ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Report No.:	50-275/97-10 50-323/97-10
Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Nuclear Power Plant, Units 1 and 2
Location:	7 1/2 miles NW of Avila Beach Avila Beach, California
Dates:	June 8 through July 19, 1997
Inspectors:	M. Tschiltz, Senior Resident Inspector D. Allen, Resident Inspector B. Olson, Project inspector W. Ang, Reactor Inspector
Approved By:	H. Wong, Chief, Reactor Projects Branch E
ATTACHMENT:	Supplemental Information

EXECUTIVE SUMMARY

Diablo Canyon Nuclear Power Plant, Units 1 and 2
NRC Inspection Report 50-275/97-10; 50-323/97-10

Operations

- Control room operator's actions in response to the condensate/feedwater transient and subsequent Unit 2 trip on July 2 were timely and appeared to mitigate the effect of the loss of both main feedwater pumps (MFPs) on other plant parameters. The evaluation of the event and plant response was thorough. Subsequent corrective actions taken to improve the unit's ability to withstand similar transients were based upon a thorough investigation of the event. The subsequent power ascension was cautious and involved several verifications of condensate and feedwater system operation prior to returning to 100 percent power (Section O1.2).

- A violation was identified when inspectors noted that the licensee's procedures for alignment of a vital 480V power source to an instrument panel had not been followed during the restoration from maintenance that deenergized the normal power supply to the panel. Walkdowns of the power supply alignment verified that normal power had been properly aligned; however, a sealed component checklist had not been completed for the transfer switch and the backup power supply breaker was not positioned in accordance with procedural requirements (Section O1.4).

Maintenance

- Postmodification testing for replacement of engineered safety feature timers for Containment Fan Cooler Unit (CFCU) 2-3 was thorough and demonstrated the acceptable performance of the design change. The test participants performed the test cautiously, with good coordination with operations (Section M1.1).

- Good test coordination occurred between reactor engineering and operations during performance of Nuclear Power Range Incore/Excore Multiple-point Calibration surveillance. Additionally, the control operator made good use of plant process computer to monitor plant parameters and control core reactivity to obtain accurate data in support of the test (Section M1.2).

- A violation was identified involving a containment isolation valve in an instrument line that was not included in the monthly surveillance and had not been verified closed at least once per 31 days (Section M1.3).

- A noncited violation was identified when the licensee noted that Mode 3 had been entered with an inoperable centrifugal charging pump (CCP). CCP 1-1 was determined to have been inoperable, due to total pump flow in excess of that allowed by Technical Specifications (TSs), as a result of erosion of the miniflow restriction orifice (Section M8.1).

- Review of the licensee's control of RTV-732 caulking material revealed that the licensee allowed its use in containment without administrative controls. Tests of the RTV-732 sealant conducted by the licensee demonstrated that the sealant was not qualified for use in a harsh environment (Section M8.2).

- Temporary modification/Jumper 97-017, which installed instrumentation and recorders to assist in troubleshooting the cause of the July 2 trip, was well planned and properly implemented. However, the inspectors noted several errors in the documentation of the jumper indicating a lack of attention to detail (Section O1.3).

Engineering

- The design change for the replacement of the CFCU engineered safety feature timers effectively coordinated the activities from initial design planning to final implementation. Cross-discipline engineering reviews, probabilistic risk analysis assessment and a licensing basis impact evaluation (LBIE) evaluation were appropriately performed. Affected procedures and drawings were identified and revised (Section E1.2).

- In the control room several drawings and schematics issued by engineering were noted to have portions that were illegible. After identifying this situation to the licensee, an extensive review was performed which identified numerous drawings and schematics that also had portions that were illegible. Based upon the number of discrepancies noted, the inspectors concluded that the licensee's program for maintaining drawings and schematics in the control room failed to ensure that the drawings were legible (Section O1.1).

Plant Support

- During the performance of the reactor coolant sample, the chemistry technician was noted to be knowledgeable of primary sampling procedures, equipment use, and radiological controls. Primary samples were obtained satisfactorily (Section R4.1).

Report Details

Summary of Plant Status

Unit 1 began this inspection period at 100 percent power. The unit remained at 100 percent power throughout the inspection period.

Unit 2 began this inspection period at 100 percent power. On July 2, the unit was manually tripped by control room operators when a loss of suction pressure to the MFPs resulted in the pumps tripping on overspeed. The unit was returned to 100 percent power on July 10. Unit 2 completed the inspection period at 100 percent power.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

Using Inspection Procedure 71707, the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety conscious.

Review of drawings and schematics issued by engineering in the control room revealed that on some drawings there were areas which were illegible. After inspectors raised the concern, the licensee performed a review of all control room drawings and schematics. The review identified numerous drawings and schematics that had portions that were illegible and needed replacement. The licensee issued an action request (AR) to document the problem and initiated actions to replace drawings that were not completely legible. The inspectors found that the actions taken by the licensee were responsive to the concerns.

