UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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July 23, 2010

John T. Conway
Senior Vice President-Energy &
Supply and Chief Nuclear Officer
Pacific Gas and Electric Company
Diablo Canyon Nuclear Plant
77 Beale Street, Mail Code B32
San Francisco, CA 94105

SUBJECT: DIABLO CANYON POWER PLANT - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000275/2010007 AND 05000323/2010007

Dear Mr. Conway:

On March 5, 2010, the US Nuclear Regulatory Commission (NRC) completed a component design bases inspection at your Diablo Canyon Power Plant. The enclosed report documents our inspection findings. The preliminary findings were discussed on March 4, 2010, with Mr. J. Becker, Site Vice President, and other members of your staff. After an additional in-office inspection, a final telephonic exit meeting was conducted on June 10, 2010, with Mr. J. Becker and others of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

Based on the results of this inspection, the NRC has identified ten findings that were evaluated under the risk significance determination process. Violations were associated with all of the findings. All ten of the findings were found to have very low safety significance (Green) and the violations associated with these findings are being treated as noncited violations, consistent with Section VI.A.1 of the NRC Enforcement Policy. In addition, a licensee identified violation, which was determined to be of very low safety significance is described in the report. If you contest any of the noncited violations, or the significance of the violations you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 612 East Lamar Blvd., Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Diablo Canyon Power Plant. In addition, if you disagree with the crosscutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at the Diablo Canyon Power Plant.

In accordance with Code of Federal Regulations, Title 10, Part 2.390 of the NRC's Rules of Practice, a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

T.R.

Thomas Farnholtz, Chief
Engineering Branch 1
Division of Reactor Safety

Dockets: 50-275; 50-323
License: DPR-80: DPR-82

Enclosure:
Inspection Report 05000275/2010007 AND 05000323/2010007
w/Attachments: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-275, 50-323
License:	DPR-80, DPR-82
Report Nos.:	05000275/2010007 and 05000323/2010007
Licensee:	Pacific Gas and Electric
Facility:	Diablo Canyon Power Plant, Units 1 and 2
Location:	7 ½ miles NW of Avila Beach Avila Beach, California
Dates:	February 8 through June 10, 2010
Team Leader:	R. Kopriva, Senior Reactor Inspector, Engineering Branch 1
Inspectors:	S. Makor, Reactor Inspector, Engineering Branch 1 B. Correll, Reactor Inspector, Engineering Branch 2 S. Hedger, Operations Inspector, Operations Branch
Accompanying Personnel:	H. Campbell, Mechanical Contractor, Beckman and Associates J. Nicely, Electrical Contractor, Beckman and Associates
Others:	T. Buchanan, Reactor Inspector, Plant Support Branch 2 M. Runyan, Senior Reactor Analyst, Division of Reactor Safety
Approved By:	Thomas Farnholtz, Branch Chief Engineering Branch 1

SUMMARY OF FINDINGS

IR 05000275/2010007; 05000323/2010007; 02/08/2010 through 06/10/2010; Diablo Canyon Power Plant: baseline inspection, NRC Inspection Procedure 71111.21, "Component Design Basis Inspection."

The report covers an announced inspection by a team of four regional inspectors, two contractors and one inspector in training. Ten findings were identified. All of the findings were of very low safety significance. The final significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified Findings

Cornerstone: Mitigating Systems

- Severity Level IV. 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)] (Section 2.4.b.i).

- Severity Level IV. 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 relay 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-049424, 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 Notifications 50302467 and 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 crosscutting 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 licensee did not thoroughly evaluate the original problem of using the 10 CRF 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)] (Section 2.4.b.ii).

- Green. 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 licensee'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] (Section 2.9.b.i).

- Severity Level IV. 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 section, 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 allowable 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 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 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 process. 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)] (Section 2.9.b.ii).

- Green. 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 230kV 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 230kV load tap changer. 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 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)] (Section 2.9.b.iii).

- Green. 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 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 (Section 2.9.b.iv).

- Green. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" involving multiple mechanical and electrical errors in documentation associated with torque, thrust, and operation of certain safety related motor operated valves. Specifically, on March 3, 2010, the team identified seven different numerical inputs to the operation, margin, and functionality of certain safety related motor operated valves, which had non-conservative inputs for the mechanical and electrical settings of the motor operated valves. The composite of these nonconservative inputs could have a detrimental effect on the operation of these motor operated valves. Contrary to above, on March 3, 2010, per 10 CFR Part 50, Appendix B, Criterion III, "Design Control", the licensee did not establish measures to assure that applicable regulatory requirements and the design basis, for those structures, systems, and components were correctly translated into specifications, drawings, procedures, and instructions. The licensee has entered this issue into their corrective action process as Notification 50302437.

  The failure of the licensee to correctly translate design basis information into specifications, drawing, procedures, and instructions 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. 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 inspectors reviewed the finding for crosscutting aspects and none were identified (Section 2.14.b).

- Green. 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 to 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 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)] (Section 4.b.i).

- Severity Level IV. 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 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 this violation is of very low safety significance and 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 (Section 4.b.ii)

- Severity Level IV. 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 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 this violation is of very low safety significance and 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 (Section 4.b.iii).

B. <u>Licensee-Identified Violations.</u>

The inspectors have reviewed a violation of very low safety significance, which was identified by the licensee. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. This violation and the applicable corrective action tracking numbers are listed in Section 4OA7.

REPORT DETAILS

1 REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and important design features may be altered or disabled during modifications. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This area inspected verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity Cornerstones for which there are no indicators to measure performance.

1R21 <u>Component Design Bases Inspection (71111.21)</u>

The inspectors selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than two or a Birnbaum value greater than 1E-6.

<u>Inspection Scope</u>

To verify that the selected components would function as required, the inspectors reviewed design basis assumptions, calculations, and procedures. In some instances, the inspectors performed calculations to independently verify the licensee's conclusions. The inspectors also verified that the conditions of the components were consistent with the design bases and that the tested capabilities met the required criteria.

The inspectors reviewed maintenance work records, corrective action documents, and industry-operating experience records to verify that licensee personnel considered degraded conditions and their impact on the components. For the review of operator actions, the inspectors observed operators during simulator scenarios, as well as during simulated actions in the plant.

The inspectors performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues as a result of modifications, and margin reductions identified because of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC inspectors' input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in-depth margins.

The inspection procedure requires a review of 20 to 30 total samples, including 10 to 20 risk-significant and low design margin components, 3 to 5 relatively high-risk operator actions, and 4 to 6 operating experience issues. The sample selection for this inspection was 18 components, 6 operator actions, and 5 operating experience items.

.2 Results of Detailed Reviews for Components

.2.1 Unit 1 - 125Vdc Class 1E Battery 12

a. Inspection Scope

The inspectors conducted a review of the Unit 1, 125Vdc safety-related battery bank (battery 12) to assess the design aspects of the battery. The inspectors reviewed sizing calculations, short circuit current calculations, coordination studies, design specifications, installation drawings, modifications made to the battery and battery rack, seismic tests and analysis, battery vendor manual, maintenance activities performed on the battery, and conducted a system walkdown with system engineering personnel to assess the material condition of the battery. A review of the battery testing methodology was conducted to verify the batteries are being tested to ensure that design requirements are being met. The inspectors conducted interviews with the battery system engineer and a battery maintenance technician. The inspectors also inspected the spare battery cell storage area to verify material condition of the spare cells and maintenance performed to preserve the condition of the spare cells.

b. Findings

No findings were identified.

.2.2 Unit 1 – 125Vdc Battery Charger ED12

a. Inspection Scope

The inspectors inspected the Unit 1, 125Vdc battery charger to ensure the charger meets design basis specifications. The inspectors reviewed short circuit calculations, sizing calculations, circuit breaker coordination studies, and voltage drop calculations to ensure the battery charger design criteria and maintenance requirements are met.

b. Findings

No findings were identified.

.2.3 Unit 1 - 125Vdc Distribution Bus – DC12

a. Inspection Scope

The inspectors inspected the Unit 1,125Vdc distribution bus to ensure design basis specifications were being met. The inspectors reviewed short circuit calculations, sizing calculations, circuit breaker coordination studies, voltage drop calculations, and circuit breaker maintenance activities to ensure the bus is designed and maintained to ensure design criteria were met. Circuit breaker sizing calculations and dc bus circuit breaker testing procedures were reviewed to ensure the installed circuit breakers were appropriate for the design of the system. Maintenance activities for the distribution bus and circuit breakers were verified to maintain the system according to manufacturer recommendations. Separation criteria for Class 1E loads were reviewed to ensure the dc bus met required separation criteria between Class 1E and Nonclass 1E loads.

b. Findings

No findings were identified.

.2.4 Unit 1 – 4160Vac Emergency Diesel Generator 12

a. Inspection Scope

The inspectors reviewed loading and voltage regulation calculations, including the bases for brake horsepower values used, to verify that design bases and design assumptions have been appropriately translated into the design calculations and procedures. The inspectors reviewed protection/coordination and short-circuit calculations to verify that the emergency diesel generator was adequately protected including short-circuit capability of the output breaker under worst fault conditions. The inspectors reviewed analyses and surveillance testing to assess emergency diesel generator operation under required operating conditions. The inspectors reviewed calculations and technical evaluations to verify that: (1) steady-state and transient loading are within design capabilities, (2) adequate voltage would be present to start and operate connected loads, and (3) operation at maximum allowed frequency would be within the design capabilities. The inspectors reviewed the dc control circuit loop analysis associated with the emergency diesel generator breaker trip/close circuits and spring charging motors to ensure adequate control voltage would be available. The inspectors reviewed the basis for the emergency diesel generator load sequence time delay setpoints, calibration intervals, and results of last calibration for accuracy. The inspectors reviewed the interfaces and interlocks associated with 4kV engineered safety feature switchgear bus 1G, including voltage protection schemes that initiate connection to the emergency diesel generator. Modifications to the system were reviewed against design documents to verify that performance capabilities of selected components had not been degraded. The inspectors reviewed selected industry operating experience and any plant actions to address the applicable issues to ensure that applicable insights from operating experience have been applied. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented and the component replacement was consistent with in service/equipment qualification life. The inspectors performed a visual non-intrusive inspection of observable portions of the emergency diesel generator to assess the installation configuration, material condition, and potential vulnerability to hazards. The inspectors reviewed the diesel generator design requirements concerning power factor limitations, and applicable surveillance requirements for diesel generator testing. The inspectors reviewed the licensee's actions following identification of a nonconservative technical specification dealing with diesel generator power factor values. For the emergency diesel generator voltage regulators, governor, and exciter control circuits, the inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. This review included the licensee's design basis documentation as well as various calculations, condition reports, procedures, test results, and operability determinations. The inspectors also performed walkdowns and performed interviews with design and system engineering personnel to ensure the capability of this component to perform its required function. Specifically, the inspectors reviewed control circuit permissives used to close breaker, protective circuit logic tests, and the instrument uncertainty calculations.

b. <u>Findings</u>

i. <u>Less than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update.</u>

<u>Introduction</u>. The inspectors identified a Severity Level IV 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 degraded voltage protection scheme as described in the Final Safety Analysis Report Update.

