Diablo Canyon 2
2Q/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)*

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)*

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)*

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 : September 02, 2010