Based upon the number of discrepancies noted the inspectors concluded that the licensee's program for maintaining up-to-date drawings in the control room failed to ensure that the drawings were completely legible.

O1.2 Unit 2 Manual Reactor Trip on Loss of MFPs

a. Inspection Scope (71707)

On July 2, at 4:56 a.m. PDT, Unit 2 operators initiated a manual reactor trip after the overspeed trip of both MFPs. The plant response after the trip was uncomplicated. The inspectors reviewed the operator's response to the event, the investigation of the cause, and corrective actions prior to the return to power.

b. Observations and Findings

Unit 2 was at 100 percent power when operators observed a decrease in generator output, followed by indications of loss of suction pressure to the MFPs. The speed of both MFPs increased in response to the decreased flow to the steam generators

and the MFPs tripped on overspeed. The operators responded to the secondary plant transient by decreasing reactor power and manually tripped the reactor when the MFPs tripped. The unit was stabilized in Mode 3 with the reactor coolant system (RCS) at normal no-load temperature and normal operating pressure. Emergency Diesel Generator (EDG) 2-2 had started on the electrical power transfer from auxiliary power to startup power (as had occurred in past reactor trips). Main Feedwater Regulating Valve 2-2 had shifted to manual during the transient (due to controller setpoints for limiting plant response to out-of-range parameters). Both of these automatic actions were considered to be expected responses to the transient. The licensee developed a plan to evaluate the cause of the loss of suction pressure to the MFPs, reviewed the plant response to the transient, evaluated the EDG and main feedwater regulating valve response, and implemented their 2-day outage plan.

The review of the transient identified that the interaction of several control valves in the condensate system, and their associated controllers, caused the flow to the suction of the MFPs to decrease below that required to maintain adequate feedwater flow to the steam generators. A recent throttling of a manual valve in parallel with these control valves is believed to have contributed to the plant's inability to recover from this condensate flow transient. The licensee fully opened the manual valve and adjusted the controllers for the control valves to slow the valve response to system transients. The unit was returned to power operation, with holdpoints at several power levels to obtain condensate and feedwater system data and evaluate the plant's response to normal steady state conditions and planned transients.

The evaluation of the trip indicated that the initiating event was a sudden loss of condenser vacuum. The licensee identified all known causes for a loss of vacuum and by review of recorders, inspection of equipment, and questioning of personnel eliminated each as the initiator for this plant trip. After the initial loss of vacuum, the response of other plant parameters and equipment operation appeared to have been adequately evaluated.

c. Conclusions

The operators actions in response to the transient and subsequent unit trip mitigated the effect of the loss of both MFPs on other plant parameters. The licensee performed an exhaustive evaluation of the trip and the plant response. Although the initiating cause for the loss of vacuum had not been identified, corrective actions were taken to improve the unit's ability to withstand a similar transient without a plant trip. The corrective actions have also been implemented in Unit 1 where applicable. The subsequent return to power was performed cautiously with prudent testing and adjustments made at appropriate power levels.

O1.3 Review of Temporary Modifications, Jumper Log

a. Inspection Scope (71707)

The inspectors reviewed the jumper log and walked down the jumpers (Unit 2 Jumper Log 97-017) associated with monitoring the condensate system parameters in troubleshooting the cause of the Unit 2 trip on July 2. The applicable Administrative Procedure, CF4.ID7, "Temporary Modifications - Plant Jumpers and M&TE", Revision 5, was also reviewed.

b. Observations and Findings

The jumpers did not result in inoperability of safety-related equipment and had appropriate evaluations and actions implemented. No conflicts with TS or the Updated Final Safety Analysis Report were identified. The jumpers and instrumentation associated with the condensate system monitoring were installed as described on the jumper log, with adequate restraints to prevent becoming a hazard to plant equipment. A combustible loading permit had been obtained and was located at the job site. Information tags were attached to the jumpers as required by the administrative procedure. Data from the instrumentation was being recorded periodically and the strip charts were routinely collected and evaluated. The jumper log was reviewed and contained the necessary information, licensing basis impact screening, and approvals as required by the administrative procedure. Several administrative errors were noted in the jumper log including: an error in annotating the screening for the LBIE; inappropriate line out of the signatures for installation and verification of two jumpers; and an attached drawing had an incorrect markup. These administrative discrepancies were identified to the licensee and were corrected.

c. Conclusions

The temporary modification for monitoring the condensate system after the July 2 transient had been adequately planned and reviewed, and the jumpers were properly installed as described in the jumper log. In general, the jumper logs were properly maintained. The errors noted in Jumper Log 97-017 indicated a lack of attention to detail and failed to meet management expectations.

