# Diablo Canyon 1

## Initiating Events

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

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

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

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

---

# Mitigating Systems

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

**Significance:** G May 31, 2002
Identified By: NRC
Item Type: NCV NonCited Violation

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

**Significance:** G Apr 05, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Licensee Restarted Unit 2 Before Recognizing Reactor Trip and Engineered Safety Features Actuation Associated with Lo-lo Steam Generator Water Level was Inoperable**
The failure to promptly identify and correct the steam generator narrow range water level-low low reactor trip system and engineered safety system instrumentation nonconservative setpoint bias following the Unit 2 manual reactor trip on February 9, 2002, is a violation of 10 CFR Part 50, Criterion XVI. The licensee's event review failed to recognize that an engineered safety feature, including a reactor trip, failed to actuate when required during a loss of feedwater event to Steam Generator 2-4. This failure resulted in the licensee restarting Unit 2 with the reactor trip and engineered safety system instrumentation inoperable, and in the operation of both units with the same instrumentation inoperable, in violation of Technical Specification 3.3.1. This issue is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy (50-275; 323/2002-07-01). The licensee documented this deficiency in Action Request A0549031. The failure to promptly recognize inoperable trip and actuation functions and comply with Technical Specification requirements had a credible impact on safety. The resulting delays in an automatic reactor trip and engineered safety features actuations would have delayed the plant's response to a loss of feedwater event and reduced the water mass available for the heat sink function in the affected steam generator(s). Further, this deficiency had the potential to affect the integrity of the reactor coolant system boundary. A Phase 3 Significance Determination Process evaluation concluded that the issue had very low safety significance (Green). The finding represents a condition that existed for 5-days. The significance of the steam generator narrow range water level-low low setpoint offset (bias) is reduced if feedwater flow is lost to two or more steam generators. Based on the short duration from the time a single steam generator would dryout (the limiting initiator is a loss of feedwater to a single generator) and actuation of auxiliary feedwater, the condition does not result in an appreciable increase in the probability of a steam generator tube rupture occurring. The licensee's analysis using the plant specific simulator showed that the engineered safety feature actuation and reactor trip on steam generator water level-low low would have initiated at or before steam generator dryout would occur. The reactor coolant system physical over pressure protective features (safety relief and power operated relief valves) should not be challenged and there were other protective trips in place (over temperature-delta temperature and over pressure delta-temperature) in place that would have protected the reactor coolant system and fuel integrity in the event a manual reactor trip is not initiated on a loss of feedwater flow to a steam generator [Sections 4OA2.a.(2) and 5].
Inspection Report# : 2002007*(pdf)*

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

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

**Significance:** G Jul 06, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Perform Adequate Postmaintenance Test on Auxiliary Saltwater Pump**
The inspectors identified a violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the failure to perform an adequate post-maintenance test on Auxiliary Saltwater Pump 1-2 prior to placing the pump in service. The licensee installed new packing on Auxiliary Saltwater Pump 1-2 as part of the pump replacement that occurred between May 9 -16, 2002. The licensee performed a post-maintenance test on Auxiliary Saltwater Pump 1-2 on May 17 and documented there was adequate packing leak-off flow. Then on May 30 operators started Auxiliary Saltwater Pump 1-2

but identified no leak-off flow. The post maintenance test was not adequate to identify that the packing had been improperly installed and that the packing had shifted and swelled following the May 17 pump run. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This item was placed in the corrective action system as Action Request A0560036. This violation was more than minor because if the same condition, under similar circumstances, were present for a longer period of time, the finding would be of greater safety significance. An NRC senior reactor analyst performed a significance determination process Phase 3 safety assessment. The senior reactor analyst reviewed the licensee's risk assessment, and the safety significance insights obtained from the NRC's Standardized Plant Analysis Risk (SPAR) model for Diablo Canyon Units 1 and 2 (Revision 3i) as well as NRC Manual Chapter 0609, Significance Determination Process, Appendices A and G, Significance Determination of Reactor Inspection Findings for At-Power Situations and Shutdown Safety SDP [significance determination process] for those plant conditions utilizing residual heat removal, respectively. The senior reactor analyst considered, in part, the plant conditions, availability of the steam generators as a heat sink and the low decay heat for each of the plant modes during which the condition existed, and the availability of the Auxiliary Saltwater Unit 2 crosstie in assessing the overall safety significance. It was also noted that the temperature at the packing gland was elevated following the pump run on May 30 but did not indicate early pump failure was likely. Based on the quantitative and qualitative assessment for this condition, the senior reactor analyst concluded the condition was of very low safety significance (Section 1R19).
Inspection Report# : 2002003*(pdf)*