<u>Description</u>. The inspectors identified that Pacific Gas and Electric failed to perform a 10 CFR 50.59 evaluation prior to modifying the offsite power degraded voltage protection scheme. On March 12, 2010, Pacific Gas and Electric modified the protection scheme to raise the first level setpoint from 82 volts to 106 volts per Temporary Modification 60024240 (Unit 1) and 60024244 (Unit 2). The licensee implemented this modification as a compensatory measure to preserve the safety function of the protection scheme as required by Technical Specification 3.3.5, "Loss of Power Diesel Generator Start Instrumentation." The licensee developed the compensatory measure to restore the protection scheme operability. Plant engineers identified that the existing first level voltage setpoint was inadequate to ensure that damage would not occur to operating engineered safety feature pumps during degraded offsite power grid conditions. Plant engineers made this discovery while evaluating system operability after the inspectors identified that the protection scheme design was inconsistent with the design basis. The licensee previously established Calculation 357A-dc, "Load Flow, Short Circuit, and Motor Starting Analysis," as the plant safety analyses demonstrating the degraded voltage protection scheme met the design basis, including General Design Criteria 17, "Electric Power Systems." In December 2009, the NRC identified that the existing safety analysis did not

include the most limiting transient voltage cases or assumptions (ADAMS Accession Nos. ML093130428). To support the modification, the licensee re-performed the safety analysis on March 11, 2010, Calculation 900041017, "Evaluation of Temporary First Level Undervoltage Relays Set Point Changes," to include the most limiting cases and assumptions. The revised safety analysis concluded that General Design Criteria 17 would still be met after the modification was installed. On March 11, 2010, the licensee concluded that the modification did not require a 10 CFR 50.59 evaluation to determine if NRC approval was required prior to installation. The licensee's conclusion was based, in part, on the results of the revised safety analysis.

Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," stated that the methods described in NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. NEI 96-07, Section 4.2., "Screening," stated that a 10 CFR 50.59 evaluation is required for changes that adversely affect design functions, methods used to control design functions, or evaluations that demonstrate that intended design functions will be accomplished. Section 4.2 of NEI 96 07 also stated that if the safety analyses must be re-run to demonstrate that all required safety functions and design requirements are met, then the change is adverse and must be evaluated to determine if prior NRC approval is required. The inspectors concluded that the licensee was required to perform a 10 CFR 50.59 evaluation prior to installing the modification because the safety analysis had to be revised to include the most limiting transient voltage cases and assumptions to demonstrate that General Design Criteria 17 was met. Because the revised safety analysis concluded a reduction in available transient bus voltage and the modification increased the setpoint that would result in a disconnection of offsite power, the 10 CFR 50.59 evaluation would have likely concluded that the prior NRC approval would be required because the modification increased the probability of a spurious disconnection of offsite power during an anticipated transient or accident.

The inspectors concluded that the most significant contributor to the violation was nonconservative assumptions used by the licensee when making the decision not to perform a 50.59 evaluation. This decision was made on March 11, 2010, by senior plant management at a plant safety review committee meeting. Pacific Gas and Electric was relying on the installation of the modification to restore offsite power operability and to exit the 72-hour shutdown technical specification action entered on March 9, 2010. The licensee was under time pressure to complete the modification design, 10 CFR 50.59 screen, and installation before the technical specification allowed out-of-service time expired. Performance of 10 CFR 50.59 evaluation or obtaining prior NRC approval would have challenged the licensee to complete the compensatory measures prior to a required technical specification plant shutdown. The inspectors identified a previous example involving the licensee's failure to perform a required 10 CFR 50.59 evaluation in a time critical setting (NCV 05000323/2009005-05, "Less than Adequate Change Evaluation to the

Facility as Described in the Final Safety Analysis Report Update"), November 5, 2009. In this previous example, senior plant management at a plant safety review committee meeting also decided not to perform a required the 10 CFR 50.59 evaluation. Following the plant safety review committee's decision, the licensee began plant restart activities. The licensee subsequently concluded that prior NRC approval was required for the previous change (ADAMS Accession No. ML093580092). NRC approval for the November 5, 2009, change is still pending.

Analysis. 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 inspectors evaluated this issue using the traditional enforcement process, including NRC Enforcement Policy, Supplement I, Reactor Operations, 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 reasonable likelihood the change to the facility would have required 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 Mitigating Systems Cornerstone was affected because the change affected the availability and reliability of the 230kV offsite power system. The inspectors concluded that the violation screened as very low safety significance, Green, because the design deficiency was confirmed not to result in loss of operability. 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 human performance associated with the decision-making component because the plant safety review committee failed to use nonconservative assumptions when deciding not to perform the 10 CFR 50.59 evaluation. The committee did not adopt the requirement to demonstrate that the proposed action was safe in order to precede, rather than disapprove the action [H.1(b)].

Enforcement. Title 10 CFR 50.59, "Changes, Tests and Experiments," stated that a licensee may make changes in the facility as described in the Final Safety Analysis Report Update without obtaining a license amendment if the change does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the final safety analysis report. Regulatory Guide 1.187 stated that NEI 96-07 was acceptable for complying with the provisions of 10 CFR 50.59. NEI 96-07, Section 4.2, stated that if the safety analyses must be re-run to demonstrate that all required safety functions and design requirements are met, then the change is adverse and must be evaluated to determine if prior NRC approval is required. Contrary to the above, on March 11, 2010, Pacific Gas and Electric failed to demonstrate that a change to the degraded voltage setpoints and safety analyses, as described in the Final Safety Analysis Report Update, did not require a license amendment. This change required the licensee to re-run safety analyses to demonstrate that all required safety functions and design requirements were met. The licensee failed to evaluate this change to determine if more than a minimal increase in the likelihood of an occurrence of a malfunction of a structure, system, or component important to safety, previously evaluated in the Final Safety Analysis Report, occurred. This violation was entered into the licensee's corrective action program and is being treated as a Severity Level IV noncited violation NCV 05000323/2010007-01 "Less than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update."

ii. Failure to Adequately Evaluate Changes to the Diesel Testing as Described in the Final Safety Analysis Report Update

Introduction. The inspectors identified a Severity Level IV 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 diesel testing commitments as described in the Final Safety Analysis Report Update. Specifically, in 2004, the safety system functional audit and review recommended that the Final Safety Analysis Report Update be revised to clearly describe the applicability of the three different revisions to Regulatory Guide 1.9. While reviewing the associated documents describing the applicability to the different revisions of Regulatory Guide 1.9, the inspectors identified that (1) the licensee failed to perform an adequate evaluation for the frequency and voltage and criteria and (2) did not screen or evaluate changes made to their diesel testing commitments. These changes resulted in a departure from a method of evaluation described in the Final Safety Analysis Report Update establishing the facility design bases.

Description. In March 2010, the inspectors identified two examples of where the licensee performed a less than adequate evaluation. The first example was identified in the February 26, 1998, Diablo Canyon Power Plant Design Change Package (DCP E-049424). This example involved a Final Safety Analysis Report Update change from Safety Guide 9 to Regulatory Guide 1.9, Revision 2. Specifically, the licensee removed the KWS relay and adopted only the frequency and voltage recovery criteria from Regulatory Guide 1.9, Revision 2, because they could not consistently meet the recovery of frequency to 98 percent of nominal within 40 percent of the load block interval. As a result, the licensee changed their Final Safety Analysis Report Update commitment from Safety Guide 9 to Regulatory Guide 1.9, Revision 2, which involved a change in the recovery time for the voltage and frequency of the load sequence time interval from 40 to 60 percent.

The licensee was permitted to make changes to the facility as described in the final safety analysis report update without prior NRC approval, provided that these changes did not result in a departure from a method of evaluation described in the Final Safety Analysis Report Update and used in establishing the plant design bases. Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," stated that the methods described in NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. In Section 4.3.8 of NEI 96-07, "Does the Activity Result in a Departure from a Method of Evaluation Described in the Final Safety Analysis Report Update Used in Establishing the Design Bases or in the Safety Analyses," stated that licensees can make changes to elements of a methodology without first obtaining a license amendment if the results are essentially the same as, or more conservative than, previous results.

The inspectors concluded that the changes, an increase in voltage and frequency recovery values used in Class 1 component design, resulted in a less conservative result and a reduction in acceptance criteria for the margin of safety. The Pacific Gas and Electric 10 CFR 50.59 evaluation incorrectly answered question 7[no], "Is there a reduction in the margin of safety as defined in the basis for any technical specification?". As stated in NEI 96-07 gaining margin by changing one or more elements of a method of evaluation is considered to be a nonconservative change and a departure from a method of evaluation for purposes of 10 CFR 50.59. As stated in NEI 96-07 such departures required NRC approval before using the revised method.

The new methodology and frequency and voltage values used by the licensee were consistent with Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (ONSITE) Electric Power Systems at Nuclear Power Plants."

Section 4.3.8 of NEI 96-07, permitted the use of a new NRC approved methodology to reduce uncertainty, provide precise results, or other reason, provided the 10 CFR 50.59 evaluation demonstrated the following:

a. Based on sound engineering practice,
b. Appropriate for the intended application, and
c. Within the limitations of the applicable NRC safety evaluation report.

Item (c) required the licensee to demonstrate that use of the new methodology, Regulatory Guide 1.9, provided results that were essentially the same as or more conservative than, either the previous revision of the same methodology or another methodology previously accepted by NRC through issuance of a safety evaluation report. The licensee's evaluation did not address how the change was consistent with the limitations of applicable NRC safety evaluation reports.

The inspectors concluded that the licensing basis impact evaluation was less than adequate to demonstrate that prior NRC approval was not required to incorporate the new frequency and voltage recovery values.

The inspectors 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 for impacts to the facility. In Final Safety Analysis Report Update Section 8.3.1.1.13.7, it states, "Emergency diesel generator test scope and test interval frequency meets the applicable criteria of Regulatory Guide 1.9, Revision 3." The inspectors noted that this commitment was never evaluated for impact and implemented during the change to improved technical specifications and should have been at a minimum screened.

Analysis. 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 Mitigating Systems Cornerstone because the change modified the emergency diesel generator and diesel generator testing. 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 with meeting the frequency and voltage criteria and that the resolutions address causes and extent of conditions, as necessary [P.1(c)].

Enforcement. Title 10 CFR 50.59, "Changes, Tests and Experiments," stated that a licensee may make changes in the facility as described in the Final Safety Analysis Report Update 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 Update used in establishing the design bases or the safety analyses. Contrary to the above, on November 5, 2009, Pacific Gas and Electric changed the facility as described in the Final Safety Analysis Report Update to incorporate increased voltage and frequency recovery time for their emergency diesel generators. Because this finding is of very low safety significance and was entered into the corrective action program as Notifications 50302467 and 50302481, this violation is being treated as a Severity Level IV noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000323/2010007-02, "Failure to Adequately Evaluate Changes to the Diesel Testing as Described in the Final Safety Analysis Report Update."

.2.5 Unit 1 - 4160Vac ESF Switchgear Bus 1G

a. Inspection Scope

The inspectors inspected the 4kV switchgear to verify it would operate during design basis events. The inspectors reviewed selected calculations for electrical distribution system load flow/voltage drop, degraded voltage protection, short-circuit, and electrical protection and coordination. This review was conducted to assess the adequacy and appropriateness of design assumptions, and to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values under design basis conditions. Additionally, the switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst-case, short-circuit conditions. The inspectors' evaluated selected portions of the licensee response to NRC Generic Letter (GL) 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006. The station's interface and coordination with the transmission system operator for plant voltage requirements and notification setpoints were reviewed. The inspectors reviewed that the degraded and loss of voltage relay protection schemes. To determine if breakers were maintained in accordance with industry and vendor recommendations, the inspectors reviewed the preventive maintenance inspection and testing procedures. The 125Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open/close coils and spring charging motors. Finally, the inspectors performed a visual nonintrusive inspection to assess the installation configuration, material condition, and potential vulnerability to hazards.

b. Findings:

No findings were identified.