O1.4 Walkdown of Unit 1 480V Switchgear

a. Inspection Scope (71707)

The inspector performed a routine walkdown of the Unit 1 480 volt switchgear on July 1, 1997, and reviewed the following procedures:

- OP J-10:IV, Revision 19, Instrument AC System - Transfer of Panel Power Supply

- Switching Order dated 5/24/97
- OP O-13, Revision 9, Transferring Equipment to/from Alternate Power Source
- OP1. DC20, Revision 6, Sealed Components

b. Observations and findings

During a walkdown of the Unit 1 480V Bus F switchgear, the inspector noted that Breaker 52-1F-27 was closed, although it's normal position is open. The breaker supplies backup power to 120V instrument Panels PY-15, PY-16, and PY-17. PY-15 and PY-16 are normally powered from the vital 480V Busses G and H, respectively, while PY-17 is normally powered from a nonvital 480 volt bus.

Procedure OP J-10:IV, Section 6.21, specifies the procedure for transfer of PY-16 from backup to normal power. Procedure OP J-10:IV, Step 6.21.4, requires that the backup 480V supply Breaker 52-1F-27 be opened following realignment of the normal power supply. Additionally, Note 1 in Section 6.21 specifies that the use of a transfer change form per OP1.DC20 is required. Procedure OP1.DC20, paragraph 4.2.6 requires that removal and replacement of seals shall be documented and that a transfer switch change form shall be issued and approved prior to the component seals being broken. Following completion of the sealed component change form it is required to be filed in the control room.

During 1R8, the 480V Bus G was deenergized to perform inspections of 4 kV breakers. During the time that Bus G was deenergized, the power supplying PY-16 was realigned to its alternate power source. Following completion of the maintenance, after reenergizing Bus G, the licensee aligned the normal power supply to PY-16. During the restoration, breaker 52-1F-27 was left closed contrary to procedural requirements. Additionally, there was no record of the removal and reinstallation of the seal installed on the power transfer switch, contrary to procedural requirements.

A walkdown of the other power supplies was performed and revealed that all three of the PY panels were powered from their normal power supply. Breaker 52-1F-27 being closed and out of its normal position did not adversely impact the operability of any components since the normal power supplies were properly aligned; however, the lack of documentation and the fact that breaker 52-1F-27 was not properly restored to the open position, indicated that applicable procedures were not followed when performing the restoration of normal power to the instrument panel. The failure to follow procedures for sealed components and for realignment of instrument Panel PY-16 is a violation of 10 CFR Part 50, Appendix B, Criterion V (VIO 50-275/9710-01).

c. Conclusions

Operators failed to follow procedures for aligning 480 vital power to instrument Panel PY-16.

O8 Miscellaneous Operations Issues (92700)

O8.1 (Closed) Licensee Event Report (LER) 50-275/95014 Revision 0: diesel generators started and loaded as designed upon failure of Auxiliary Transformer 1-1 due to inadequate/ineffective procedures relating to the control of grounding devices.

This LER was written to document the loss of all offsite power following the failure of Auxiliary Transformer 1-1 while the startup bus was deenergized for maintenance. Following this event the NRC conducted a special inspection, the results of which are documented in NRC Inspection Report 50-275;323/95-017. Following review of the LER, the inspector determined that no new violations were documented and that the violations issued in the inspection report documented the pertinent issues pertaining to the event. The associated corrective actions for this event are documented in the response to the violations and the NRC's review and closure of the violations, therefore, no additional review is necessary to close this LER.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations

a. Inspection Scope (62707)

The inspectors observed all or portions of the following work activities:

- C0152981 Remove and replace TM-3 and POT-32 components in the TCV-23 control circuitry
- R0162375 Inspect auxiliary saltwater (ASW) Pump 1-1 vault drain line check Valve SW-1-987

M1.2 CFCU Timer Replacement Postmodification Testing

a. Inspection Scope (62707)

The inspectors observed portions of postmodification Procedure PMT-23.26, "CFCU 2-3 Time Delay Relays Replacement Test", Revision 0.

b. Observations and Findings

The Design Control and Test Group performed the postmodification test for the replacement of the auto transfer timer and safety injection (SI) timer for CFCU 2-3, in accordance with Procedure PMT-23.26. The initial section of the procedure tested the control circuit while electrically isolated from the motor electrical power. This allowed a thorough checkout of the interlocks and seal-in paths without undue cycling of the CFCU. Later sections of the procedure simulated autotransfer signals and SI signals to the control logic and verified by actual operation of the CFCU that the design modification had been properly designed and implemented.

The test briefings covered the necessary information (including test conditions, organizational interfaces, test requisites, expected alarms, expected component changes of state, and test connections) and the operators and other test participants clearly understood their tasks and responsibilities. The instrumentation was within calibration and appropriate for the test. Electrical safety precautions were used when connecting and disconnecting from energized equipment. Three-way communications were used by the operators and test personnel. Electrical schematic and connection drawings were available in the field and used to ensure the actions in the procedure and the plant response were understood before proceeding.

The personnel performing the test were knowledgeable of the equipment operation and the intent of the design change. The test procedure was performed and signed as written, with several minor on-the-spot-changes (OTSC). One OTSC was due to the procedure specifying incorrect fuse ratings. Another was due to a procedure error that verified a relay deenergized when in fact it did not change state. Both of these errors should have been noted and corrected prior to starting the test by a review of the tests performed on the Unit 1 CFCUs.

