 480 Vac combination motor starter instantaneous magnetic circuit breakers settings, by the use of alternate or simplified calculational methods, for those breakers whose settings are higher than their manufacturers' specifications, as documented in calculation 195B-DC, “MCCB Settings for 460VAC Class 1E Motors,” to provide the required level of protection and ensure that certain failures that could be caused by sustained fault currents below the circuit breaker trip setting would not occur. In response to this finding, the licensee conducted a preliminary evaluation of some of the affected equipment and concluded that sustained fault currents below the trip settings are unlikely. This finding was entered into the licensee's corrective action program as Notification 50838071.

The team determined the failure to evaluate 480 Vac combination motor starters with instantaneous magnetic circuit breaker trip current settings higher than their manufacturers' specifications was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, improper motor starter breaker trip settings could result in a fire in the motor control center cubicle, damage to motor starter components, spurious tripping of the entire motor control center, or lack of protection for downstream components during fault conditions.

In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding did not have a cross-cutting aspect because the most significant causal factor of the performance deficiency did not reflect current licensee performance.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Promptly Correct the Lack of Design Verification of 460 Vac Motors at Maximum Allowable Frequency**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” which states, in part, “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.” Specifically, prior to March 16, 2016, the licensee failed to assure that the lack of design verification of 460 Vac motors, which could be overloaded at the maximum allowable diesel generator frequency, was promptly corrected after having been identified in a 2013 apparent cause evaluation and again in a 2015 self-assessment as documented in Notifications 50572850 and 50826105, respectively. In response to this finding, the licensee performed a preliminary evaluation of the affected 460 Vac motors and concluded that operation at maximum emergency diesel generator frequency would not cause them to overheat or trip on overcurrent. This finding was entered into the licensee's corrective action program as Notifications 50835699 and 50838988.

The team determined the failure to correct the lack of design verification of 460 Vac motors at maximum allowable frequency when powered from the emergency diesel generators was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of 460 Vac motors above their rated or analyzed maximum allowable frequencies could result in motor overheating or a trip of the thermal overload relays. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of problem identification and resolution associated with evaluation because the licensee failed to ensure that the organization thoroughly evaluated issues to ensure that resolutions address causes and extent of conditions.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Ensure Safety-Related Alternating Current and Direct Current Equipment Functionality at Maximum Allowable Voltages**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,”

which states, in part, “The design control measures shall provide for verifying or checking the adequacy of design, such as by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.” Specifically, prior to February 10, 2016, the licensee failed to verify the design of (1) equipment on the nominally 125 Vdc system at the maximum voltage specified in Procedure OP J-9:IV, “Performing a Battery Equalizing Charge,” and (2) equipment on 480 Vac and 120 Vac vital buses at maximum voltages specified in Procedure OP J-2:VIII, “Guidelines for Reliable Transmission Service for DCPP,” by the use of alternate or simplified calculational methods, to ensure equipment functionality. In response to this finding, the licensee conducted a preliminary evaluation of the affected equipment and concluded that any past exposure to voltages above their maximum rating would not have caused a loss of functionality. This finding was entered into the licensee's corrective action program as Notifications 50834558, 50835906, 50835394, 50835945, 50835949, 50836376, 50836439, 50836638, 50836872, and 50836995.

The team determined the failure to evaluate operation of 125 Vdc and 480 and 120 Vac equipment at maximum allowable voltages was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, operation of equipment outside of its rated or analyzed maximum allowable voltages adversely affects the reliability and capability of that equipment required to perform safety-related functions. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Extent of Condition for a Degraded Condition on a Nonsafety-Related 4160 Vac Breaker**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.” Specifically, in October of 2015, the licensee failed to evaluate the extent of condition of a cracked holding pawl on a nonsafety-related 4160 Vac SF6 breaker, which was procured as safety-related, in accordance with Procedure OM7.ID1, “Problem Identification and Resolution,” when the failure of the component could adversely impact safety-related breakers of the same make and model. In response to this finding, the licensee is performing a procedure review to include steps to perform an extent of condition analysis for unplanned nonsafety-related equipment issues that may also affect similar safety-related equipment. This finding was entered into the licensee's corrective action program as Notifications 50836859 and 50836689.

The team determined the failure to evaluate the impact of a cracked holding pawl identified on a nonsafety-related 4160 Vac SF6 breaker on additional safety-related 4160 Vac SF6 breakers was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the equipment performance attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the 4160 Vac breaker with the cracked holding pawl was procured as safety-related; therefore, the condition extends to safety-related 4160 Vac breakers of the same make and model and potentially adversely affects the ability to perform their safety function. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with conservative bias because the licensee failed to ensure that individuals used decision-making practices that emphasized prudent choices.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Evaluate the Voltage Effects of Limiting Design Basis Events on the 230 kV Offsite Power Circuit**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," which states, in part, "The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Specifically, prior to January 30, 2014, the licensee failed to verify the design of the 230 kV preferred offsite power source, such as by the performance of design reviews or use of alternate or simplified calculational methods, by assuming in calculation 359-DC, "Determination of 230 kV Grid Capability Limits as DCPP Offsite Power Source," that the reactor trip and engineered safety features actuation system signals are coincident in time for all postulated design basis events. However, the plant is designed such that, during some events, the signals are separate in time and would result in a greater vital bus voltage depression than analyzed. In response to this finding, the licensee conducted a preliminary evaluation and concluded that the current transmission grid conditions were such that the calculation criteria would be met in the event of a design basis event involving non-coincident reactor trip and engineered safety features actuation system signals. This finding was entered into the licensee's corrective action program as Notification 50839137.