.2.6 Unit 1 - 480Vac Switchgear Bus Section (MCC) 1G and 4160/480Vac Load Center Transformer 1G:

a. Inspection Scope:

The inspectors reviewed selected calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination. The adequacy and appropriateness of design assumptions and calculations were reviewed to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values under design basis conditions. The switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst-case short circuit conditions. To ensure that breakers were maintained in accordance with industry and vendor recommendations, the inspectors reviewed the preventive maintenance inspection and testing procedures. The inspectors performed a visual non-intrusive inspection of observable portions of the safety related 480Vac MCC to assess the installation configuration, material condition, and the potential vulnerability to hazards. The inspectors assessed the sizing, loading, protection, and voltage taps for load center transformer 1G to ensure adequate voltage to the 480V MCC 1G. The inspectors reviewed the protective device settings to ensure that the feeder cables and transformer was protected in accordance with industry standards. A review of the testing requirements and preventive maintenance was performed. The inspectors performed a visual non-intrusive inspection of observable portions of the transformer to assess the installation configuration, material condition, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.7 Standby Start-up Transformers 11 and 12:

a. Inspection Scope

The inspectors reviewed the system one-line diagrams, nameplate data, and loading requirements to determine the adequacy of the transformers to supply required power to the associated 4160Vac buses. The inspectors reviewed periodic maintenance and testing practices to ensure the equipment is maintained in accordance with industry practices. Calculations for the automatic load tap changer voltage and time delay settings were reviewed to verify that design bases and assumptions have been appropriately translated into design calculations. Support system calculations and vendor information were reviewed in order to verify that energy sources, including those used for control functions would be available and adequate during accident/event conditions. The inspectors reviewed offsite power connections and the transmission operator notification protocols for the 230kV switchyard. The inspectors interviewed system engineers and performed a visual inspection of the transformers and their connection to the 230kV switchyard to assess the installation configuration, material condition, and potential vulnerability of the transformer to external hazards.

b. Findings:

No findings were identified.

.2.8 Reactor Trip Breakers

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. This review included the licensee's design basis documentation as well as various calculations, condition reports, procedures, test results, and operability determinations. The inspectors also performed walkdowns and performed interviews with design and system engineering personnel to ensure the capability of this component to perform its required function. Specifically, the inspectors reviewed protective relaying maintenance, protection coordination calculations, and relay settings.

b. Findings

No findings were identified.

.2.9 Degraded Voltage Relays

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, selected drawings, operating procedures, inservice testing and past corrective action documents. This review included the licensee's design basis documentation as well as various calculations, condition reports, procedures, test results, and operability determinations. Specifically, the inspectors reviewed the instrument accuracy calculation, which determined the setpoint and performs the scaling/uncertainty evaluation for degraded and loss of voltage relays.

b. Findings

i. Nonconservative Decision Making resulted in a Violation of Technical Specification

Introduction. The inspectors identified a Green, noncited violation of Technical Specification 3.7.7, "Vital Component Cooling Water System," after both Unit 2 component cooling water loops were inoperable for a greater duration than permitted by the out-of-service time.

Description. On March 10, 2010, the inspectors identified that both Unit 2 vital component cooling water loops had been inoperable for a greater period than permitted by plant technical specifications during power operations. On March 9, 2010, Pacific Gas and Electric identified that the degraded voltage protection scheme was inadequate to ensure that the minimum required voltages would be available to operating engineered safety feature pumps during degraded offsite power grid conditions. 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 determined that the first level degraded voltage setpoints specified in Technical Specification 3.3.5, "Loss of Power Diesel Generator Start Instrumentation," were inadequate to ensure plant safety. Technical Specification 3.3.5 required that the licensee declare the corresponding diesel generator inoperable following the inoperability of the degraded voltage protection on an engineered safety feature 4.16 kV bus. Because the nonconforming condition affected all three engineering safety feature buses on each unit, plant technical specifications required the licensee to declare all three-diesel generators, on each unit, inoperable. Technical Specification 3.8.1, "AC Sources – Operating," prohibited continued reactor operation with two or more diesel generators inoperable on each unit. As an alternative to a dual unit plant shutdown, the licensee declared the 230kV offsite power system inoperable and applied the 72-hour out-of-service time provided by Technical

Specification 3.8.1. The licensee reasoned that the degraded voltage scheme required by Technical Specification 3.3.5 was not needed if the engineered safety feature buses would directly transfer to the diesel generators following a unit trip. The licensee implemented compensatory actions to inhibit the automatic transfer of busses with operating engineered safety feature pumps, to the 230kV offsite power source. The inspectors concluded that this action would have preserved the operating pump safety function if a design bases event occurred.

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. Technical Specification 3.7.7 required the licensee to either place the unit in Mode 3 or restore an operable loop within 6 hours. The licensee subsequently inhibited the automatic transfer of the buses supporting component cooling water pump 2-3 to the 230kV offsite power, restoring operability of the loop. The inspectors concluded that Pacific Gas and Electric operated Unit 2 without an operable vital component cooling water loop for greater than 14 hours (7:16 p.m. on March 10, 2010, through 8:58 a.m. on March 11, 2010). The inspectors concluded that the most significant contributor to the violation was nonconservative assumptions use by plant operators when deciding which engineered safety feature buses to inhibit the automatic transfer to the 230kV power source. Plant operators did not adopt the action necessary to demonstrate that the component cooling water system specific safety function was maintained by the proposed action before proceeding rather than a requirement to demonstrate that it is unsafe in order to disapprove the action.

Analysis. The failure of Pacific Gas and Electric to either 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 finding was more than minor because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. The inspectors 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 inspectors concluded that the finding was of very low safety significance 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 time that a vital component cooling water loop unavailable due to the performance deficiency. The inspectors concluded that this finding had a crosscutting aspect in the area 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)].

Enforcement. Technical Specification 3.7.7 required the licensee to place Unit 2 in Mode 3 within 6 hours without an operable vital component cooling water loop. Contrary to the above, on March 11, 2010, Pacific Gas and Electric failed to place Unit 2 in Mode 3 within 6 hours following the inoperability of both Unit 2 vital component cooling water loops. Because this finding is of very low safety significance and was entered into the corrective action program as Notification 50304802, this violation is treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000323/2010007-03, "Nonconservative Decision Making resulted in a Violation of Technical Specification."

ii. Failure to Update the Final Safety Analysis Report with the Current Plant Design Bases.

Introduction. The inspectors identified a Severity Level IV 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.

Description. The inspectors identified that Pacific Gas and Electric failed to update the Final Safety Analysis Report Update with the current offsite degraded voltage protection scheme design basis. The failure of the Final Safety Analysis Report Update to reflect current plant design basis information 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. Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report in Accordance With 10 CFR 50.71(e)," September 1999, stated that Nuclear Energy Institute (NEI), NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," June 1999, provided methods acceptable to the NRC staff for complying with the provisions of 10 CFR 50.71(e) for Final Safety Analysis Report Update updates. NEI 98-03, Section 6.1, "What the Regulations Require," stated that the Final Safety Analysis Report Update was required to include new or modified design bases (as defined in 10 CRF 50.2) and a summary of new or modified safety analyses. The inspectors identified that Final Safety Analysis Report Update, Section 8.1.4, "Electric Power Design Bases," did not include the design bases for either the allowable time delay or the limiting voltage setpoints for the offsite power degraded voltage protection scheme. The inspectors determined that the licensee's less than adequate evaluation of a previous problem was the most significant contributor to this violation. The inspectors previously identified NCV 05000275 and 323/2009003-03, "Failure to Update the Final Safety Analysis Report Update with Current Plant Design Criteria," in June 2009. This previous issue also related to examples where the licensee had failed to properly update plant design bases information in the Final Safety Analysis Report Update. The inspectors concluded that Pacific Gas and Electric had opportunity to identify and correct the current violation if the licensee had performed an adequate extent of condition evaluation of this previous issue.

Analysis. The failure of Pacific Gas and Electric to update the Final Safety Analysis Report Update with the current plant design basis was a performance deficiency. Because the issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated this violation using the traditional enforcement process. The inspectors used the General Statement of Policy and Procedure for NRC Enforcement Actions, Supplement I, "Reactor Operations," dated January 14, 2005 to evaluate the significance of this violation. 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 classified the violation as 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 also 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 condition and take appropriate corrective actions after the NRC identified a similar violation [P.1(c)].

Enforcement. Title 10 CFR 50.71(e) required Pacific Gas and Electric to periodically update the Final Safety Analysis Report Update originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. This submittal was required to contain all the changes necessary to reflect information and analyses submitted to the Commission by the applicant or licensee or prepared by the applicant or licensee pursuant to Commission requirement since the submittal of the original Final Safety Analysis Report Update. Regulatory Guide 1.181 stated that NEI 98-03 provided methods acceptable to the NRC staff for complying with the provisions of 10 CFR 50.71(e). As stated in NEI 98-03 the Final Safety Analysis Report Update was required to include new or modified design bases and a summary of new or modified safety analyses. Contrary to the above, on February 25, 2010, 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. Because this finding is of very low safety significance and was entered into the corrective action program as Notification 50313763, this violation is being treated as a Severity Level IV, noncited violation in accordance with Section VI.A.1 of the Enforcement Policy: NCV 05000275; 05000323/2010007-04, "Failure to Update the Final Safety Analysis Report with the Current Plant Design Bases."

iii. Inadequate Operability Determination Associated With The Offsite Degraded Voltage Protection Scheme

Introduction. The inspectors identified a violation of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," after Pacific Gas and Electric failed to properly perform an operability determination, per Operability Determination Procedure OM7.ID12, "Operability Determination," Revision 14, following discovery of a nonconforming condition associated with the offsite degraded voltage protection scheme.

Description. On February 27, 2010, Pacific Gas and Electric failed to complete an adequate operability evaluation, as required by Operability Determination Procedure OM7.ID12, after the inspectors identified a nonconforming condition affecting the safety function of the offsite power degraded voltage protection scheme. The protection scheme design basis included the specified safety function to protect engineered safety feature equipment from damage from degraded offsite grid condition while preserving the core cooling response times assumed in the accident analysis. The protection scheme design performed this function by restricting the duration and voltage that engineered safety feature equipment could be exposed to degraded grid conditions. This time delay should have been bound by the overall time delay assumed for engineered safety feature actuation in the safety analyses, between 6 and 10 seconds. The inspectors identified that the actual time delay was 20 seconds. The additional delay would have prevented full emergency core cooling system flow within the 25 seconds assumed 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.IED12. On February 27, the plant operating authority concluded that the protection scheme was operable based on the information provided in the operability determination.