During the autotransfer part of the test, CFCU 2-5 was expected to stop and restart in slow speed; however, it initially remained running in fast speed and then shifted to slow speed. The test personnel and operators worked well together to stop the test and return plant equipment and test jumpers to a normal and safe configuration. The response of the CFCU was documented in an AR and evaluated before proceeding with the remainder of the test. The evaluation concluded CFCU 2-5 has performed as designed.

c. Conclusions

The personnel were well prepared and knowledgeable about the test. The test demonstrated the design modification had been implemented properly and satisfied the intent of the design change. In general, the procedure was thorough and properly used. Procedural deficiencies that required an OTSC should have been identified prior to the start of the test from lesson learned from the Unit 1

performance of the equivalent tests. Involved personnel responded cautiously and appropriately to the unexpected performance of CFCU 2-5.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

Selected surveillance tests required to be performed by TS were reviewed on a sampling basis to verify that: (1) the surveillance tests were correctly included on the facility schedule; (2) a technically adequate procedure existed for the performance of the surveillance tests; (3) the surveillance tests had been performed at a frequency specified in the TS; and (4) test results satisfied acceptance criteria or were properly dispositioned.

The inspectors observed all or portions of the following surveillances:

- STP R-13 Nuclear Power Range Incore/Excore Multiple-Point Calibration, Revision 9B
- STP R-41 RCS Temperature Instrumentation Data, Revision 6
- STP M-9A Diesel Engine Generator Routine Surveillance Test, Revision 43A

b. Observations and Findings

During the performance of STP R-13, the shift foreman, shift technical advisor and reactor engineering provided good support for the evolution. The reactor engineers clearly communicated the necessary plant conditions for data collection and the control operator made good use of the plant process computer and its displays to monitor and control plant parameters critical to obtaining high quality data for the test.

c. Conclusions

The inspectors found that the surveillances observed were being scheduled and performed at the required frequency. The procedures governing the surveillance tests were technically adequate and personnel performing the surveillance demonstrated an adequate level of knowledge. The inspectors noted that test results appeared to have been appropriately dispositioned.

M1.3 Surveillance Requirements for Containment Isolation Valve SI-1-8964

a. Inspection Scope (61726)

The inspectors reviewed STP I-1D, "Routine Monthly Checks Required By Licenses", Revision 48, for compliance with TS 4.6.1.1.a, which requires containment isolation valves to be verified closed every 31 days.

b. Observations and Findings

The operating valve identification diagram for the SI system shows containment Penetration 51B with a line used to test check valve leakage and to fill the SI accumulators. A branch of this line, outside containment, has a pressure Indicator, 1-PI-942, with a normally closed manual isolation Valve, SI-1-8964. Valve SI-1-8964 is therefore part of the containment isolation barrier for a penetration that is required to be closed during accident conditions and should be verified to be in its closed position at least once per 31 days as required by containment integrity TS Surveillance Requirement 4.6.1.1.a. This valve was not verified as part of STP I-1D and was not verified closed every 31 days. The licensee was informed and the operating crew verified the valves on both units to be closed.

The licensee's position was that this valve is not a containment isolation valve and was not within the scope of Surveillance Requirement (SR) 4.6.1.1.a. Containment isolation valves operability is covered by TS 3/4.6.3 which clearly applies to active valves. The action statement for an inoperable isolation valve includes isolating the affected penetration by use of a deactivated automatic valve secured in the isolation position, or isolation by use of a closed manual valve or blind flange. The surveillance requirement for containment integrity, SR 4.6.1.1.a, addresses penetrations not capable of being closed by an operable containment automatic isolation valve, and verifying closed valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative controls as permitted by TS 3.6.3. The licensee believed that these requirements applied to main process valves only, and not to test vents and drains, or to instrumentation valves that might form part of the containment penetration boundary. The local leak rate test valves were included in Procedure STP I-1D because Standard Review Plan, Section 6.2.6, requires test, vent, and drain connections that are used to facilitate local leak rate testing or the performance of the containment integrated leak rate test should be administratively controlled and should be subject to periodic surveillance.

The NRC's TS Branch interpretation is that any valve associated with a containment penetration, no matter how small, is a containment isolation valve and the requirements of TS 3.6.1.1 apply.

Licensee's Corrective Actions

Following discussions with the NRC inspectors and TSB, the licensee began a thorough review of the containment penetrations to identify any additional valves that had not been included in Procedure STP I-1D. The penetrations to be reviewed were prioritized, with those penetrations most likely to have valves that were excluded reviewed first. Penetrations which receive local leak rate testing were reviewed last, since the valves associated with them were already included in Procedure STP I-1D. The review also included valves inside containment, which are excepted for the 31-day requirement of SR 4.6.1.1. if they are locked, sealed, or otherwise secured in the closed position. These valves inside containment are required to be verified closed during each cold shutdown except such verification need not be performed more often than once per 92 days. Those valves outside containment and accessible valves inside containment that had not been verified closed are being checked by operations and verified closed, and where applicable, downstream pipe caps were verified to be in place.