The team determined the failure to evaluate the voltage effects of a limiting design basis event with non-coincident reactor trip and engineered safety features actuation system signals on the 230 kV offsite power circuit was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure adequate bus voltages as a result of a design basis event with non-coincident reactor trip and engineered safety features actuation system signals would result in a trip of the undervoltage relays and the loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

**Significance:** G Mar 10, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Translate Appropriate Load Tap Changer Timing Acceptance Criteria into Periodic Tests**
The team identified a Green, non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” which states, in part, “Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.” Specifically, prior to November 25, 2015, the licensee failed to include appropriate quantitative acceptance criteria in Procedure MP E-62.3, “Tap Changer Functional Test for Standby-Startup Transformer 11,” to ensure that the load tap changer speed for standby-startup transformer 11 was adequate to restore vital bus voltages to the required level during design basis events. In response to this finding, the licensee performed a preliminary evaluation of the condition and concluded that the most recently measured speed of the load tap changer was adequate to ensure that it would restore vital bus voltage within the required time. This finding was entered into the licensee's corrective action program as Notification 50839333.

The team determined the failure to translate appropriate load tap changer timing acceptance criteria into functional tests to ensure that design assumptions were being maintained was a performance deficiency. The performance deficiency was more-than-minor, and therefore a finding, because it related to the design control attribute of the Mitigating Systems cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the load tap changer could meet its functional test acceptance criterion, but not operate fast enough to restore vital bus voltages within the required time during design basis events, which would result in an undervoltage trip and loss of the preferred offsite power circuit. In accordance with Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated July 19, 2012, the finding screened as having very low safety significance (Green) because it was a design or qualification deficiency that did not represent a loss of operability or functionality; did not represent an actual loss of safety function of the system or train; did not result in the loss of one or more trains of non-technical specification equipment; and did not screen as potentially risk-significant due to seismic, flooding, or severe weather. This finding had a cross-cutting aspect in the area of human performance associated with design margins because the licensee failed to ensure that the organization operated and maintained equipment within design margins and that margins were carefully guarded and changed only through a systematic and rigorous process.

Inspection Report# : 2016007 *(pdf)*

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## Barrier Integrity

**Significance:** G Jun 30, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Misplaced Spent Fuel Assembly in the Spent Fuel Pool**
The inspectors reviewed a self-revealed, non-cited violation of Technical Specification (TS) 5.4.1.a, "Procedures," for the licensee’s failure to place a spent fuel assembly in its correct location in the spent fuel pool (SFP) in accordance with Procedure OP B-8H, "Spent Fuel Pool Work Instructions." Specifically, the fuel handling crew moved spent fuel assembly TT69 to location E-37 rather than its intended location E-27. In response to this error, reactor engineering performed a technical specification verification in order to ensure that fuel assembly TT69 could remain in Cell E-37. The licensee suspended further fuel movements pending corrective action and remediation of the operators. The licensee entered this into the corrective action program as Notifications 50846834 and 50847067.

The licensee’s failure to place a spent fuel assembly in its correct location in the SFP was a performance deficiency. The performance deficiency is more than minor, and therefore a finding, because it is associated with the configuration control attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Using Inspection Manual Chapter 0609, Attachment 04, "Initial Characterization of Findings," and Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," the inspectors determined that the finding was of very low safety significance (Green) because: (1) the finding did not adversely affect decay heat removal capabilities from the spent fuel pool causing the pool temperature to exceed the maximum analyzed temperature limit specified in the site-specific licensing basis, (2) the finding did not result from fuel handling errors, dropped fuel assembly, dropped storage cask, or crane operations over the SFP that caused mechanical damage to fuel clad and a detectible release of radionuclides, (3) the finding did not result in a loss of spent fuel pool water inventory decreasing below the minimum analyzed level limit specified in the site-specific licensing basis, and (4) the finding did not affect the SFP neutron absorber, fuel bundle misplacement (i.e., fuel loading pattern error) or soluble Boron concentration. This finding had a cross-cutting aspect in the area of human performance associated with avoiding complacency. Specifically, individuals failed to recognize and plan for the possibility of mistakes, latent issues, and inherent risk, even while expecting successful outcomes and individuals failed to implement appropriate error reduction tools. [H.12]
Inspection Report# : 2016002 *(pdf)*

**Significance:** G May 11, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Insufficient procedural direction contained within EOP E-2, Faulted Steam Generator Isolation**
The examiners identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” Specifically, Procedure EOP E-2, “Faulted Steam Generator Isolation,” does not contain sufficient procedural direction for isolating auxiliary feedwater flow to a faulted steam generator in the event that auxiliary feedwater control valves cannot be closed from the control room. Procedure EOP E-2, Appendix HH, “Isolated Faulted Steam Generator,” Step 1.d, and its associated column, Response Not Obtained, does not ensure that a faulted steam generator would remain isolated under all conditions. The Response Not Obtained column permits operators to either locally close auxiliary feedwater control valves OR secure the auxiliary feedwater pump feeding the faulted steam generator. However, due to the absence of pull-to-lock or hard stop switches for the auxiliary feedwater pumps, the possibility exists for an automatic restart of an auxiliary feedwater pump and a re-initiation of feedwater to a faulted steam generator.

The failure to ensure that Procedure EOP E-2 contained sufficient direction to isolate a faulted steam generator when auxiliary feedwater flow control valves cannot be closed from the control room was a performance deficiency. This performance deficiency was of more than minor safety significance because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone (reactor coolant system and containment) and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the re-initiation of feedwater to an isolated, faulted steam generator has the potential to adversely affect the reactor coolant system barrier by causing an additional unintended cooldown of the reactor coolant system, increased potential for pressurized thermal shock, and thermal stress to the steam generator u-tubes. Additionally, the containment barrier would be affected by the re-initiation of feedwater to a steam line break within containment. Using Inspection Manual Chapter 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” dated June 19, 2012, the team determined that the finding required a detailed risk evaluation due to the potential to affect the reactor coolant system boundary. A senior reactor analyst performed a bounding detailed risk evaluation and estimated the maximum increase in core damage frequency to be 5.9E-8/year, and therefore the finding was determined to be of very low safety significance (Green). This increase in core damage frequency was mitigated by the low probability of multiple equipment failures in the auxiliary feedwater system when combined with the low initiating event frequency of a faulted steam generator.