On March 2, 2010, the inspectors concluded that the licensee's operability determination was inadequate to demonstrate protection scheme operability. Plant engineering only addressed the capability of the protection scheme at normal grid voltage following a mechanical failure of the 230kV 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. As described in Supplemental Safety Evaluation Report 9, the protection scheme specified safety function included protection of engineered safety feature equipment from actual degraded grid conditions. The licensee re-evaluated the protection scheme operability and on March 9, 2010, concluded that the offsite power degraded voltage protection scheme was inoperable. The performance deficiency resulted in the Technical Specification 3.3.5, "Loss of Power Diesel Generator Start Instrumentation," allowed out of service time to be exceeded. Technical Specification 3.3.5 required the licensee to immediately restore the safety function or declared the corresponding emergency diesel generator inoperable. The inspectors concluded that the most significant contributor to the finding was a less than adequate evaluation by plant engineers of the original nonconforming condition identified by the NRC.

Analysis. The inspectors concluded that the failure of the licensee to adequately evaluating the operability of the degraded voltage protection scheme, in accordance with Operability Determination Procedure OM7.ID12, was a performance deficiency. 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. 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 represent the actual loss of safety function for greater than the Technical Specification 3.3.5 allowed outage time. The inspectors concluded that 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 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)].

Enforcement. Title 10 CFR Part 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," required that activities affecting quality be accomplished in accordance with instructions or procedures. Operability Determination Procedure OM7.ID12, Section 5.3, required that operability determinations address the potential effect of nonconforming conditions to perform the specified safety function. Contrary to the above, on February 27, 2010, Pacific Gas and Electric failed to perform an activity affecting quality in accordance with Operability Determination Procedure OM7.ID12. The operability determination, performed as part of Notification 50301167, did not address the potential effect of nonconforming conditions to perform the specified safety function during actual degraded grid conditions. Because this finding is of very low safety significance and was entered into the corrective action program as Notification 50301167, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000323/2010007-05, "Inadequate Operability Determination."

iv. Second Level Undervoltage Relay Time Delay to Initiate Load Shed and Sequencing Upon the Diesel Generator is Adequate to Assure Plant Safety

Introduction. The inspectors identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" involving the licensee's failure to assure and verify that Technical Specification 3.3.5 "Loss of Power Diesel Generator Start Instrumentation," (SR3.3.5.3) pertaining to the second level undervoltage relay time delay of 20 seconds to initiate load shed and sequencing upon the diesel generator is adequate to assure plant safety.

Description. Requirements in Supplemental Safety Evaluation Report 9, Section 8.1 requires that 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. The Final Safety Analysis Report Update Chapters 6 and 15 accident analyses assume that full engineered safety feature flow be achieved within 25 seconds; this includes the time for emergency diesel generators to start and reach rated speed, all load sequencing, time delays and valve repositioning. The Final Safety Analysis Report Update analyses are based on a worst-case design basis event concurrent with a loss of offsite power. The licensee previously had not considered a degraded voltage concurrent with a design basis event to be credible, and had therefore not analyzed for it. Contrary to the Supplemental Safety Evaluation Report 9, requirement, the licensee's second level undervoltage relay time delay setpoint of 20 seconds is not enveloped by (exceeds) the Final Safety Analysis Report Update accident analysis assumptions. Additionally, contrary to Supplemental Safety Evaluation Report 9, Section 8.1, Subsection (1)(c)(i), the allowable second level undervoltage relay time delay is not reflected in the Final Safety Analysis Report Update accident analyses and per subsection (1)(c)(iii) a degraded voltage condition for the allowable second level undervoltage relay time duration was not evaluated to determine that there would be no failure of safety systems or components. At the time the issue was identified, the licensee did not have an analysis that determines the impact on vital 4kV and 480V buses and on the Final Safety Analysis Report Update accident analyses of a sustained under voltage condition on the 230kV system that is within the settings of the first level and second level undervoltage relays. Preliminary licensee analyses indicated that the resulting required engineered safety feature flow may exceed the Final Safety Analysis Report Update accident analyses by up to 50 seconds and therefore the total impact to safety related equipment and operability of the affected systems were unknown at that point. The licensee entered the issue into their corrective action program as Notification 50301167.

The licensee while analyzing the consequences of design basis accidents concurrent with a postulated degraded grid voltage concluded that both Units 1 and 2 were in an unanalyzed condition. The postulated sustained degraded voltage condition could have resulted in normally operating safety-related motors tripping on over current and as a result, these pumps would not have been immediately available to mitigate a postulated accident as credited in the Final Safety Analysis Report Update accident analyses. On March 9, 2010, the licensee reported this as an unanalyzed condition (8-hour report) to the NRC in accordance with 10 CFR 50.72(b)(3)(ii)(B). In the interim, the licensee implemented compensatory measures via Shift Orders to prevent auto transfer to startup power of one vital 4kV bus per unit, which resulted in offsite startup power being also considered inoperable in both Units 1 and 2 and entered 72-hour Technical Specification 3.8.1 Action A.2 for one inoperable offsite power source.

On March 12, 2010, the licensee modified the protection scheme to raise the first level setpoint from 82 volts to 106 volts per Temporary Modification 60024240 (Unit 1) and 60024244 (Unit 2). The licensee implemented this modification as a compensatory measure to preserve the safety function of the protection scheme as required by Technical Specification 3.3.5. The licensee developed the compensatory measure to restore the protection scheme operability. Plant engineers identified that the existing first level voltage setpoint was inadequate to ensure that damage would not occur to operating engineered safety feature pumps during degraded offsite power grid conditions.

The licensee additionally determined that for the identified scenario an increase in the expected time to establish full-engineered safety feature flow would be seven seconds in addition to the established Final Safety Analysis Report Update accident analysis. The licensee contracted with Westinghouse Electric to evaluate the increase of seven seconds to the previously analyzed accident analyses and concluded that it would be acceptable.

The licensee's modifications, compensatory measures, and re-evaluation of the Final Safety Analysis Report Update accident analyses have given the inspectors an assurance that the plant would be capable of performing its required safety functions during a degraded voltage condition concurrent with a design basis event.

Analysis. The inspectors determined that failing to verify the adequacy of the design of the essential 4160Vac and 480Vac distribution systems during a degraded voltage condition concurrent with a design basis event was a performance deficiency. The inspectors concluded that the finding was more than minor in accordance with Inspection Manual Chapter 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Disposition Screening," issued on September 20, 2007. The finding involved the attribute of design control and could have 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 inspectors determined the failure to assure and verify that operation of the required safety loads within the analyzed Final Safety Analysis Report Update accident analysis could have affected the capability of safety-related equipment to respond to initiating events. By the end of the inspection, the licensee was able to demonstrate operability, however, at the time of discovery, there was reasonable doubt on the operability of the essential 4160V and 480V auxiliary power system and required essential loads that would have operate for a design basis event. The inspectors evaluated the significance of this issue under the Significance Determination Process using Inspection Manual Chapter 0609, Appendix A, "Determining the

Significance of Reactor Inspection Findings for At Power Situations." The inspectors concluded that 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 exposure time that protection scheme was available due to the performance deficiency. The team reviewed the finding for crosscutting aspects and none were identified.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," required, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculation methods, or by the performance of a suitable testing program. Contrary to the above, as of March 2010, the licensee had failed to ensure the adequacy of voltages to the 4160 and 480Vac equipment in support of mitigating system loads during a degraded voltage concurrent with a design basis event. The licensee entered this issue into the corrective action program as Notification 50301167. The inspectors determined this finding and violation of regulatory requirements to be of very low safety significance (Green) and were not judged to reflect current licensee performance. Because this finding is of very low safety significance and was entered into the corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000323/2010007-06, "Second Level Undervoltage Relay Time Delay to Initiate Load Shed and Sequencing upon the Diesel Generator is Adequate to Assure Plant Safety."

.2.10 Refueling Water Storage Tank

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, and selected drawings, past corrective action documents, condition reports and operability determinations to verify that the tank was capable of supplying the required volume of borated water during design basis accidents. Further, the inspectors inspected the design features and vendor-supplied testing features of the vortex suppression modification. Finally, the inspectors reviewed tank level scaling and instrument uncertainty calculations, and recent operator surveillance recorded data to ensure that technical specification requirements were satisfied.

b. Findings

No findings were identified.

.2.11 Motor-Driven Auxiliary Feedwater Pump 1-2

a. Inspection Scope

The inspectors performed a walkdown of auxiliary feedwater system pump 1-2, followed by a discussion with current and past system engineers. The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, selected drawings, past corrective action documents and condition reports, calculations, and operability determinations pertinent to this pump to verify its ability to accomplish design required functions. Particular attention was given to the testing procedures and associated instrumentation used to demonstrate adequacy of the pump to perform its required design basis functions. The inspectors reviewed completed surveillance tests to confirm the acceptance criteria and test results that demonstrated the capability of the pump to provide required flow rates at corresponding pump head for the specified required accident scenarios. Further, inservice test results were reviewed to assess potential component degradation and impact on design margins.

The inspectors also reviewed calculations that establish voltage drop, protection and coordination, motor break horsepower requirements, and short circuit for the motor power supply and feeder cable to verify that design bases and design assumptions have been appropriately translated into design calculations

b. Findings

No findings were identified.

.2.12 Residual Heat Removal Pump 1-1

a. Inspection Scope

The inspectors performed a walkdown of residual heat removal pump 1-1 followed by a discussion with design and system engineers. The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, and selected drawings, past corrective action documents, condition reports and operability determinations to verify the capability of this residual heat removal pump to operate under post-accident conditions. The inspectors reviewed pump specifications and pump curves to ensure that these parameters had been correctly translated into calculations, as required. Inservice test results were reviewed to assess potential component degradation and impact on design margins. In addition, the licensee responses and actions to Bulletin 88-04, "Potential Safety-Related Pump Loss," were reviewed to assess implementation of operating experience. The calculations that addressed net positive suction head, available and required, during the postulated recirculation phase, were also reviewed.

The inspectors reviewed calculations that establish voltage drop, protection and coordination, motor break horsepower requirements, and short circuit for the motor power supply and feeder cable to verify that design bases and design assumptions have been appropriately translated into design calculations

b. Findings

No findings were identified.

.2.13 Containment Supply Fan S-43/44 and Exhaust Fans E-43/44

a. Inspection Scope

Supply fans S-43/44 and exhaust fans E-43/44 provide coolant air to the 480V Switchgear and Inverter rooms. The inspectors reviewed the Final Safety Analysis Report Update, heating, ventilation, and air conditioning (HVAC) training manuals, system design criteria, current system health report, past corrective documents and several system drawings to verify their capability to operate and satisfy design basis requirements under post-accident conditions. Further, the inspectors reviewed calculations related to design changes affecting room heat loads, and undertook discussions with HVAC system engineers.

b. Findings

No findings were identified.

.2.14 Residual Heat Removal Pump 1-1 Suction Valve Motor Operator Valve 8700A

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, current system health report, selected drawings, testing procedures, inservice test results, past corrective action documents and condition reports, calculations, and operability determinations, including the licensee's design basis documentation. Further, the inspectors performed a walkdown of valve and undertook several discussions with the motor operated valve coordinator. Specifically, the inspectors inspected the applicable motor-operated valve calculations, testing procedures and results, and modifications related to the valve to verify its capability to operate under post-accident conditions.