Related Industry Problems

In December 1992, Florida Power and Light Company was issued a violation for failure to maintain containment integrity by opening a drain valve in the containment spray system which was also a containment penetration boundary valve (documented in NRC Inspection Report 50-335;389/92-21). The licensee stated that they were not in TS 3.6.1.1 Action Statement because only valves identified by number in the TS or the Updated Final Safety Analysis Report were containment isolation valves. The licensee specifically stated that vents, drain and test valves were not within the scope of the TS 3.6.1.1. The NRC Staff review concluded any valve which isolates a containment penetration, no matter how small, is a containment isolation valve and is required to be within the scope of TS 3.6.1.1.

In September 1996, NRC Region II requested NRR to evaluate a Catawba site document entitled "Catawba Nuclear Station Containment Integrity Review" for consistence with various regulations and codes. The licensee's document contended that ANSI N271-1976 clearly differentiated between test connection vents and drains and containment isolation valves. It further contended that the Standard Review Plan Sections 6.2.4 and 6.2.6 also distinguish between containment isolation valves and test connections vents and drains, and therefore, do not consider test vent and drain connections to be containment isolation valves. The staff's response was that test vents and drains (TVDs) are containment isolation valves (CIVs). The fact that ANSI N271-1976 has a definition for TVDs does not mean the TVDs cannot also be CIVs. The staff's position was that the TVDs should meet the 31-day SR 4.6.1.1.a for CIVs.

In November 1996, the resident inspector for Braidwood Station requested NRR's assistance in interpreting TS 3/4.6, "Primary Containment" SR 4.6.1.1.a on a vent and drain valve on a SI test line local pressure indicator. The licensee believed that

SR 4.6.1.1.a , which requires the licensee to verify that all penetrations are secured every 31 days, applies only to penetrations that are not capable of being automatically isolated. They stated that because the penetration attached to these valves is capable of being closed by an operable containment automatic isolation valve, SR 4.6.1.1.a does not apply to these drain and vent valves. NRR's TS Branch reviewed the request and concluded that these drain and vent valves should be included in the SR testing. The staff considers any valve associated with a containment penetration no matter how small to be a CIV and therefore the requirements of TS 3.6.1.1 apply to them. Part of the basis for the applicable general design criteria is that a single failure will not prevent or degrade containment integrity.

c. Conclusions

The licensee's implementation of TS SR 4.6.1.1.a was inadequate in that it failed to verify that Valve SI-1(2)-8964 was closed at least once per 31 days. This is a violation of TSs (VIO 50-275;323/9710-02).

M8 Miscellaneous Maintenance Issues (92700)

M8.1 (Closed) LER 50-275/94-023-00: TS 3.0.4 was not met when Unit 1 entered Mode 3 on May 4, 1994 with CCP 1-1 inoperable. CCP 1-1 was inoperable because the total pump flow was greater than the TS 4.5.2h.1(d) limit.

The excessive CCP 1-1 flow was caused by erosion of the CCP miniflow restricting orifices and leakage through flow control bypass Valves 8387B and C. The excessive erosion of the restricting orifices is believed to have been caused by extended use of the CCPs for normal charging. The licensee's corrective actions included verifying the current flow balance satisfies the TS requirements, replacement of the restricting orifices and adding flow measurement equipment in the mini-flow path, and the use of the positive displacement pump as the normal supply for charging flow. The equipment modifications were completed in Unit 1 during the 1R8 outage, and are scheduled to be implemented in Unit 2 during the 2R8 outage. The review of the LER by the inspector revealed no new information. This licensee-identified and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-275/97010-03).

M8.2 Use of Dow Corning RTV-732 in Containment

a. Inspection Scope (92902)

As documented in NRC Inspection Report 50-275;323/97-06 the appropriateness of the use of RTV-732 sealant to fill in gaps between the base of the reactor coolant pumps and the lubricating oil collection pans was questioned by the inspectors.

The inspectors performed additional reviews of the actions taken by the licensee following identification of this issue.

b. Observations and Findings

Following the initial identification of the concern,the licensee determined that RTV-732 sealant had not been qualified for use in containment and would be replaced with a qualified caulking material. Testing performed by the licensee determined under harsh environmental conditions, that could be encountered in certain areas of containment during accident conditions, that the RTV-732 sealant material may not adhere to the surface that it had been applied. A search was performed to determine the work orders which specified the use of the RTV-732 sealant stock code for work performed in containment. This search identified that the use of the RTV-732 sealant had been specified on the CFCU 2-2 fan housing. The licensee subsequently reviewed the calculation for debris clogging the containment sump and determined that the additional caulking material, if transported to the containment sump during an accident, would not significantly reduce the margin to safety in the calculation for adequate emergency core cooling system suction from the sump. Based upon the remaining margin in the calculation this conclusion appeared reasonable. A prompt operability assessment was written based upon this information.