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

**Significance:** G May 19, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**Technical Specification 3.0.3 violation for rendering all three emergency power sources for Unit 2 Vital Bus H inoperable**
A violation of Technical Specification 3.0.3 and 3.8.1.1 occurred because operators rendered two sources of offsite

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

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

**Significance:** Jan 12, 2001
Identified By: NRC
Item Type: NCV NonCited Violation
**CCW system outside of design basis because valves would not close properly**
The Unit 1 component cooling water system was operated in a condition outside its design basis because of excessive leakage through the component cooling water crosstie valves. This resulted in a condition where the component cooling

water system could not be completely separated into its two trains to mitigate a system leak or protect against a single failure was compromised. The licensee had not translated adequate design controls for component cooling water system train isolation into procedures or instructions to ensure the ability to isolate a single train and prevent a single failure from rendering the component cooling water system inoperable. The failure to translate this design basis information into instructions or procedures is a violation of 10 CFR 50, Appendix B, Criterion III (Design Control). This violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. This issue was entered into the licensee's corrective action program as Nonconformance Report N0002117. The inspectors determined that this issue was of very low risk significance. The inspectors noted that the licensee's analysis assumed that a safety-related 250 gpm makeup source was available to replenish the component cooling water system. In addition, two other nonsafety-related makeup sources were available. The inspectors noted that although the ability to split the trains was compromised, the component cooling water system could have met its intended safety function despite the condition with adequate normal and backup makeup systems available.
Inspection Report# : 2000016*(pdf)*

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

**Significance:** G Aug 09, 2000
Identified By: Self Disclosing
Item Type: NCV NonCited Violation
**Work on wrong equipment resulted in failure to follow procedures (Section 1R13.2)**
Personnel failed to follow maintenance procedures on two occasions in working on the wrong component or wrong unit. These errors resulted in the control room ventilation system and the main annunciator systems being inadvertently unavailable for time periods less than the Technical Specification allowed outage times. These errors were two

examples of a violation of Technical Specification 5.4.1.a. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. Several similar occurrences were noted in which personnel performed work on the wrong trains or wrong unit, indicating that a continuing adverse trend existed with respect to human performance. These errors were placed in the corrective action program as Action Requests A0512713 and A0512203. The inspectors assessed the risk significance of these errors using the significance determination process. The inspectors determined that these issues were of very low risk significance, and thus constituted a green finding. The inspectors used the significance determination process Phase 1 screening worksheet and noted that the control room ventilation was considered a support system for the unavailability of the solid state protection system. However, only one train of the control room ventilation system was inadvertently inoperable for a time period less than the Technical Specification limiting condition for operation. The main annunciator system was inoperable for only a short time and the system is designed with redundant annunciation that was available. Thus, these items screened to green
Inspection Report# : 2000010*(pdf)*

**Significance:** G Jun 29, 2000
Identified By: NRC
Item Type: FIN Finding
**Assumed failure to torque bus bar splice following transformer explosion in 1995**
The inspectors found that this event likely resulted from a combination of design deficiencies and the potential failure to properly torque the failed joint following the 1995 Unit Auxiliary Transformer 1-1 explosion. In order to determine the increase in risk associated with this event, the inspectors assumed the bus bar failure resulted from inadequate torquing of the joint (inadequate corrective actions) following the 1995 transformer explosion. Using the significance determination process, the Senior Reactor Analyst found that this performance issue of inadequate corrective actions had very low risk significance. The Phase 2 risk assessment determined that there was no appreciable change in the core damage frequency. However, the worksheets for Transient and Loss of Offsite Power did not fully account for the loss of the nonvital bus bars from the unit auxiliary and startup transformers. Consequently, the Senior Reactor Analyst conducted a Phase 3 evaluation, using site specific probabilistic risk assessment information related to the mitigating capability of and initiating frequency contribution to the 12 kV nonvital switchgear. From evaluation of the accident sequences that this event impacted, the Senior Reactor Analyst again concluded that minimal change in the core damage frequency occurred. The minimal changes in core damage frequency resulted because the licensee had already modeled that this event had a high likelihood of occurring. This event did not disable any vital equipment and all vital equipment operated as designed (Section 4OA3.3).
Inspection Report# : 2000009*(pdf)*