The inspectors reviewed the calculations for the degraded voltage at the motor operated valve terminals, to ensure the proper voltage was utilized in the review of the motor operated valve torque calculations. The inspectors reviewed motor surveillance data and calculations that establish control circuit voltage drop, short circuit, and protection/coordination including thermal overload sizing and application. Additionally, motor control center thermal overload testing and bypass programs were reviewed.

b. Findings

Introduction. The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" involving multiple mechanical and electrical errors in documentation associated with torque, thrust, and operation of certain safety related motor operated valves. Specifically, on March 3, 2010, the inspectors identified seven different numerical inputs and/or criteria for the operation, margin, and functionality of certain safety related motor operated valves, which had nonconservative inputs for the mechanical and electrical settings of the motor operated valves.

Description. During the inspection period, the inspectors reviewed several attributes of residual heat removal pump 1-1 suction valve motor operated valve 8700A. These included both mechanical and electrical attributes that affect the operation of motor operated valves. While reviewing the calculations and procedures governing the setting of torque and thrust equations, operation, and testing of safety related motor operated valves, the inspectors identified several errors, or apparent nonconservative values used in the procedures and calculations. These errors included:

i. The licensee considered the starting current for motor-operatored valves during an accident as negligible in electrical calculations (357, 359), where the NRC typically finds that these calculations are modeled to verify starting currents.

ii. In calculation 195F-DC, the inspectors identified two areas of nonconservatism and one error:

   a. In the calculation, the cable resistance was normalized to 40 degrees Celsius. The industry-accepted practice is to use 90 degrees Celsius.

   b. The starting power factor for motor operator valves used in the calculation was 60 percent. IEEE 1290, Section 4.2.1.3, indicates that power factor range for starting motor-operatored valves is 60-95 percent. Industry operator experience suggests the use of 90 percent. According to the calculation, the operator motor vendor, Reliance, provided a power factor range of 60-80 percent. The higher power factor ratings are more conservative and should have been used in the calculation.

   c. In Table 2, page 104, motor operated valve 8700A overload heater is designated as model H32. This overload heater is actually model FH32 (according to calculation 195C-DC) and has a different resistance, which would be a different input into the calculation.

iii. Regarding calculation 192A-dc, the inspectors identified that the available voltage was 419Vac and the bounding circuit voltage is 418Vac, leaving a 1V margin. In addition, the inspectors identified that the licensee used 50 degrees Celsius as the cable temperature. The typical, more conservative temperature used for this type of cable is 75 degrees Celsius. The inspectors questioned the impact of the temperature difference on the 1V margin at a higher cable temperature.

iv. The licensee has not evaluated the terminal voltage at the motor operated valves at the time each motor operated valve must function during transient voltage conditions.

v. The square root of the sum of the squares method Diablo Canyon power plant uses when combining instrument uncertainties in motor operated valve differential pressure tests will always result in a 2.8 percent correction to the difference between the readings of instruments with plus or minus 2 percent accuracy. Note that the full-scale range of the instruments should affect the total uncertainty when taking the difference between upper and lower pressures. For example, if both instruments had a full-scale range of 100 psig and 2 percent accuracy, the uncertainty should be approximately 5.4 percent of the measured value.

vi. A "head correction worksheet," used in some differential pressure test evaluations, contained a tank head formula that was incorrect. This causes a higher than actual differential pressure to be used in the test evaluation, resulting in a potential nonconservatism.

vii. The component cooling water surge tank head, calculated as downstream pressure in a number of component cooling water motor operated valve differential pressure tests, did not account for tank cover gas pressure. This causes a higher than actual differential pressure to be used in the test evaluation, resulting in a potential nonconservatism.

The composite of these identified errors and nonconservative inputs into calculations and procedures that govern the operation and setting of safety related motor operated valves, brought into question whether the motor operated valves would functioned as designed in all required conditions. The inspectors questioned the licensee on the operability of the safety related valves under all conditions in which the valves were required to operate. The licensee believed the errors and the nonconservative inputs to be small values, and therefore, they would not have an appreciable impact on the settings and operation of the safety related valves, but they were unable to confirm that all of the required safety related valves would operate as expected under all emergency conditions. The inspectors concluded that these valves were in an unanalyzed condition. The licensee entered these concerns into corrective action document Notification 50302437. Upon completion of Notification 50302437, the licensee re-analyzed all of the inputs into the operation of the safety related motor operated valves, and concluded that there was a decrease in margin, but all of the safety related motor operated valves would functioned at designed.

Analysis. Failure to correctly translate design basis information into specifications, drawing, procedures, and instructions 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. 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 inspectors reviewed the finding for crosscutting aspects and none were identified.

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states in part that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Contrary to the above, on March 3, 2010, the inspectors identified several numerical inputs for the torque, thrust, operation, functionality, and margin of certain safety related motor -operated valves which had nonconservative inputs for the mechanical and electrical settings associated with the operability of these motor operated valves. After using corrected inputs, the licensee determined that the motor operated valves were operable. The licensee has entered this issue into their corrective action process as Notification 50302437. Because this finding is of very low safety significance (Green) and was entered into the corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000275/2010007-07, "Nonconservative Inputs into Motor Operated Valve Calculation."

.2.15 Residual Heat Removal Discharge to Charging Pump Motor Operator Valve 8804A

a. Inspection Scope:

The inspectors performed a walkdown of residual heat removal pump discharge to charging pump motor operated valve. The inspectors also held several discussions with the motor operated valve system coordinator. The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, and current system health report, selected drawings, and testing procedures, inservice test results, past corrective action documents and condition reports, and operability determinations to ensure that this valve could satisfy design requirements. Specifically, the inspectors inspected the applicable motor-operated valve calculations, testing procedures and results, and modifications related to the valve to verify its capability to operate under post-accident conditions.

The inspectors reviewed the calculations for the degraded voltage at the motor operator valve terminals, to ensure the proper voltage was utilized in the inspectors' review of the motor-operatored valve torque calculations. The inspectors reviewed the calculations that establish control circuit voltage drop, short circuit, and protection/coordination including thermal overload sizing and application. Additionally MCC thermal overload testing and bypass programs were reviewed.

b. Findings

No findings were identified.

.2.16 Motor-Driven Auxiliary Feedwater Pump 1-2 Discharge Check Valve

a. Inspection Scope

The inspectors performed a walkdown of motor-driven auxiliary feedwater pump 1-2 including the pump discharge check valve. The inspectors held discussions with the system coordinator. The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, and current system health report, selected drawings, and testing procedures, inservice test results, past corrective action documents and condition reports, and operability determinations to ensure that this valve could satisfy design requirements. The inspectors also reviewed the vendor manual for correct installation criteria, testing procedures and results, and modifications related to the valve to verify its capability to operate under post-accident conditions.

b. Findings

No findings were identified.

.2.17 Unit 1 – Turbine-Driven Auxiliary Feed Pump Steam Stop Valve FCV-95 (electrical)

a. Inspection Scope

The inspectors reviewed the voltage drop calculation and motor-operated valve motor sizing calculation to ensure the electrical supply to the valve met design requirements, and compared these requirements with the mechanical torque requirements to ensure the valve is capable to perform its intended function.

b. Findings:

No findings were identified.

.2.18 Emergency Diesel Generator Fuel Oil Transfer System

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report Update, system design criteria, system flow diagrams, operating procedures, technical specifications, calculations, and license amendments. Specifically, the inspectors inspected the applicable fuel oil storage requirement calculations, operating procedures and modifications related to the fuel oil transfer system to verify its capability to operate under post-accident conditions and to assure that the current capacity of the tanks and capabilities of the transfer system are sufficient to meet the design basis and technical specification requirements.

b. Findings

No findings were identified.

.3 Results of Reviews for Operating Experience

.3.1 Inspection of Generic Letter 2007-01, "Inaccessible of Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients"

a. Inspection Scope

The inspectors reviewed the licensee response to NRC Regulatory Guide 2007-01 "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," dated February 7, 2007, and randomly selected pull-box inspections over the past 3 years in order to assess the adequacy of the maintenance activities associated with submerged or underground cables. The inspectors verified that the licensee's program is consistent with the response provided to the NRC for the generic letter, and verified that the program is maintaining the system in accordance with the response provided.

b. Findings

No findings were identified.

.3.2 Inspection of Information Notice 2006-31, "Inadequate Fault Interrupting Ratings of Circuit Breakers"

a. Inspection Scope

The inspectors reviewed the applicability/actions taken by Diablo Canyon Power Plant to address Information Notice 2006-31 that addressed concerns of inadequate fault interrupting rating of breakers. Diablo Canyon Power Plant evaluated the information notice under Action Request A0685732. Diablo Canyon Power Plant electrical calculations show that the postulated worst-case fault currents are within the breaker, switchgear, and bus bracing ratings and therefore no additional actions were required.

b. Findings

No findings were identified.

.3.3 Inspection of Information Notice 2007-34 "Electric Circuit Breakers"

a. Inspection Scope:

The inspectors reviewed the applicability/actions taken by Diablo Canyon Power Plant to address Information Notice 2007-34 that addressed concerns operating history and maintenance issues with circuit breakers. Diablo Canyon Power Plant reviewed the information notice under Action Request A0710423. The Diablo Canyon Power Plant review concluded that sufficient circuit breaker programmatic standards are in place that are in accordance with the NRC recommendations stated in the information notice and that additional corrective actions are not required.

b. Findings

No findings were identified.

.3.4 Inspection of Information Notice 2009-10, "Transformer Failures"

a. Inspection Scope

The inspectors reviewed the applicability/actions taken by Diablo Canyon Power Plant to address Information Notice 2009-10 that addressed concerns that large power transformer failures have resulted in eight declared plant events from 2007 to 2009. Diablo Canyon Power Plant reviewed the information notice under Notification 50254726. Some of the events described in the information notice were applicable to Diablo Canyon Power Plant and additional actions are documented in AR 0703222, DN 50043092, and system analysis and program development, Order 0005406.

b. Findings:

No findings were identified.

.3.5 Inspection of Information Notice 2006-26, "Failure of Magnesium Rotors in Motor Operator Valve Actuators"

a. Inspection Scope

The inspectors reviewed NRC Information Notice 2006-26, which documented recent failures of motor-operated valve actuators because of galvanic corrosion, general corrosion, and/or thermally induced stress. These failures highlight the particular vulnerabilities of motor actuators with magnesium rotors, particularly when the motor is located in a high humidity and/or high temperature environment. These motor-operated valve failures illustrate the necessity of adequate inspection and/or preventive maintenance on actuators manufactured with magnesium rotors. The inspectors reviewed current inspection work orders instructions, and actual inspection documentation for inspections performed.

b. Findings:

No findings were identified.

.4 Results of Reviews for Operator Actions

The inspectors selected risk-significant components and operator actions for review using information contained in the licensee's individual plant examination probabilistic risk assessment. In addition, operator actions that had certain assumptions made about them in the Final Safety Analysis Report Update document were evaluated. This included components and operator actions that had a risk achievement worth factor greater than 1.3 or Fussell-Vesely values greater than 0.30.

a. Inspection Scope

For the review of operator actions, the inspectors observed operators during simulator scenarios associated with the selected components as well as observing simulated actions in the plant.

Inspection Procedure 71111.21 requires a review of three to five relatively high-risk operator actions. The sample selection for this inspection was six operator actions.