The licensee's assessment was questioned by the inspector, since the work order search did not identify all of the reactor coolant pump oil collection trays as having had RTV-732 applied. Further review by the licensee revealed that RTV-732 had been available for use in the tool crib in containment during outages. Specifically, during 1R8, six tubes of RTV-732 sealant had been used in containment based upon tool crib inventory records. The licensee was unable to identify where the RTV-732 sealant had been used. Based upon the previous analysis, the inspector agreed with the licensee's conclusion that this did not change the validity of the licensee's initial conclusion regarding the potential for clogging the containment sump. However, the use of unauthorized material in containment is being considered as an inspection followup item pending the licensee's review of the impact of RTV use during the outage to assure environmental qualification requirements continue to be met (IFI 50-275;323/9710-04).

c. Conclusions

The licensee's initial assessment of the effect of the use of RTV-732 sealant in containment was incomplete in that it failed to identify the total quantity of the sealant used and the locations where it had been applied.

III. Engineering

E1 Conduct of Engineering

E1.1 Calculation in Support of Component Cooling Water (CCW) Flow Balance (37551)

The licensee identified an error in a calculation for the CCW system. Calculation M-916 specified instructions to be incorporated into test Procedure STP V-13A, "CCW Flow Balancing", to establish the CFCU outlet valve positions to maintain CCW system hydraulic flow balance. The instructions included a correction for a difference in elevation of pressure indicators. This correction was subtracted from the measured values; however, the correction should have been added. The licensee performed a prompt operability assessment and concluded that the CCW and CFCU systems remained operable. The licensee indicated that both the calculation and the surveillance procedures were being revised to correct this error.

The surveillance procedure was reviewed by the inspector and an additional error in correcting for elevation of the test pressure indicators was identified. The elevation of each indicator was determined in reference to the floor elevations which were incorrectly specified in the procedure. This was identified to the licensee for incorporation into the procedure revision. The operability assessment and corrective actions adequately addressed these errors.

E1.2 CFCU Auto Bus Transfer and SI Timer Replacement Design Change

a. Inspection Scope (37551)

The inspector reviewed the Design Change Package E-049344, Revision 1, and Design Change Notice 1-EE-049355, Revision 1, to assess the design control and installation of plant modification processes.

b. Observations and Findings

Design Change Package E-049344 replaces all Unit 1 existing 10 CFCU timers (5 SI and 5 auto bus transfer timers) with more accurate digital Agastat DSC timers. The wiring changes use existing 42X relays as auxiliary relays to meet seismic requirements and facilitate installation, as well as change the CFCU start logic such that an autotransfer signal will start the CFCU in slow speed regardless of the High/Low speed switch position. The design change was the result of Quality Evaluation Q0010264, which identified the existing Agastat 7000 series auto bus transfer timers drift outside the range allowed by TS.

The effect on plant operation was addressed by the design change package, including changes to operating, surveillance and administrative procedures. The engineering evaluations included seismic and environmental considerations as well

as electrical calculations for loading of the EDGs and dropout voltages of the timers. The probabilistic risk assessment concluded that the change would be beneficial since it does not introduce any new failure modes and the new timers are more reliable.

The design change package specified the functional tests to be performed and the acceptance criteria to be satisfied to demonstrate the modification functions as intended. The design was reviewed by appropriate interface organizations, including operations, maintenance and other engineering disciplines. An LBIE screen was performed and determined the need for a LBIE. The LBIE was performed, but was not properly annotated as to whether an unreviewed safety question was involved or whether a TS change was required. It appeared from the LBIE that neither was involved. The discrepancy was pointed out to the licensee and corrected. A Final Safety Analysis Report (FSAR) change was identified and the FSAR Update change request was submitted. Revisions to Design Criteria Memorandum S-23, heating, ventilation and air condition, and S-63, 4KV Systems, have been submitted but not yet incorporated.

Design Change Notice 1-EE-049355, Revision 1, contained the installation requirements to implement the above plant modification. It specified the necessary sequence of installation and the system lineup, and identified the TS requirements applicable during installation. The postmodification/functional tests and applicable acceptance criteria were specified. The associated field change forms were reviewed and found to correct a sneak circuit, clarify instructions, allow use of new wire or existing spare wire and to correct minor errors.

The identified procedure changes were reviewed and found to have been implemented. Control room drawings were reviewed and contained the applicable markups.

c. <u>Conclusions</u>

The design control process effectively coordinated the necessary activities from the initial design planning to final implementation. Interfaces between various organizations were controlled to ensure supporting activities were completed to fully implement the plant modification. The number of field changes was not excessive, but does indicate improvement could be made in the planning of the installation. The design change appeared to accomplish its intended function.

