Diablo Canyon 2
3Q/2010 Plant Inspection Findings

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)*

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: 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: 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)*

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)*

Emergency Preparedness

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)*

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 : January 06, 2011