The selected operator actions were:

- A bounding case major steam line break with feedwater regulating valve failure and containment fan cooler impairment (Scenario)

- A loss of offsite power with the turbine-driven auxiliary feedwater pump 1-1 out of service, a differential trip of Bus G, a failure of emergency diesel generator 1-1 to start, and auxiliary feedwater pump 1-3 motor failure (scenario)

- A bounding case main feedwater line break with auxiliary feedwater pump 1-3 out of service (scenario)

- Establish makeup to the auxiliary feedwater system via the Diablo Canyon Creek via the raw water storage reservoir pools (job performance measure)

- Establish decay heat removal by cross tying the auxiliary saltwater system to the auxiliary feedwater system via the component cooling water heat exchangers (job performance measure)

- Align emergency makeup to the component cooling water system expansion tank via the firewater storage tank to support time requirement to provide makeup and isolate a component cooling water system leak (job performance measure)

b. Findings

i. Inadequate Drawings and Procedures to Align Emergency Makeup Water Supply from Diablo Canyon Creek to Support the Auxiliary Feedwater System

Introduction. The inspectors identified a noncited violation of Diablo Canyon Technical Specification 5.4.1. "Procedures," for failure to provide a procedure or instructions and acceptance criteria to perform an emergency makeup water alignment to the auxiliary feedwater system.

Description. Based on review of the Diablo Canyon Final Safety Analysis Report Update document (Revision 18), the design documentation on the auxiliary feedwater system credits eight sources of water that can provide backup means of supply to it in the event that its primary source of water, the condensate storage tank, becomes exhausted. This is particularly important in the event there is a complete loss of offsite and main generator electrical power to the station. These sources, stated in Final Safety Analysis Report Update Section 6.5.2.1.1, include the Diablo Canyon Creek. Review of operating procedures for instructions on aligning the supply source to the auxiliary feedwater system resulted in a reference in Procedure OP D-1:V, Revision 20, which states in Step 6.5.2, Sub-Step b.2: "WHEN required, THEN refill the Raw Water Reservoir from Diablo Creek using the Diesel-Driven Portable Long Term Cooling Water Pump (OVID 107031 sheet 26)."

This step reference refers the operations staff to a drawing as direction for performance (OVID 107031, Sheet 26). The drawing visually shows that a diesel-driven portable long-term cooling water pump is to be connected, taking suction from the Diablo Canyon Creek, discharging to the raw water storage reservoir pools, which provide makeup to the auxiliary feedwater system. However, there is no explicit procedure that describes precautions associated with the task, where the equipment needed for the task is located or can be obtained, or actions in a sequential order meant to ensure that positive flow is provided from the Diablo Canyon Creek to the raw water storage reservoir pools. This was in conflict with site procedural direction that says that operators shall operate plant components using written guidance (Procedure OP1.DC10, Section 5.2.3, Revision 22).

During the course of the inspection, the following activities were undertaken to assess the situation:

- A group comprised of non-licensed operators and a licensed operator conducted an observed walk-through with the referenced drawing to determine whether conditions and equipment at the location would allow for this lineup to be completed

- Feedback from the walk-through was provided to engineering and fire department staff to evaluate the available equipment for the lineup as well as history of prior testing of equipment in this lineup

Input from the engineering staff indicated that the equipment referenced in both the Final Safety Analysis Report Update document and design basis document design criteria Memorandum T-17 was never tested in the application identified. Information provided by the site fire department indicates that the site has the capability to align supply water from the Diablo Canyon Creek to the raw water storage reservoir pools, if necessary. However, there was no assurance that all of the equipment needed to prepare the line-up with the credited diesel-driven portable long term cooling water pump can successfully perform its role in transferring water between the two locations as annotated presently in the Final Safety Analysis Report Update document, design basis document design criteria Memorandum T-17, and Procedure OP D-1:V.

Analysis. The evaluation is based on implementation of Inspection Manual Chapter 0612, Appendix B. The issue is a performance deficiency because it was a failure to meet a requirement, and it was within the licensee's ability to correct this problem. The performance deficiency did not meet the screening criteria for traditional enforcement. Upon review versus Inspection Manual Chapter 0612, Appendix E, the performance deficiency does not meet the descriptions of minor or more-than-minor examples. 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) (the cornerstone objective).

Therefore, the performance deficiency is a finding. Using the Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 Worksheets, the finding is determined to have very low safety significance (Green) because it did not represent a design issue resulting in the 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. Review of the finding per Inspection Manual Chapter 0310 results in a crosscutting aspect being assigned in the area of problem identification and resolution, in that the licensee's 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. However, this effect was not identified as part of the changes, so no review of procedures related to emergency auxiliary feedwater system alignment in question took place. If this had occurred, the licensee may have been able to identify the issue that was the basis for this finding, and address it internally. Since these actions occurred within the last 3 years, this performance characteristic reflects present performance [P.1 (c)].

Enforcement. A violation of Diablo Canyon Technical Specification section 5.4.1. "Procedures," was identified. The licensee did not have a procedure to establish makeup from the Diablo Canyon Creek to the auxiliary feedwater system. 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 to it in the event that its primary source of water (the condensate storage tank) becomes exhausted. One of the sources includes 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 inspectors identified that the licensee did not have an established procedure to 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 was entered into the licensee's corrective action system as Notification 50298563. Because this finding is of very low safety significance (Green) and was entered into the corrective action program, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000323/2010007-08

ii. <u>Failure to Update Feedwater Rupture Accident Analysis in the Final Safety Analysis Report Update</u>

<u>Introduction</u>. The inspectors identified a Severity Level IV violation for failure to update the Final Safety Analysis Report Updated in accordance with 10 CFR 50.71(e). Specifically, the licensee failed to update Section 15.4.2.2.2, to reflect changes in the assumed operation of pressurizer safety valves.

<u>Description</u>. During the inspection period, documentation related to analysis of pressurizer safety valve operation during a feedwater line break accident was reviewed. In the 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, an update to the study (ESBU/WOG-98-154, July 31, 1998), 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 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 further 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. Currently, the Final Safety Analysis Report Update document is on Revision 18, and the reference to the Westinghouse study is still there. This leads the reader to believe that operator actions are credited with precluding consideration of challenges with pressurizer safety valve operation in the feedwater line break analysis, when there are no specific operator actions credited.

Per response provided in Notification 50301747, it has been acknowledged that the text needs to be revised to show the latest information on the feedwater line break analysis. Discussion with the staff that worked on the spurious safety injection analysis on March 3, 2010, indicates that they are knowledgeable of the differences in overpressure response characteristics between this analysis and the feedwater line break analysis to determine that the spurious safety injection analysis is bounding for the feedwater line break analysis. This analysis will be documented in support of the 10 CFR 50.59 evaluations, which will support the needed change in the Final Safety Analysis Report Update text.

Analysis. Using Inspection Manual Chapter 0612, Appendix B, the issue is a performance deficiency because it was a failure to meet a requirement, and it was within the licensee's ability to correct this problem. The performance deficiency is assessed through traditional enforcement because the finding impacted the NRC's regulatory process. Screening of the performance deficiency versus the Inspection Manual Chapter 0612, Appendix E, screening criteria results in a minor performance deficiency (see Item 3, Example I). Using Supplement I (Section D, Item 6) of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation.

Enforcement. Title 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." Contrary to this, since 2006 (Final Safety Analysis Report Update, Revision 16), the licensee did not update Final Safety Analysis Report Update, Section ,15.4.2.2.2. Due to the fact that this violation is of very low safety significance and was entered into the licensee's corrective action program as Notification 50301747, this violation is being treated as a Severity Level IV, noncited, violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000323/2010007-09, "Failure to Update Feedwater Rupture Accident Analysis in the Final Safety Analysis Report Update."

iii. 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.

Introduction. The inspectors identified a Severity Level IV violation for failure to update the Final Safety Analysis Updated report in accordance with 10 CFR 50.71(e). Specifically, the licensee failed to update Section 9.2.2.3.3, to reflect the licensee's Design Class I makeup flowpath to the component cooling water expansion tank.

Description. During the inspection, documentation related to credited makeup flowpaths to the component cooling water expansion tank was reviewed. In the Final Safety Analysis Report Update Table 9.2-7, Component 5, it says, "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 tanks (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 flowpath from the firewater tank was not Design Class I. Therefore, the issue became which lineup(s) are credited as Design Class I in pedigree, and how do the operator action time assumptions apply. Review by the licensee staff revealed that the only completely Design Class I flow path to provide makeup to the component cooling water expansion tank is via the condensate storage tank. Therefore, the time critical operator action of restoring makeup flow within 10 minutes applies to this flowpath. However, 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.

Analysis. Using Inspection Manual Chapter 0612, Appendix B, the issue is a performance deficiency because it was a failure to meet a requirement, and it was within the licensee's ability to correct this problem. The performance deficiency was assessed through Traditional Enforcement because the finding impacted the NRC's regulatory process. Screening of the performance deficiency versus the Inspection Manual Chapter 0612, Appendix E, screening criteria results in a minor performance deficiency (see Item 3, Example I). Using Supplement I (Section D, Item 6) of the NRC Enforcement Policy, this performance deficiency will be treated as a Severity Level IV violation.

Enforcement. Title 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 Update originally submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed." Contrary to this, 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. Due to the fact that this violation is of very low safety significance and was entered into the licensee's corrective action program as Notification 50301884, this violation is being treated as a Severity Level IV noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000323/2010007-010, "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."

4 OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

The inspectors reviewed actions requests associated with the selected components, operator actions and operating experience notifications. In addition, this report contains the following issue that has problem identification crosscutting aspects.

4OA6 Meetings

Exit Meeting Summary

On March 4, 2010, the inspectors' presented the preliminary inspection results to Mr. J. Becker, Site Vice President, and other members of the licensee's staff. On June 10, 2010, the inspectors conducted a telephonic final exit meeting with Mr. J. Becker and other members of the licensee's staff. The licensee acknowledged the findings during each meeting. While some proprietary information was reviewed during this inspection, no proprietary information was included in this report.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG 1600, for being dispositioned as a noncited violation.

- Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control", required Pacific Gas and Electric to establish a test program to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is performed in accordance with written test procedures, which incorporate the requirements, and acceptance limits contained in applicable design documents. Contrary to the above, in February 2010, Pacific Gas and Electric discovered that they had failed to incorporate into Procedure STP M-15, "Integrated Test of Engineered Safeguards and Diesel Generators," requirements and acceptance limits contained in Regulatory Guide 1.9, Revision 2, regarding voltage and frequency dips and recovery during design basis loading sequencing onto the emergency diesel generators. After self-identification of the issue, the licensee entered this issue in their corrective action program as Notification 50294602. This finding is of very low safety significance because the condition did not screen as potentially risk significant, in that it was a design deficiency that did not result in actual loss of safety function.