Because the violation was of very low safety significance (Green) and the issue was entered into the licensee's corrective action program as Notification 50847218, this violation is being treated as a non-cited violation consistent with Section 2.3.2.a of the Enforcement Policy: NCV 05000275/2016301; 05000323/2016301-01, "Insufficient Procedural Direction Contained Within E-2, Faulted Steam Generator Isolation." This finding has a cross-cutting aspect in the area of human performance associated with resources because the organization did not ensure procedures are available and adequate to support nuclear safety.
Inspection Report# : 2016301 *(pdf)*

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## Emergency Preparedness

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## Occupational Radiation Safety

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## Public Radiation Safety

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## Security

Although the Security Cornerstone is included in the Reactor Oversight Process assessment program, the Commission has decided that specific information related to findings and performance indicators pertaining to the Security Cornerstone will not be publicly available to ensure that security information is not provided to a possible adversary. Other than the fact that a finding or performance indicator is Green or Greater-Than-Green, security related information will not be displayed on the public web page. Therefore, the cover letters to security inspection reports may be viewed.

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## Security

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## Miscellaneous

Last modified : February 01, 2017

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# Diablo Canyon 2 – Quarterly Plant Inspection Findings

## 2Q/2017 – Plant Inspection Findings

On this page:



## Initiating Events

## Mitigating Systems

**Significance:** G Dec 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Follow Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors identified a non-cited violation of Technical Specification 5.4.1.a,"Procedures," for the failure to follow Procedure AD7.ID16, "Tool Pouch and MinorMaintenance Program," Revision 2. Specifically, the licensee failed to screen work on the safety-related rupture restraint as acceptable to be worked as tool pouch work or minor maintenance. As a result, a safety-related main steam line rupture restraint (MS-41RR) was not properly returned to service and left in an inoperable condition following maintenance. As corrective actions, the licensee returned MS-41RR to an operable condition and initiated a review of the maintenance database to ensure that work performed on main steam line rupture restraints is completed in accordance with appropriate written inspections. The licensee entered the issue into their corrective action program as Notifications 50872133, 50872056, and 50872789.

The failure to properly preplan and perform maintenance affecting the performance of safety-related equipment was a performance deficiency. The inspectors determined that the finding was more than minor because it was associated with the configuration control attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, because of not following maintenance procedures, a safety-related main steam rupture restraint was left in a disengaged or inactive configuration such that following a postulated line break, the main steam line would be unrestrained. This resulted in a potential of high-energy pipe impacting structures and components designed to be protected from high-energy pipe whip. Using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent an actual loss of function of a mitigating system. Specifically, the single restraint condition would only affect a very limited range of breaks and no

risk significant systems would be adversely impacted. The inspectors concluded that this finding affected the cross cutting area of human performance, documentation, because the licensee did not maintain up to date documentation to ensure work planning on safety related equipment are complete, thorough, accurate, and current such that main steam pipe restraints are maintained within design requirements [H.7].
Inspection Report# : 2016004 *(pdf)*

**Significance:** W Sep 12, 2016
Identified By: NRC
Item Type: VIO Violation
**Failure to Establish Adequate Work Instructions for Installation of Namco□ Snap Lock Limit Switches**
The inspectors identified a preliminary White finding associated with an apparent violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to develop adequate instructions for the installation, adjustment, and testing of Namco□ Model EA170 snap lock limit switches. Specifically, the licensee failed to provide site-specific instructions for limiting the travel of these external limit switches when installed in safety-related motor operated valves. Consequently, the lever switch actuator for valve RHR-2-8700B, residual heat removal pump 2-2 suction from the refueling water storage tank, was installed such that the limit switch was operated repeatedly in an over-travel condition resulting in a sheared internal roll pin that ultimately caused the limit switch to fail. Following identification of this issue, the licensee replaced the limit switch for valve RHR-2-8700B and implemented actions to modify maintenance procedures for installing, calibrating, and testing motor-operated valve external limit switches. The licensee entered this issue into their corrective action program as Notification 50852345.

The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, maintenance procedure MP E-53.10R, "Augmented Stem Lubrication for Limitorque Operated Valves," used to perform limit switch adjustments on the Unit 2 valve RHR-2-8700B, did not provide adequate acceptance criteria to prevent overtravel of the limit switch actuating lever. This resulted in a subsequent failure of the limit switch, preventing the open permissive signal for valve SI-2-8982B, residual heat removal pump 2-2 suction from the containment recirculation sump, used during the emergency core cooling system (ECCS) recirculation mode. The inspectors evaluated the finding using the Attachment 0609.04, "Initial Characterization of Findings," worksheet to Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," issued June 19, 2012. The attachment instructs the inspectors to utilize IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined that the finding required a detailed risk evaluation because it represented an actual loss of function of the train B ECCS for greater than its technical specification allowed outage time. A senior reactor analyst performed a detailed risk evaluation in accordance with IMC 0609, Appendix A, Section 6.0, "Detailed Risk Evaluation." The calculated increase in core damage frequency was dominated by small and medium loss of coolant accident initiators with failures of the opposite train of ECCS or related support systems. The analyst did not evaluate the large early release frequency because this performance deficiency would not have challenged the containment. The NRC preliminarily determined that the increase in core damage frequency for internal and external initiators was 7.6E-06/year, a finding of low to moderate risk significance (White). The inspector did not identify a cross-cutting aspect with this finding because it was not reflective of current performance. The inadequate procedure was developed in 2011 and did not reflect the licensee's current performance related to procedure development.
(IR 05000275; 05000323/2016010, dated October 3, 2016, ML16277A340)

(FIRST UPDATE)
The finding was determined to be of low-to-moderate safety significance (White), because the NRC's calculated lower

and upper estimations of the increase in core damage frequency of the performance deficiency were both greater than 1.0E-6 per year but less than 1.0E-5 per year. The NRC concluded that the preliminary significance determination change in core damage frequency result of 7.6E-6 per year represents the upper range of the increase in core damage frequency associated with the performance deficiency. Based on the information provided by the licensee at the November 15, 2016 regulatory conference, the NRC adjusted a number of assumptions used in the preliminary significance determination. Specifically, the NRC lowered the common cause alpha factors and adjusted several assumptions related to medium break loss-of-coolant accidents. The NRC also performed a variety of human error probability calculations to determine the likelihood of recovering the functionality of valve SI-2-8982B. The results of these calculations, which removed much of the conservativism from the assumptions used in the preliminary risk assessment, predicted a high likelihood of success (96.4 percent success) for recovering valve SI-2-8982B. Using these assumptions, the NRC concluded the lower range of increase in core damage frequency associated with the performance deficiency to be 1.3E-6 per year.
(Letter to E. Halpin from K. Kennedy, dated December 28, 2016, ML16363A429)
Inspection Report# : 2016010 *(pdf)*