E8 *Miscellaneous Engineering Issues (92903)*

E8.1 <u>(Closed) IFI 50-275/96006-04:</u> licensee long-term corrective actions to improve source range instrument reliability. The control console startup rate meter for source range Instrument NI-32 stuck on its bottom peg on several occasions. The licensee had previously identified that the source and intermediate range startup rate meters were the wrong current range (-0.1 to 0.9 madc verses the correct

range of -0.1 to 1.0 madc) and were procuring replacements. The wrong range meters were calibrated and performed the required function, and therefore were acceptable until the new meters could be installed. During 1R8 outage, the startup rate meters for source and intermediate instruments were replaced with new meters. The applicable calibration Procedure, STP I-37-N46 NIS Comparator and Rate N46/N37 Calibration, was performed and the meters were observed prior to startup to operate properly without sticking. The Unit 2 meters are scheduled for replacement during the next refueling outage.

E8.2 (Closed) URI 50-275;323/96021-06: ASW TS interpretation more restrictive than TS.

The inspector reviewed the unresolved item (URI) that was written after noting that the TS 3.7.4.1 ACTION statement, as written, no longer specified the lowest functional capability or performance levels for the ASW system.

Current TS 3.7.4.1 Requirements
The TS limiting condition for operation, requires two operable ASW trains, with an ACTION statement that with only one ASW train operable the second train must be restored to OPERABLE status within 72 hours or be in HOT STANDBY within the following 6 hours and in COLD SHUTDOWN within the following 30 hours.

Licensee TS 3.7.4.1 Interpretation
The TS interpretation 95-08, implemented by the licensee, imposed more restrictive requirements for ASW pumps and CCW heat exchangers in order to ensure that adequate equipment was available to prevent overheating the CCW system. Although this was different than originally stated in the FSAR, the licensee had performed a 10 CFR 50.59 evaluation and revised the FSAR to reflect the system configuration requirements.

Issues identified during previous NRC inspections
During the review of this issue, the inspector determined that the NRC had reviewed related issues in a safety system functional inspection in January 1989 (see NRC Inspection Rreport 50-275/89-01) and later during the review of the LER submitted by the licensee following the inspection (LER 50-275/84-40, see NRC Inspection Report 50-275;323/89-013). Following these inspections, the NRC determined that the licensee had not adequately translated plant system configuration design assumptions into procedures. Consequently, operating procedures or instructions did not provide adequate guidance for system operation under certain design basis conditions. A recent review of the associated procedures determined that the licensee had completed revisions to the procedures which appeared to provide adequate guidance.

CCW System Design Change Raised Maximum Allowed Temperature
The licensee recently issued a design change which raised the design temperature of the CCW system to allow a temperature transient in the system to 140°F for a period not to exceed 6 hours. Based upon the increase in the maximum allowable system temperature, the additional restrictions implemented by the TS interpretation are no longer required; however, in order to ensure maximum heat removal capability during emergencies operators are instructed to place a second CCW heat exchanger in service early in the emergency operating procedures. Additionally, the licensee has submitted a license amendment, based upon the new analysis, which requires only one ASW pump and one CCW heat exchanger, as assumed in the safety analysis, in order to provide sufficient heat removal to mitigate a design basis accident. Based upon the analyses, this unresolved item is closed.

E8.3 (Closed) Violation 50-323/96023-05: failure to initiate an AR to address loose fasteners. On November 19, 1996, during a routine tour of the Unit 2 480V vital switchgear rooms, the resident inspectors identified a number of loose fasteners associated with the breaker front panels. These findings were discussed with the system engineer and the onsite mechanical engineer responsible for seismic qualification and evaluation of plant equipment. On November 22, 1996, the system engineer provided the resident inspectors a technical evaluation regarding the seismic qualifications of the affected breaker panels with the observed loose fasteners and resolved the resident inspectors' concerns regarding the impact of the observed loose fasteners. However, the resident inspectors determined that as of December 2, 1996, an AR had not been initiated to address and document the quality problem associated with the loose fasteners contrary to Procedure AD4.ID8, Revision 1, "Identification and Resolution of Loose, Missing or Damaged Fasteners."

On February 14, 1997, the licensee responded to the Notice of Violation by means of PG&E letter DCL-97-024 and agreed with the violation. The licensee stated that the following corrective actions had been taken:

- An AR was written to document the observed fastener deficiencies and the corrective actions that were taken.
- The system engineer was trained regarding the need to document quality problems as required by Procedure AD4.1D8.
- A case study was written and distributed to plant personnel to provide lessons learned regarding the event.
- Electrical and instrumentation and controls system engineers were given specific training regarding the requirements of Procedure AD4.1D8.

During the current inspection period, the inspectors reviewed the licensee corrective actions for the violation as stated in the licensee response. The inspectors determined the following:

- The licensee issued AR A0418696 on December 3, 1996, to document the observed loose fasteners and the corrective actions for the fasteners. The AR also addressed the procedure violation regarding the failure to write the AR.