Attachments: Supplemental Information

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

J. Bailey, Supervisor, Engineering
S. Baker, Supervisor, Design Engineering
T. Baldwin, Manager, Regulatory Services
J. Becker, Site Vice President
E. Brackeen, Senior Engineer, Primary Systems
M. Coward, Operations Training Manager, Operations
T. Cutts, Nuclear Records Analyst, Site Services
M. Downum, Routine Plant Clerk, Fire/Safety/Medical
S. Dunlap, Supervisor, BOP Engineering
M. Fantz, Primary Systems Supervisor, Engineering
N. Gaudiuso. Supervisor DS, Site Services
B. Guldemond, Director, Site Services
M. Jackson, Engineer, Design Engineering, Mechanical
R. Klimczak, Engineer, Design Engineering
K. Kubran, Engineer, System Engineering
W. Landreth, Engineer, Design Engineering, Mechanical
A. Linn, Engineer, Design Engineering
R. Lovell, Senior Engineer, Engineering
M. MacIntyre, EM Manager, Maintenance
A. Maple, Engineer, Systems Engineering
M. McCoy, NRC Interface, Regulatory Services
K. Millenaw, Engineer, System Engineering
M. Munoz, Engineer, ICE Systems Engineering
M. Nowlen, Supervisor, Engineering
P. Nugent, Manager, Technical Support Engineering
L. Parker, Supervisor, Regulatory Services
K. Peters, Station Director
D. Peterson, Director, Quality Verification
G. Porter, Engineer, Systems Engineering
R. Prentice, Engineer, Design Engineering, Instrumentation and Controls
B. Rimers, Senior Engineer, Design Engineering
J. Rhodes, Engineer, Systems Engineering
S. Santiao, Engineer, Systems Engineering
L. Sharp, Senior Director, Engineering Services
M. Sharp, Engineer, Probalistic Risk Assessment
J. Shoulders, CDBI Team Engineering Lead, Engineering
M. Somerville, Manager, Radiation Protection
C. Sorensen, IST Coordinator, Engineering/ETI
M. Sullivan, Engineer, ICE Systems Engineering
B. Tripp, Engineer, Systems Engineering
L, Walter, Director, Training

R. Waltos, Supervisor, EFIN Engineering
M. Williamson, MOV Engineer, Engineering
I. Zakaria, Senior Engineer, ICE Engineering
S. Zawalick, NRC Interface and Commitment Management, Regulatory Services

NRC personnel

M. Peck, Senior Resident Inspector, Branch B, Division of Reactor Projects
T. Pruett, Deputy Director, Division of Reactor Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Open and Closed

05000323/2010007-01	NCV	Less than Adequate Change Evaluation to the Facility as Described in the Final Safety Analysis Report Update (Section 1R21.2.4.b.i)
05000323/2010007-02	NCV	Failure to Adequately Evaluate Changes to the Diesel Testing as Described in the Final Safety Analysis Report Update (Section 1R21.2.4.b.ii)
05000323/2010007-03	NCV	Non-Conservative Decision Making resulted in a Violation of Technical Specification (Section 1R21.2.9.b.i)
05000275; and 05000323/2010007-04	NCV	Failure to Update the Final Safety Analysis Report Update with the Current Plant Design Bases Section 1R21.2.9.b.ii)
05000323/2010007-05	NCV	Inadequate Operability Determination Associated With the Offsite Degraded Voltage Protection Scheme. (Section 1R21.9.b.iii)
05000323/2010007-06	NCV	Second Level Undervoltage Relay Time Delay to Initiate Load Shed and Sequencing Upon the Diesel Generator is Adequate to Assure Plant Safety. (Section 1R21.9.b.iv)
05000275/2010007-07	NCV	Nonconservative Inputs into Motor Operated Valve Calculation (Section 1R21.15.b)
05000323/2010007-08	NCV	Inadequate Drawings and Procedures to Align Emergency Makeup Water Supply from Diablo Canyon Creek to Support the Auxiliary Feedwater System (Section 1R21.4.b.i)

Open and Closed

05000323/2010007-09	NCV	Failure to Update Feedwater Rupture Accident Analysis in the Final Safety Analysis Report Update (Section 1R21.4.b.ii)
05000323/2010007-10	NCV	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 (Section 1R21.4.b.iii)

LIST OF DOCUMENTS REVIEWED

CORRECTIVE ACTION DOCUMENTS

50034917	50035102	50200201	50231639	50231656
50232181	50232184	50297173	50297692	50297854
50298148	50298241	50299307	50299379	50299689
50299776	50300249	50300608	50300666	50300677
50301331	50301839	50301888	50286743	50302073
50301793	50301341	50301963	50300809	50301068
50301792	50302050	50301840	50302439	50302349
50302437				

CALCULATIONS

NUMBER	TITLE	DATE/REVISION
234B DC	125Vdc Vital Battery 12 – Coordination Evaluation	Rev. 4
235B DC	Battery 12 Sizing, Voltage Drop, Short Circuit and Charger Sizing Calculation	Rev. 10
9000037760	Diesel Generator Loading for 4160V Vital Buses	Rev. 20
9000006964	Diesel Generators	Rev. 15
ES 15 1	Seismic Qualification of C&D Model LCUN 33 Batteries and Two Step Battery Rack	Rev. 0
V 07, Appendix B	Various Motor Operated Valves	Rev. 8
195G DC	Turbine Steam Supply – DC Operated Motor Operated Valve FCV 95	Rev. 5
PG&E Calculation No. 9000040736 (STA-274)	“RETRAN Evaluation of GDC 17 Loss of AC Scenario”	2008/Rev. 0

CALCULATIONS

NUMBER	TITLE	DATE/REVISION
Westinghouse WCAP-8822	"Mass and Energy Releases Following a Steam Line Rupture"	September 1976
Westinghouse WCAP-11677	"Pressurizer Safety Relief Valve Operation for Water Discharge During a Feedwater Line Break"	January 1988
PG&E Calculation STA-248	"RETRAN Spurious Safety Injection Analysis for DCPP with RSG"	Rev. 0
Westinghouse WCAP-13908	"Analysis of Containment Response Following Main Steam line Break Accidents for Diablo Canyon Units 1 and 2"	December 1993
SC-I-9-L921	Refueling Water Storage Tank 1-1 Level Channel LT-921	February 2009
SC-I-9-L922	Refueling Water Storage Tank 1-1 Level Channel LT-922	Rev. 6
SC-I-9-L920	Refueling Water Storage Tank 1-1 Level Channel LT-920	Rev. 7
9000028686-005 (J-142A)	RWST Nominal Setpoint and Uncertainty Calculations	Rev. 5
N-178	RWST Initial Temperature Assumptions in Safety Related Analyses	Rev. 0
J-080	RHR HX Outlet Flow Indication Uncertainty	Rev. 6
M-179	Check on NPSH Calculation for MDAFW Pumps 1-2 and 1-3 Assuming a Simultaneous Failure of SGs 1-2 and 1-3	Rev. 1
PAM-0-3-171	Auxiliary Feedwater System Flow Indicator Uncertainty	Rev. 3
SC-I-3-F50	Auxiliary Feedwater Flow Channels FT-50, FT-77, FT-78, and FT-79	Rev. 4
N-100	Maximum Flow from ECCS Pumps and Minimum Flow to Containment Spray Header (900006525)	Rev. 4
Report No. R91.124 DC663219-764-1	Maximum & Required Thrust Analysis 14"-Class 300 D.D. Gate Valve with Limitorque SMB Motor Operator	July 1991
Report No. R91.125 DC663219-761-1	Maximum and Required Thrust Analysis 8"-Class 300 D.D. Gate Valve with Limitorque SMB Motor Operator	July 1991

CALCULATIONS

NUMBER	TITLE	DATE/REVISION
9000006092 (N-013)	Motor Operated Valve Limiting Process Conditions Evaluation	Rev. 20
M-786	Determine the Required Diesel Fuel Oil Storage needed to meet DCPP Licensing Basis for Operating "Minimum ESF Loads"	Rev. 16
9000035426 (V-07)	Rising-Stem MOV Actuator Sizing and Setpoint Calculation	Rev. 9
J-076	Validate Various Excel Spreadsheets for use in performing ICE12 Evaluations	Rev. 8
N-060	ECCS STP V-15 Flow Limits	Rev. 1
N-064	RHR Valve Replacement	Rev. 1
STA-135	AFW Pump Acceptance Curves	Rev. 2
92-09	480 V SWGR Fans Upgrade – Duct Pressure Loss Calculation	Rev. 0
HVAC-92-16	480V Switchgear & 125V DC Inverter Rooms Airflow	Rev. 1

DESIGN BASIS DOCUMENTS

NUMBER	TITLE	DATE/REVISION
FSARU Section 8.3	Onsite Power Systems	Rev. 18
TS 3.8.1	A. C. Sources – Operating	Rev. 7
TS Bases B 3.8.1	A. C. Sources – Operating	Rev.5
DCM T-17	"Long Term Cooling Water"	Rev. 4
DCM S-3B	Auxiliary Feedwater System	Rev. 19A
DCM S-9	Safety Injection System	Rev. 29
DCM S-9A	Containment Recirculation Sump & Strainer Function	Rev. 7A
DCM S-14	Component Cooling Water System	Rev. 20

DESIGN BASIS DOCUMENTS

NUMBER	TITLE	DATE/REVISION
DCM S-8	Chemical and Volume Control System	Rev. 33
DCM S-21	Diesel Engine System	Rev. 22
DCM S-23C	Miscellaneous auxiliary Building HVAC Systems	Rev. 12C
DCM T-17	Long Term Cooling Water	Rev. 4
B-STP P-3A	Bases Document for STP P-3A Performance Test of Residual Heat Removal Pumps	Rev. 1
B-STP P-5A	Bases Document for STP P-5A Performance Test of MDAFW Feed Pumps	Rev. 0
B-STP P-AFWM	Bases Document for STP P-AFW-12, -13, Routine Surveillance Tests of MDAFW Pumps	Rev. 1

DESIGN CHANGE PACKAGE

NUMBER	TITLE	DATE/REVISION
DCP No.H-47456	HVAC-480V Switchgear/125V Inverter RMS Vent System	Rev. 1
DCP No. N-44172	Add Check Valves to each RHR Train	Rev. 1
DCP No. J-47277		Rev. 3

DRAWINGS

NUMBER	TITLE	DATE/REVISION
496146 Sh. 1	Battery Rack 1-2 Area H, Elevation 115'-0" Auxiliary Building	Rev. 1
496148 Sh. 1	Battery Rack 1-2 Area H, Elevation 115'-0" Auxiliary Building	Rev. 3
4005620 Sh. 1	Arrangement and Details for Battery No. 12 Replacement	Rev. 2
4005623 Sh. 1	Vital Batteries Cell Layout Area "H" Elev. 115	Rev. 6
663343 Sh. 4	Rack One Tier with E.P. for 15-LCU-27	Rev, 4
445075 Sh. 1	Single Line Meter and Relay diagram 125Volt DC System	Rev. 17
OVID 107031, Sh 26	Drawing Showing Lineup to Raw Water Reservoir from Diablo Creek using the Diesel-Driven Portable Long Term Cooling Water Pump (No Official Title)	Change 5