**Significance:** G Jul 14, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Inadequate Maintenance Procedure affected the Performance of Safety-Related Emergency Diesel Generator**
The inspectors assessed a self-revealed, non-cited violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to implement properly preplanned maintenance procedures that affected the performance of safety-related equipment. Specifically, two maintenance procedures associated with the emergency diesel generators' fuel injectors lacked adequate details on specific key mechanical parameters (capscrew bolt torque setup and fuel injection pump alignment) to ensure that maintenance activities were performed in a manner adequate to the circumstances. In both examples, the licensee entered the issues into the corrective action program and corrected the condition to restore the emergency diesel generators to an operable status.

This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At Power," issued June 19, 2012, the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent the loss of a system or function, the loss of a train of a technical specification safety system for greater than its allowed outage time, or the loss of a non-technical specification high-safety-significant system for greater than 24 hours. This finding had a cross-cutting aspect in the area of human performance associated with work management - "organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority." Specifically, work on the emergency diesel generators fuel oil system components was not effectively planned and executed by incorporating conditions to ensure a successful outcome [H.5].

Inspection Report# : 2016009 *(pdf)*

## Barrier Integrity
## Emergency Preparedness
## Occupational Radiation Safety
## Public Radiation Safety
## Security

The security cornerstone is an important component of the ROP, which includes various security inspection activities the NRC uses to verify licensee compliance with Commission regulations and thus ensure public health and safety. The

Commission determined in the staff requirements memorandum (SRM) for SECY-04-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure," dated November 9, 2004, that specific information related to findings and performance indicators associated with the security cornerstone will not be publicly available to ensure that security-related information is not provided to a possible adversary. Security inspection report cover letters will be available on the NRC Web site; however, security-related information on the details of inspection finding(s) will not be displayed.

## Miscellaneous

Current data as of : August 03, 2017

*Page Last Reviewed/Updated Wednesday, August 10, 2016*

U.S.NRC
United States Nuclear Regulatory Commission
*Protecting People and the Environment*

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# Diablo Canyon 2 – Quarterly Plant Inspection Findings

## 2Q/2017 – Plant Inspection Findings

On this page:



## Initiating Events

## Mitigating Systems

**Significance: N/A** Jun 30, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Conduct Required Biennial Medical Examinations within Two Years**
The inspectors identified a Severity Level IV, non-cited violation of 10 CFR 55.21, "Medical Examination," for the licensee's failure to ensure that a medical examination by a physician to determine satisfaction of 10 CFR 55.33(a)(1) requirements was conducted every 2 years for two licensed senior operators. Specifically, one licensed senior operator exceeded the two-year medical examination requirement by approximately 16 months between November 27, 2015, and April 6, 2017. A second licensed senior operator exceeded the 2-year medical examination requirement by 4 months between November 19, 2016, and April 6, 2017. As a corrective action, the licensee has conducted the required medical examination for one senior operator and initiated a license termination request for the other senior operator. This issue was entered into the licensee's corrective action program as Notification 50912407.

The failure of the facility licensee to conduct required biennial medical examinations for two licensed senior operators was a performance deficiency. This issue was evaluated using the traditional enforcement process because it negatively impacted the NRC's ability to perform its regulatory oversight function. Specifically, the failure to comply with medical testing requirements for two operators compromised the facility licensee's ability to assure conformance to medical standards, detect non-conforming medical conditions, and report non-conformances to the NRC. This performance deficiency was determined to be Severity Level IV because it fits the Severity Level IV example of Enforcement Policy Section 6.4.d.1, "Violation Examples: Licensed Reactor Operators." This section states, "Severity Level IV violations involve, for example □ (b) an individual operator who did not meet the American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.4," "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," Section 5, "Health Requirements and Disqualifying Conditions," as certified on NRC Form 396, "Certification of Medical Examination by Facility Licensee," required by 10 CFR 55.23, Certification, but who did not perform the functions of a licensed operator or senior operator while having a

disqualifying medical condition." No cross-cutting aspect was assigned because the violation was processed using traditional enforcement.
Inspection Report# : 2017002 *(pdf)*

**Significance: N/A** Jun 30, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Report a Permanent Medical Condition within 30 Days**
The inspectors identified a Severity Level IV, non-cited violation of 10 CFR 55.25, "Incapacitation Because of Disability or Illness," for the licensee's failure to notify the NRC within 30 days of a change to one licensed senior operator's medical condition. Specifically, the licensed senior operator developed a permanent medical condition which caused him to permanently leave the site on December 1, 2014, and transition into a long-term disability program on April 23, 2015. The licensee did not notify the NRC of this change in medical condition. As a corrective action, the licensee initiated a license termination request for the affected operator, effective April 6, 2017. This issue was entered into the licensee's corrective action program as Notification 50912407.