**Significance:** G May 15, 2000
Identified By: Licensee
Item Type: NCV NonCited Violation
**Operator prematurely energized accumulator isolation valves, violating TSs**
During the reactor coolant system cool down from normal operating pressure and temperature to Mode 4 (Hot Shutdown), operators energized the safety injection accumulator discharge isolation valves with reactor coolant system pressure at 1500 psig. With the valves, energized operators could have inadvertently isolated the safety injection accumulators. Approximately 3 hours later, with reactor coolant system pressure at 1122 psig, the operators recognized that Technical Specification 3.5.1 prohibited these valves from being energized and entered Technical Specification 3.0.3. Reenergizing the safety injection accumulator discharge isolation valves violated Technical Specification 3.5.1. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The inspectors assessed the risk significance of this finding using the significance determination process and determined that this event had very low risk significance. The inspectors used the Phase I worksheet and determined that this event affected a mitigating system. The inspectors assumed that the operators had closed the safety injection

accumulator isolation valves and would not be able to recover the components. With this assumption, the Phase 1 screening required a Phase 2 review because of the inability to inject water using the accumulators. From the Phase 2 review, the inspectors identified one sequence affected in the large break loss of coolant accident and one sequence affected in the small break loss of coolant accident. These evaluations resulted in very low risk significance based on the low initiating event frequency for loss of coolant events and availability of other mitigating equipment. The licensee included this item in their corrective action program as Action Request A0508060
Inspection Report# : 2000009*(pdf)*

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

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

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

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

---

# Barrier Integrity

---

# Emergency Preparedness

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

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

---

# Occupational Radiation Safety

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

**Significance:** Jul 06, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Barricade a High Radiation Area**
The inspectors identified a violation of Technical Specification 5.7.1.a because the entrance to a high radiation area boundary surrounding the reactor vessel head on the 140-foot elevation of the containment building was not barricaded. General radiation levels in the area were as high as 120 millirem per hour. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0555046. The issue was more than minor because the failure to barricade a high radiation area has a credible impact on safety and the occurrence had the potential to involve a worker's unplanned dose if radiation levels had been significantly greater. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised (Section 2OS1).
Inspection Report# : 2002003*(pdf)*

**Significance:** Jul 06, 2002
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Post a Radiation Area**
The inspectors identified a violation of 10 CFR 20.1902, based on the area outside the drum compactor room on the 115-foot elevation of the auxiliary building was not posted as a radiation area. On May 6, 2002, the licensee performed a survey of the area which identified that general radiation levels were as high as 8 millirem per hour. However, on May 7, 2002, the inspectors found that the area was not posted as a radiation area. The failure to post a radiation area is a 10 CFR 20.1902 violation. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as Action Request A0554991. The issue was more than minor because the failure to post a radiation area has a credible impact on safety and the occurrence had the potential to involve a worker's unplanned dose if radiation levels had been significantly greater. The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because it was not an ALARA finding, there was no overexposure or substantial potential for an overexposure and the ability to assess dose was not compromised (Section 2OS1).
Inspection Report# : 2002003*(pdf)*

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

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

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

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

**Significance: N/A** Oct 20, 2000
Identified By: NRC
Item Type: NCV NonCited Violation
**Failure to Survey**
No color. On October 17, 2000, the inspector identified that radiation protection personnel failed to perform a contamination survey of the upper internal lifting rig platform prior to a worker entering the area. 10 CFR Part 20, Section 1501(a), states, in part, each licensee shall make or cause to be made, surveys that are reasonable under the circumstances to evaluate concentrations or quantities of radioactive material, and the potential radiological hazards. The failure to perform a contamination survey of the above area was a violation of 10 CFR 20.1501(a). This violation is being treated as a noncited violation and is in the licensee's corrective action program as Action Request A0517252. Although the significance of this violation was determined to be more than minor, because there was a credible impact on a worker's radiation safety, it did not affect the cornerstone (Section 2OS1).
Inspection Report# : 2000014*(pdf)*

---

# Public Radiation Safety

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

**Significance:** Sep 20, 2000
Identified By: NRC
Item Type: FIN Finding
**Licensee failed to follow waste disposal facility site criteria requirement.**
On December 8, 1999, the Chem-Nuclear Systems radioactive waste disposal facility accepted radioactive waste Shipment RWS-99-004 without comment and buried the radioactive waste in a near-surface burial area. The licensee had shipped the Class C waste to the Chem-Nuclear Systems radioactive waste disposal facility in accordance with 10

CFR 61.55, Table 1. On April 21, 2000, a licensee audit identified a calculation error associated with the waste classification of Shipment RWS-99-004. This error resulted in the shipment not meeting Chem-Nuclear System's acceptance criteria. However, there was no violation of NRC requirements. Although not a violation of NRC requirements, the failure to meet Chem-Nuclear System's acceptance criteria in this instance was characterized as a "green" finding. Based on the public radiation safety significance determination process, the issue had very low safety significance because the Carbon-14 concentration in the radioactive waste did not exceed the value in 10 CFR 61.55, Table 1, when calculated in accordance with 10 CFR 61.55 (a)(8). This finding is in the licensee's corrective action program as Action Requests A0506728 and A0508956.
Inspection Report# : 2000012*(pdf)*

---

## Physical Protection

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

---

## Miscellaneous

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

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

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

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

Last modified : August 29, 2002