DRAWINGS

NUMBER	TITLE	DATE/REVISION
59346	Ventilation, Plan-Area "H"-Elevation 100' 0"	Rev. 32
59347	Ventilation, Plan-Area "H"-Elevation 115' 0"	Rev. 48
59349	Ventilation, Plan-Area "H"-Elevation 163' 6"	Rev. 32
102023 (Sht. 13)	Ventilation & Air Conditioning Systems	Rev.102
102023 (Sht. 13A)	Ventilation & Air Conditioning Systems	Rev.11
102023 (Sht. 14)	Ventilation & Air Conditioning Systems	Rev.90
102003	Piping Schematic Feedwater System	Rev. 81
102008	Piping Schematic Chemical & Volume Control System	Rev. 108
102009	Safety Injection System	Rev. 78
94-13295	8" No. S70 W DD Weld Ends Outside Screw & Yoke Double Disc Gate Valve with SMB-1 Limitorque Valve Control Lip Seal and Limit Switches	Rev. F November 1975
663219	ASA Series 300 14" No. S70DD Gate Valve	Rev. 12
508845	Diesel Fuel Oil Transfer Pump Vaults	Rev. 13
102021	Piping Schematic Diesel Engine-Generator Systems	Rev. 72
S-6707-11-13	Flow Nozzle Weld-In Type Throat Tap DCI -26-214, -215, -216	December 1993
438165	Emergency Diesel Generator Fuel Oil System	Rev. 14
108021	Piping Schematic Diesel Engine-Generator Systems	Rev. 51
102010	Piping Schematic Residual Heat Removal System	Rev. 46
438039	Requirements for Water Storage Tanks	Rev. 5
438038	Requirements for Water Storage Tanks	Rev. 8
464831	Vortex Suppression Cages Outdoor Water Storage Tanks	Rev. 3
DC-663217-21-2	Characteristic Curve, Pump No. 037050, (RHR Pump)	June 1971

DRAWINGS

NUMBER	TITLE	DATE/REVISION
6006473 (NC-291245-02)	W/Pneumatic Actuator, Positioner, Regulator & Volume Booster	Rev. 4

ENGINEERING REPORTS

NUMBER	TITLE	DATE/REVISION
Test Procedure 3.11	"Long Term Cooling Water Supply From Auxiliary Saltwater"	Rev. 1
Test Procedure 3.10	"Long Term Cooling Water Supply from Raw Water Reservoir"	Rev. 2

WORK ORDERS

64035163	R0239570-01	A0571777
64035723	64031185	64028049
R0312775-01	R0312984-01	64007927
A0739575	R0300830	50094526
50197704	A0690169	A642776
C0198517	A0739575	R0300830-01

MODIFICATIONS/ENGINEERING CHANGE PACKAGES

NUMBER	TITLE	DATE/REVISION
DCP 3362	"Station Battery Replacement,"	Rev. 2

OPERATOR ACTION SUMMARY DATA SHEETS

NUMBER	TITLE	DATE/REVISION
DCL-92-087	Diablo Canyon Power Plant Units 1 and 2 Individual Plant Examination Report"	April 14, 1992

PROCEDURES

NUMBER	TITLE	DATE/REVISION
EOP E-0	Reactor Trip or Safety Injection	Rev. 37
EOP E-2	Faulted Steam Generator Isolation	Rev. 17
EOP ECA-0.3	Restore 4kV Buses	Rev. 14

PROCEDURES

NUMBER	TITLE	DATE/REVISION
EV2.DC4	Hazardous Materials Management Program	Rev. 6
MP E-64.1B	Molded Case Circuit Breaker Exercise and Maintenance	Rev. 9
MP E-64.6A	Maintenance of ABB K-Line Circuit Breaker	Rev. 35
MP E-67.6	Station Battery Preventative Maintenance	Rev. 9A
OM7.ID1	Problem Identification and Resolution	Rev. 32
OM8.ID4	Control of Flammable and Combustible Materials	Rev. 17
OP AP-6	Emergency Boration	Rev. 19
OP D-1:V	Auxiliary Feedwater System – Alternate Auxiliary Feedwater Supplies	Rev. 20
OP F-2:VII	Alternate Makeup Water Sources to the CCW System	Rev. 5
OP J-2:V	Back feeding the Unit from the 500 kV System	Rev. 12
OP K-2A:III	Alternate Methods of Pressurizing and Filling the Firewater System	Rev. 10
OP O-27	Coordination of 500/230kV Activities	Rev. 5
OP O-39	Reservoir Inventory Management	Rev. 0
OP1.DC10	Conduct of Operations	Rev. 22
STP M-11A	Station Battery and Pilot Cell Condition Monitoring	Rev. 21
STP M-11B	Station battery Condition Monitoring	Rev. 26
STP M-11C	Terminal Resistance Measurements and Inspection For Vital Station Batteries	Rev. 16
STP M-11D	Station Battery Terminal Voltage and Float Current Monitoring	Rev. 2
STP M-12A	Vital Station Battery Modified Performance Test	Rev. 15
STP M-9D1	Diesel Generator Full Load Rejection Test	Rev. 16
STP M9D2	Diesel Generator Partial Load Rejection Test	Rev. 17

PROCEDURES

NUMBER	TITLE	DATE/REVISION
XI1.ID2	Regulatory Reporting Requirements and Reporting Process	Rev. 28
XI3.ID6	Technical Specification bases Control Program	Rev. 2
OP J-6C:I	Diesel Fuel Oil Transfer System – Make Available and Place In Service	Rev. 12
STP V-2D2	Exercising and Position Verification of Valves 8700A and 8700B	Rev. 10
ENGNTS13	Motor Operated Valve Diagnostic Test Trace Analysis, (Task Qualification) OT/TPE Guide	Rev. 0
MP E-53.10S	Limitorque Swap-Out and Switch Settings	Rev. 10
STP V-3M4A	Exercising RHR Pump 1 Suction Valve 8700A	Rev. 6
OM7.ID1	Problem Identification and Resolution	Rev. 32
MAI.ID1	Motor-Operated Valve (MOV) Program Plan	Rev. 9
MA1.ID6	Check Valve Maintenance, Testing and Inspection Program	Rev. 4
MA2.ID2	Performance Monitoring Equipment Calibration and Usage Control	Rev. 0
Report	Software Quality Assurance Plan, CRANE Nuclear Powerhouse Software, Version 2.0, (MOV Program Documentation)	02/04/08
STP V-2V1	Exercising and Position Verification of Valves 8804A and 8894B	Rev.10A
ICE-12	I&C Engineering Procedure for Preparation of MOV Sizing and Switch Setpoint Calculations	Rev. 12
STP R-20	Boric Acid Inventory	Rev. 34
STP P-AFW-12	Routine Surveillance Test of MDAFW Pump 1-2	Rev. 16
OP H-10:11	Auxiliary Building General Ventilation – Normal Operation	Rev. 7

VENDOR MANUALS

NUMBER	TITLE	DATE/REVISION
Vendor Manual DC-663219-505	Installation, Operation, and Maintenance of Darling OS&Y Motor Operated Gate Valves	Rev. 9
C&D Technical Manual	Standby Battery, Flooded Cell, Installation and Operating Instructions (LCUN-33)	

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE/REVISION
	Battery 12 Trend Data Voltage, Specific Gravity, Temperature, Level	April 2007 to January 2010
	Battery 12, Cell #8 Voltage	April 2007 to January 2010
Material Safety Data Sheet; E.K. Industries, Inc.	Sodium Bicarbonate	
	Material Safety Data Sheet; C&D Technologies, Inc. Sulfuric Acid	
DCPP Green Label S/C # 95-1405	Battery Electrolyte	
DCPP Green Label S/C # 74-5397	Sodium Bicarbonate Powder	
	C&D Letter Battery Seismic Qualification for DCPP	February 1983
	DC Response Letter for GL 2007-01	May 2007
	NRC Closeout Letter for GL 2007-01	October 2008
	Letter from Ken J. Vavrek, Project Engineer, Westinghouse Electric Corporation, to Westinghouse Owners Group, "Westinghouse Owners Group Notification for Pressurizer Safety Valve Operability Issue (MUHP-8098)"	July 1998
DCL-92-084	Letter from Gregory M. Rueger, PG&E, to US NRC, Docket No. 50-275, OL-DPR-80;	April 1992
	Docket No. 50-323, OL-DPR-82; Diablo Canyon Units 1 and 2, Revised Response to Station Blackout"	

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE/REVISION
DCL-02-115	Letter from Gregory M. Rueger, PG&E, to US NRC, Docket No. 50-275, OL-DPR-80; Docket No. 50-323, OL-DPR-82; Diablo Canyon Units 1 and 2, License Amendment Request (LAR) 01-08 Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves; Pressurizer Safety Valve Loop Seal Temperature"	September 2002
	Letter from Girija S. Shukla (NRR), US NRC, to Gregory M. Rueger, PG&E, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 – Issuance of Amendment RE: Credit for Automatic Actuation of Pressurizer Power Operated Relief Valves (TAC Nos. MB6758 and MB6759)"	July 2004
SSER 7	"Supplement No. 7 to the Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, in the Matter of Pacific Gas and Electric Company, Diablo Canyon Nuclear Power Stations, Units 1 and 2, Docket Nos. 50-275 and 50-323"	May 1978
SSER 8	"Supplement No. 8 to the Safety Evaluation Report by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, in the Matter of Pacific Gas and Electric Company, Diablo Canyon Nuclear Power Stations, Units 1 and 2, Docket Nos. 50-275 and 50-323"	November 1978
A0257289	"Corrective Actions for NCR DC0-92-TN-N004"	September 1994
A0200015	"Please Modify Design of 'Long Term Cooling System"	March 1992
RLOC 10661-2979	Letter from E. C. Connell to B. H. Patton, "RE: Long Term Cooling Water System SAN 73"	May 1982
Notification 50298563	Creek Not a Viable Source for LTCW	February 2010
Notification 50263235	TCOA training for IOTCs	August 2009
Notification 50301747	CDBI update FSAR 15.4.2	March 2010
Notification 50301884	2010 CDBI: MWTP supply configuration	March 2010

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE/REVISION
	White Paper from Diablo Canyon Fire Department – Ability to Pump Water from Diablo Canyon Creek to Raw Water Storage Reservoirs in an Emergency	March 2010
PGE Letter Regarding GL 89-10 Program	Subject: NRC INSPECTION Report 50-275/95-01	March 1995
DCL-89-127	PG&E Letter, (Response to Bulletin 88-04)	May 1989
PGE-89-785	PG&E Letter, Minimum Flow Evaluation for Motor-Driven Auxiliary Feedwater Pump	October 1989
PGE-01-524	PG&E Letter, Motor Driven Auxiliary Feedwater Pump Minimum Mechanical Flow	June 2001
DCL-92-036	PG&E Letter	February 1992
DCL-05-034	PG&E Letter	April 1995
DCL-92-131	PG&E Letter	June 1992
86B0215 (13-8)	PGE Letter, SER 20-03 & Supplemented: VELAN	December 1986
	RHR System Health Report	1st quarter 2010
DCL-08-012	PG&E Letter (Response to Request for Additional Information on License Amendment Request 07-02, "Revision to Technical Specification (TS) 3.5.4, 'Refueling Water Storage Tank (RWST)'"	February, 2008
DCL-08-022	PG&E Letter, "Response to Request for Additional Information on License Amendment Request 07-02, "Revision to Technical Specification (TS) 3.5.4, 'Refueling Water Storage Tank (RWST)'"	March 2008
DCL-07-093	PG&E Letter, License Amendment Request 07-02, Revision to Technical Specification (TS) 3.5.4, "Refueling Water Storage Tank (RWST)"	October 2007
Specification Number 8844	Specification for Design and Construction of Outdoor Storage Tanks	Rev. 6
B16.5	300 Cl Flange Ratings	1968

MISCELLANEOUS DOCUMENTS

NUMBER	TITLE	DATE/REVISION
Limitorque Technical Update 98-01	Actuator Output Torque Calculation	98-01
H10	Miscellaneous Building Ventilation System	Rev. 13