The failure of the facility licensee to notify the NRC within 30 days of a change in a licensed senior operator's medical condition was a performance deficiency. This issue was evaluated using the traditional enforcement process because it negatively impacted the NRC's ability to perform its regulatory oversight function. Specifically, the failure to report changes in a licensed senior operator's medical condition prevented the NRC from taking action to issue either a license amendment or termination, as appropriate. This performance deficiency was determined to be Severity Level IV because it fits the Severity Level IV example of Enforcement Policy Section 6.4.d.1, "Violation Examples: Licensed Reactor Operators." This section states, "Severity Level IV violations involve, for example □ (b) an individual operator who did not meet the American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.4," "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," Section 5, "Health Requirements and Disqualifying Conditions," as certified on NRC Form 396, "Certification of Medical Examination by Facility Licensee," required by 10 CFR 55.23, Certification, but who did not perform the functions of a licensed operator or senior operator while having a disqualifying medical condition." No cross-cutting aspect was assigned because the violation was processed using traditional enforcement.
Inspection Report# : 2017002 *(pdf)*

**Significance:** G Dec 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Follow Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors identified a non-cited violation of Technical Specification 5.4.1.a,"Procedures," for the failure to follow Procedure AD7.ID16, "Tool Pouch and MinorMaintenance Program," Revision 2. Specifically, the licensee failed to screen work on the safety-related rupture restraint as acceptable to be worked as tool pouch work or minor maintenance. As a result, a safety-related main steam line rupture restraint (MS-41RR) was not properly returned to service and left in an inoperable condition following maintenance. As corrective actions, the licensee returned MS-41RR to an operable condition and initiated a review of the maintenance database to ensure that work performed on main steam line rupture restraints is completed in accordance with appropriate written inspections. The licensee entered the issue into their corrective action program as Notifications 50872133, 50872056, and 50872789.

The failure to properly preplan and perform maintenance affecting the performance of safety-related equipment was a performance deficiency. The inspectors determined that the finding was more than minor because it was associated with the configuration control attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, because of not following maintenance procedures, a safety-related main steam

rupture restraint was left in a disengaged or inactive configuration such that following a postulated line break, the main steam line would be unrestrained. This resulted in a potential of high-energy pipe impacting structures and components designed to be protected from high-energy pipe whip. Using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent an actual loss of function of a mitigating system. Specifically, the single restraint condition would only affect a very limited range of breaks and no risk significant systems would be adversely impacted. The inspectors concluded that this finding affected the cross cutting area of human performance, documentation, because the licensee did not maintain up to date documentation to ensure work planning on safety related equipment are complete, thorough, accurate, and current such that main steam pipe restraints are maintained within design requirements [H.7].
Inspection Report# : 2016004 *(pdf)*

**Significance:** W Sep 12, 2016
Identified By: NRC
Item Type: VIO Violation
**Failure to Establish Adequate Work Instructions for Installation of Namco□ Snap Lock Limit Switches**
The inspectors identified a preliminary White finding associated with an apparent violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to develop adequate instructions for the installation, adjustment, and testing of Namco□ Model EA170 snap lock limit switches. Specifically, the licensee failed to provide site-specific instructions for limiting the travel of these external limit switches when installed in safety-related motor operated valves. Consequently, the lever switch actuator for valve RHR-2-8700B, residual heat removal pump 2-2 suction from the refueling water storage tank, was installed such that the limit switch was operated repeatedly in an over-travel condition resulting in a sheared internal roll pin that ultimately caused the limit switch to fail. Following identification of this issue, the licensee replaced the limit switch for valve RHR-2-8700B and implemented actions to modify maintenance procedures for installing, calibrating, and testing motor-operated valve external limit switches. The licensee entered this issue into their corrective action program as Notification 50852345.

The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, maintenance procedure MP E-53.10R, "Augmented Stem Lubrication for Limitorque Operated Valves," used to perform limit switch adjustments on the Unit 2 valve RHR-2-8700B, did not provide adequate acceptance criteria to prevent overtravel of the limit switch actuating lever. This resulted in a subsequent failure of the limit switch, preventing the open permissive signal for valve SI-2-8982B, residual heat removal pump 2-2 suction from the containment recirculation sump, used during the emergency core cooling system (ECCS) recirculation mode. The inspectors evaluated the finding using the Attachment 0609.04, "Initial Characterization of Findings," worksheet to Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," issued June 19, 2012. The attachment instructs the inspectors to utilize IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined that the finding required a detailed risk evaluation because it represented an actual loss of function of the train B ECCS for greater than its technical specification allowed outage time. A senior reactor analyst performed a detailed risk evaluation in accordance with IMC 0609, Appendix A, Section 6.0, "Detailed Risk Evaluation." The calculated increase in core damage frequency was dominated by small and medium loss of coolant accident initiators with failures of the opposite train of ECCS or related support systems. The analyst did not evaluate the large early release frequency because this performance deficiency would not have challenged the containment. The NRC preliminarily determined that the increase in core damage frequency for internal and external initiators was 7.6E-06/year, a finding of low to moderate risk significance (White). The inspector did not identify a cross-cutting aspect with this finding because it was not reflective of current

performance. The inadequate procedure was developed in 2011 and did not reflect the licensee's current performance related to procedure development.
(IR 05000275; 05000323/2016010, dated October 3, 2016, ML16277A340)

(FIRST UPDATE)
The finding was determined to be of low-to-moderate safety significance (White), because the NRC's calculated lower and upper estimations of the increase in core damage frequency of the performance deficiency were both greater than 1.0E-6 per year but less than 1.0E-5 per year. The NRC concluded that the preliminary significance determination change in core damage frequency result of 7.6E-6 per year represents the upper range of the increase in core damage frequency associated with the performance deficiency. Based on the information provided by the licensee at the November 15, 2016 regulatory conference, the NRC adjusted a number of assumptions used in the preliminary significance determination. Specifically, the NRC lowered the common cause alpha factors and adjusted several assumptions related to medium break loss-of-coolant accidents. The NRC also performed a variety of human error probability calculations to determine the likelihood of recovering the functionality of valve SI-2-8982B. The results of these calculations, which removed much of the conservativism from the assumptions used in the preliminary risk assessment, predicted a high likelihood of success (96.4 percent success) for recovering valve SI-2-8982B. Using these assumptions, the NRC concluded the lower range of increase in core damage frequency associated with the performance deficiency to be 1.3E-6 per year.
(Letter to E. Halpin from K. Kennedy, dated December 28, 2016, ML16363A429)
Inspection Report# : 2016010 *(pdf)*

**Significance:** G Jul 14, 2016
Identified By: Self-Revealing
Item Type: NCV Non-Cited Violation
**Inadequate Maintenance Procedure affected the Performance of Safety-Related Emergency Diesel Generator**
The inspectors assessed a self-revealed, non-cited violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to implement properly preplanned maintenance procedures that affected the performance of safety-related equipment. Specifically, two maintenance procedures associated with the emergency diesel generators' fuel injectors lacked adequate details on specific key mechanical parameters (capscrew bolt torque setup and fuel injection pump alignment) to ensure that maintenance activities were performed in a manner adequate to the circumstances. In both examples, the licensee entered the issues into the corrective action program and corrected the condition to restore the emergency diesel generators to an operable status.

This finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and affects the associated cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events. Using Inspection Manual Chapter 0609, Appendix A, "The Significance Determination Process for Findings At Power," issued June 19, 2012, the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent the loss of a system or function, the loss of a train of a technical specification safety system for greater than its allowed outage time, or the loss of a non-technical specification high-safety-significant system for greater than 24 hours. This finding had a cross-cutting aspect in the area of human performance associated with work management - "organization implements a process of planning, controlling, and executing work activities such that nuclear safety is the overriding priority." Specifically, work on the emergency diesel generators fuel oil system components was not effectively planned and executed by incorporating conditions to ensure a successful outcome [H.5].

Inspection Report# : 2016009 *(pdf)*

## Barrier Integrity

## Emergency Preparedness

## Occupational Radiation Safety

## Public Radiation Safety

## Security

The security cornerstone is an important component of the ROP, which includes various security inspection activities the NRC uses to verify licensee compliance with Commission regulations and thus ensure public health and safety. The Commission determined in the staff requirements memorandum (SRM) for SECY-04-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure," dated November 9, 2004, that specific information related to findings and performance indicators associated with the security cornerstone will not be publicly available to ensure that security-related information is not provided to a possible adversary. Security inspection report cover letters will be available on the NRC Web site; however, security-related information on the details of inspection finding(s) will not be displayed.

## Miscellaneous

Current data as of : September 05, 2017

*Page Last Reviewed/Updated Wednesday, June 07, 2017*

United States Nuclear Regulatory Commission
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# Diablo Canyon 2 – Quarterly Plant Inspection Findings

## 3Q/2017 – Plant Inspection Findings

On this page:



## Initiating Events

## Mitigating Systems

**Significance: N/A** Aug 10, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Conduct Required Biennial Medical Examinations within Two Years**

Inspection Report# : 2017002 *(pdf)*

**Significance: N/A** Aug 10, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Report a Permanent Medical Condition within 30 Days**

Inspection Report# : 2017002 *(pdf)*

**Significance:** G Dec 31, 2016
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Follow Maintenance Procedure Resulted in Improper Configuration of Safety Related Equipment**
The inspectors identified a non-cited violation of Technical Specification 5.4.1.a,"Procedures," for the failure to follow Procedure AD7.ID16, "Tool Pouch and MinorMaintenance Program," Revision 2. Specifically, the licensee failed to screen work on the safety-related rupture restraint as acceptable to be worked as tool pouch work or minor maintenance. As a result, a safety-related main steam line rupture restraint (MS-41RR) was not properly returned to service and left in an inoperable condition following maintenance. As corrective actions, the licensee returned MS-41RR to an operable condition and initiated a review of the maintenance database to ensure that work performed on

main steam line rupture restraints is completed in accordance with appropriate written inspections. The licensee entered the issue into their corrective action program as Notifications 50872133, 50872056, and 50872789.

The failure to properly preplan and perform maintenance affecting the performance of safety-related equipment was a performance deficiency. The inspectors determined that the finding was more than minor because it was associated with the configuration control attribute of the Mitigating System Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesired consequences. Specifically, because of not following maintenance procedures, a safety-related main steam rupture restraint was left in a disengaged or inactive configuration such that following a postulated line break, the main steam line would be unrestrained. This resulted in a potential of high-energy pipe impacting structures and components designed to be protected from high-energy pipe whip. Using IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined the finding was of very low safety significance (Green) because the finding did not represent an actual loss of function of a mitigating system. Specifically, the single restraint condition would only affect a very limited range of breaks and no risk significant systems would be adversely impacted. The inspectors concluded that this finding affected the cross cutting area of human performance, documentation, because the licensee did not maintain up to date documentation to ensure work planning on safety related equipment are complete, thorough, accurate, and current such that main steam pipe restraints are maintained within design requirements [H.7].
Inspection Report# : 2016004 *(pdf)*

**Significance:** W Sep 12, 2016
Identified By: NRC
Item Type: VIO Violation
**Failure to Establish Adequate Work Instructions for Installation of Namco□ Snap Lock Limit Switches**
The inspectors identified a preliminary White finding associated with an apparent violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to develop adequate instructions for the installation, adjustment, and testing of Namco□ Model EA170 snap lock limit switches. Specifically, the licensee failed to provide site-specific instructions for limiting the travel of these external limit switches when installed in safety-related motor operated valves. Consequently, the lever switch actuator for valve RHR-2-8700B, residual heat removal pump 2-2 suction from the refueling water storage tank, was installed such that the limit switch was operated repeatedly in an over-travel condition resulting in a sheared internal roll pin that ultimately caused the limit switch to fail. Following identification of this issue, the licensee replaced the limit switch for valve RHR-2-8700B and implemented actions to modify maintenance procedures for installing, calibrating, and testing motor-operated valve external limit switches. The licensee entered this issue into their corrective action program as Notification 50852345.

The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, maintenance procedure MP E-53.10R, "Augmented Stem Lubrication for Limitorque Operated Valves," used to perform limit switch adjustments on the Unit 2 valve RHR-2-8700B, did not provide adequate acceptance criteria to prevent overtravel of the limit switch actuating lever. This resulted in a subsequent failure of the limit switch, preventing the open permissive signal for valve SI-2-8982B, residual heat removal pump 2-2 suction from the containment recirculation sump, used during the emergency core cooling system (ECCS) recirculation mode. The inspectors evaluated the finding using the Attachment 0609.04, "Initial Characterization of Findings," worksheet to Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," issued June 19, 2012. The attachment instructs the inspectors to utilize IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined that the finding required a detailed risk

evaluation because it represented an actual loss of function of the train B ECCS for greater than its technical specification allowed outage time. A senior reactor analyst performed a detailed risk evaluation in accordance with IMC 0609, Appendix A, Section 6.0, "Detailed Risk Evaluation." The calculated increase in core damage frequency was dominated by small and medium loss of coolant accident initiators with failures of the opposite train of ECCS or related support systems. The analyst did not evaluate the large early release frequency because this performance deficiency would not have challenged the containment. The NRC preliminarily determined that the increase in core damage frequency for internal and external initiators was 7.6E-06/year, a finding of low to moderate risk significance (White). The inspector did not identify a cross-cutting aspect with this finding because it was not reflective of current performance. The inadequate procedure was developed in 2011 and did not reflect the licensee's current performance related to procedure development.
(IR 05000275; 05000323/2016010, dated October 3, 2016, ML16277A340)

(FIRST UPDATE)
The finding was determined to be of low-to-moderate safety significance (White), because the NRC's calculated lower and upper estimations of the increase in core damage frequency of the performance deficiency were both greater than 1.0E-6 per year but less than 1.0E-5 per year. The NRC concluded that the preliminary significance determination change in core damage frequency result of 7.6E-6 per year represents the upper range of the increase in core damage frequency associated with the performance deficiency. Based on the information provided by the licensee at the November 15, 2016 regulatory conference, the NRC adjusted a number of assumptions used in the preliminary significance determination. Specifically, the NRC lowered the common cause alpha factors and adjusted several assumptions related to medium break loss-of-coolant accidents. The NRC also performed a variety of human error probability calculations to determine the likelihood of recovering the functionality of valve SI-2-8982B. The results of these calculations, which removed much of the conservativism from the assumptions used in the preliminary risk assessment, predicted a high likelihood of success (96.4 percent success) for recovering valve SI-2-8982B. Using these assumptions, the NRC concluded the lower range of increase in core damage frequency associated with the performance deficiency to be 1.3E-6 per year.
(Letter to E. Halpin from K. Kennedy, dated December 28, 2016, ML16363A429)
Inspection Report# : 2016010 *(pdf)*

## Barrier Integrity
## Emergency Preparedness
## Occupational Radiation Safety
## Public Radiation Safety
## Security
The security cornerstone is an important component of the ROP, which includes various security inspection activities the NRC uses to verify licensee compliance with Commission regulations and thus ensure public health and safety. The Commission determined in the staff requirements memorandum (SRM) for SECY-04-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure," dated November 9, 2004, that specific information related to findings and performance indicators associated with the security cornerstone will not be publicly available to ensure that security-related information is not provided to a possible adversary. Security inspection report cover letters will be available on the NRC Web site; however, security-related information on the details of inspection finding(s) will not be displayed.

## Miscellaneous
Current data as of : November 29, 2017

*Page Last Reviewed/Updated Monday, November 06, 2017*

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# Diablo Canyon 2 – Quarterly Plant Inspection Findings

## 4Q/2017 – Plant Inspection Findings

On this page:



### Initiating Events

### Mitigating Systems

**Significance: N/A** Aug 10, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Conduct Required Biennial Medical Examinations within Two Years**

Inspection Report# : 2017002 *(pdf)*

**Significance: N/A** Aug 10, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Report a Permanent Medical Condition within 30 Days**

Inspection Report# : 2017002 *(pdf)*

**Significance: NOPD** Aug 10, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Report a Permanent Medical Condition Within 30 Days**

The inspectors identified a Severity Level IV, non-cited violation of 10 CFR 55.25, "Incapacitation Because of Disability or Illness," for the licensee's failure to notify the NRC within 30 days of a change to one licensed senior operator's medical condition. Specifically, the licensed senior operator developed a permanent medical condition which caused him to permanently leave the site on December 1, 2014, and transition into a long-term disability program on April 23, 2015. The licensee did not notify the NRC of this change in medical condition. As a corrective action, the licensee initiated a license termination request for the affected operator, effective April 6, 2017. This issue was entered into the licensee's corrective action program as Notification 50912407.

The failure of the facility licensee to notify the NRC within 30 days of a change in a licensed senior operator's medical condition was a performance deficiency. This issue was evaluated using the traditional enforcement process because it negatively impacted the NRC's ability to perform its regulatory oversight function. Specifically, the failure to report changes in a licensed senior operator's medical condition prevented the NRC from taking action to issue either a license amendment or termination, as appropriate. This performance deficiency was determined to be Severity Level IV because it fits the Severity Level IV example of Enforcement Policy Section 6.4.d.1, "Violation Examples: Licensed Reactor Operators." This section states, "Severity Level IV violations involve, for example \x85 (b) an individual operator who did not meet the American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.4," "Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants," Section 5, "Health Requirements and Disqualifying Conditions," as certified on NRC Form 396, "Certification of Medical Examination by Facility Licensee," required by 10 CFR 55.23, Certification, but who did not perform the functions of a licensed operator or senior operator while having a disqualifying medical condition." No cross-cutting aspect was assigned because the violation was processed using traditional enforcement.

Inspection Report# : 2017002 *(pdf)*

**Significance: NOPD** Aug 10, 2017
Identified By: NRC
Item Type: NCV Non-Cited Violation
**Failure to Conduct Required Biennial Medical Examinations Within Two Years**

The inspectors identified a Severity Level IV non-cited violation of 10 CFR 55.21, ?Medical Examination,? for the licensee?s failure to ensure that a medical examination by a physician to determine satisfaction of 10 CFR 55.33(a)(1) requirements was conducted every two years for two licensed senior operators. Specifically, one licensed senior operator exceeded the two-year medical examination requirement by approximately 16 months between November 27, 2015, and April 6, 2017. A second licensed senior operator exceeded the two-year medical examination requirement by four months between November 19, 2016, and April 6, 2017. As a corrective action, the licensee has conducted the required medical examination for one senior operator and initiated a license termination request for the other senior operator. This issue was entered into the licensee?s corrective action program as Notification 50912407.

The failure of the facility licensee to conduct required biennial medical examinations for two licensed senior operators was a performance deficiency. This issue was evaluated using the traditional enforcement process because it negatively impacted the NRC?s ability to perform its regulatory oversight function. Specifically, the failure to comply with medical testing requirements for two operators compromised the facility licensee?s ability to assure conformance to medical standards, detect non-conforming medical conditions, and report non-conformances to the NRC. This performance deficiency was determined to be Severity Level IV because it fits the SL-IV example of Enforcement Policy Section 6.4.d.1, ?Violation Examples: Licensed Reactor Operators.? This section states, ?Severity Level IV violations involve, for example ? (b) an individual operator who did not meet the American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.4, ?Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants,? Section 5, ?Health Requirements and Disqualifying Conditions,? as certified on NRC Form 396, ?Certification of Medical Examination by Facility Licensee,? required by 10 CFR 55.23, Certification, but who did not perform the functions of a licensed operator or senior operator while having a disqualifying medical condition.? No cross-cutting aspect was assigned because the violation was processed using traditional enforcement. (Section 1R11.3)

Inspection Report# : 2017002 *(pdf)*

**Significance: N/A** Sep 12, 2016
Identified By: NRC
Item Type: VIO Violation
**Failure to Establish Adequate Work Instructions for Installation of Namco Snap Lock Limit Switches**
The inspectors identified a preliminary White finding associated with an apparent violation of Technical Specification 5.4.1.a, "Procedures," for the licensee's failure to develop adequate instructions for the installation, adjustment, and testing of Namco Model EA170 snap lock limit switches. Specifically, the licensee failed to provide site-specific instructions for limiting the travel of these external limit switches when installed in safety-related motor operated valves. Consequently, the lever switch actuator for valve RHR-2-8700B, residual heat removal pump 2-2 suction from the refueling water storage tank, was installed such that the limit switch was operated repeatedly in an over-travel condition resulting in a sheared internal roll pin that ultimately caused the limit switch to fail. Following identification of this issue, the licensee replaced the limit switch for valve RHR-2-8700B and implemented actions to modify maintenance procedures for installing, calibrating, and testing motor-operated valve external limit switches. The licensee entered this issue into their corrective action program as Notification 50852345.

The performance deficiency is more than minor, and therefore a finding, because it is associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, maintenance procedure MP E-53.10R, "Augmented Stem Lubrication for Limitorque Operated Valves," used to perform limit switch adjustments on the Unit 2 valve RHR-2-8700B, did not provide adequate acceptance criteria to prevent overtravel of the limit switch actuating lever. This resulted in a subsequent failure of the limit switch, preventing the open permissive signal for valve SI-2-8982B, residual heat removal pump 2-2 suction from the containment recirculation sump, used during the emergency core cooling system (ECCS) recirculation mode. The inspectors evaluated the finding using the Attachment 0609.04, "Initial Characterization of Findings," worksheet to Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," issued June 19, 2012. The attachment instructs the inspectors to utilize IMC 0609, Appendix A, "Significance Determination Process (SDP) for Findings At-Power," issued June 19, 2012. In accordance with NRC Inspection Manual Chapter 0609, Appendix A, Exhibit 2, "Mitigating Systems Screening Questions," the inspectors determined that the finding required a detailed risk evaluation because it represented an actual loss of function of the train B ECCS for greater than its technical specification allowed outage time. A senior reactor analyst performed a detailed risk evaluation in accordance with IMC 0609, Appendix A, Section 6.0, "Detailed Risk Evaluation." The calculated increase in core damage frequency was dominated by small and medium loss of coolant accident initiators with failures of the opposite train of ECCS or related support systems. The analyst did not evaluate the large early release frequency because this performance deficiency would not have challenged the containment. The NRC preliminarily determined that the increase in core damage frequency for internal and external initiators was 7.6E-06/year, a finding of low to moderate risk significance (White). The inspector did not identify a cross-cutting aspect with this finding because it was not reflective of current performance. The inadequate procedure was developed in 2011 and did not reflect the licensee's current performance related to procedure development.
(IR 05000275; 05000323/2016010, dated October 3, 2016, ML16277A340)

(FIRST UPDATE)
The finding was determined to be of low-to-moderate safety significance (White), because the NRC's calculated lower and upper estimations of the increase in core damage frequency of the performance deficiency were both greater than

1.0E-6 per year but less than 1.0E-5 per year. The NRC concluded that the preliminary significance determination change in core damage frequency result of 7.6E-6 per year represents the upper range of the increase in core damage frequency associated with the performance deficiency. Based on the information provided by the licensee at the November 15, 2016 regulatory conference, the NRC adjusted a number of assumptions used in the preliminary significance determination. Specifically, the NRC lowered the common cause alpha factors and adjusted several assumptions related to medium break loss-of-coolant accidents. The NRC also performed a variety of human error probability calculations to determine the likelihood of recovering the functionality of valve SI-2-8982B. The results of these calculations, which removed much of the conservativism from the assumptions used in the preliminary risk assessment, predicted a high likelihood of success (96.4 percent success) for recovering valve SI-2-8982B. Using these assumptions, the NRC concluded the lower range of increase in core damage frequency associated with the performance deficiency to be 1.3E-6 per year.
(Letter to E. Halpin from K. Kennedy, dated December 28, 2016, ML16363A429)
Inspection Report# : 2016010 *(pdf)*
Inspection Report# : 2017008 *(pdf)*

## Barrier Integrity
## Emergency Preparedness
## Occupational Radiation Safety
## Public Radiation Safety
## Security

The security cornerstone is an important component of the ROP, which includes various security inspection activities the NRC uses to verify licensee compliance with Commission regulations and thus ensure public health and safety. The Commission determined in the staff requirements memorandum (SRM) for SECY-04-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure," dated November 9, 2004, that specific information related to findings and performance indicators associated with the security cornerstone will not be publicly available to ensure that security-related information is not provided to a possible adversary. Security inspection report cover letters will be available on the NRC Web site; however, security-related information on the details of inspection finding(s) will not be displayed.

## Miscellaneous

Current data as of : February 01, 2018

*Page Last Reviewed/Updated Monday, November 06, 2017*