- AR A0418696 also documented the preparation and issuance of a case study regarding the violation on February 28, 1997. The inspectors reviewed case study 63 dated February 27, 1997, regarding the violation. The inspectors noted that the case study provided a good description of the problems identified and lesson learned. The case study appropriately discussed the need to comply with procedural requirements and the need to document identified fastener problems.

- AR A0418696 also documented the completion of training of all electrical and instrumentation and controls system engineers on March 21, 1997. The inspectors discussed the training with the original system engineer involved with the problem. The inspectors noted that the system engineer was also the individual who prepared the case study and was now fully aware of the procedural requirement associated with documenting fastener problems.

The inspectors concluded that the licensee had performed reasonable corrective actions for the identified violation. Additional problems noted with the control of 4 kV breakers were documented in a violation (VIO 50-275;323/97-06-02). Corrective actions for those specific instances will be evaluated during the review of the licensee's response to the violation and will consider the adequacy of the corrective actions for previous similar violations. Based on the above, Violation 50-323/96023-05 was closed.

IV. Plant Support

R4 Staff Knowledge and Performance in Radiological Protection and Chemistry Controls

R4.1 Primary Coolant Sample Procedure

a. Inspection Scope (71750)

On June 12, the inspector observed the drawing of RCS daily samples at Units 1 and 2 primary sample sinks. Procedure CAP E-1, Revision 17A, "Sampling of Primary Systems," was also reviewed.

b. Observations and Findings

The chemistry technician was knowledgeable of the sampling procedure and the configuration and operation of valves at the primary sample sinks. The chemistry technician was sensitive to potentially contaminated areas and demonstrated proper

radiological controls while working in the sample sink. A sufficient volume of coolant was purged through the sample lines to ensure that representative samples were drawn.

The inspector verified that the"issued-for-use" copy of Procedure CAP E-1 located in the primary sample room for each unit was the latest revision. After the samples were drawn, the inspector observed portions of the sample analysis. The inspector observed that the chemistry technician was knowledgeable of the required analysis and use of the analytical equipment.

c. Conclusions

The primary samples were obtained satisfactorily. The chemistry technician was knowledgeable of the sampling procedure, equipment use, and radiological controls.

P1 Conduct of Emergency Preparedness Activities

P1.1 Licensee Preparedness for Criticality Accidents

a. Inspection Scope (71750)

The inspectors examined the implementation of 10 CFR 70.24, "Criticality Accident Requirements."

b. Observations and Findings

The licensee did not have an exemption from the requirements of 10 CFR 70.24 in the operating license. However, on April 3, 1997, the licensee submitted a request for exemption from the requirements of 10 CFR 70.24. Currently, the licensee has procedures and equipment in place that appear to meet the requirements of 10 CFR 70.24.

The inspectors verified that Diablo Canyon has two area radiation monitors positioned near the fuel storage and handling areas which are intended for the monitoring of potential criticality events. Area Radiation Monitor (RM)-59 was positioned near the new fuel storage vault and area RM-58 was positioned near the spent fuel pool. In addition, licensee procedures specified the placement of portable criticality monitors near the new fuel storage vault and cask washdown areas, as applicable, during fuel handling activities. Licensee personnel indicated that the radiation monitors were capable of meeting the requirements of 10 CFR 70.24(a)(1). The monitors energized local audible alarm signals if setpoints are reached, and RM-58 and RM-59 energize control room annunciators upon alarm.

The alarm response procedures for the remote alarm feature in the control room have instructions that specify the actions to be taken and the requirements for initiating the emergency response plan. The actions include the evacuation of

personnel from the fuel handling areas. Procedure OP B-8H, "Nonrefueling Fuel Handling Instructions," also specifies the actions to be taken if any of the criticality monitors alarm.

Licensee procedures specify the performance of an on-station evacuation drill prior to receiving fuel for each operating cycle. Licensee procedures also specify preshift briefings of actions to be taken in the event of a radiation alarm or portable criticality monitor alarm.

c. Conclusions

The licensee currently meets the requirements of 10 CFR 70.24 for criticality accidents.

P8 Miscellaneous Security and Safeguards Issues (92904)

P8.1 (Closed) Violation 50-275/9103-04: inattentiveness to duties.

This violation was issued after a security officer, assigned as a compensatory measure watch, was observed to be inattentive. The licensee acknowledged the violation and described corrective actions in a letter to the NRC dated May 3, 1991. Due to an oversight, this item remained open in NRC records. However, the inspector was onsite at the time of the violation, recalls the licensee's immediate corrective actions and actions to prevent recurrence, and recalls that those actions were appropriate and effective. The inspector confirmed that the licensee's actions were as described in their May 3, 1991, letter, and concludes that no additional action is warranted for this matter.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management at the conclusion of the inspection on July 25, 1997. In the meeting the licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. E. Coley, Engineer, Regulatory Services
W. G. Crockett, Manager, Nuclear Quality Services
T. F. Fetterman, Director, Instrumentation and Control Engineering
T. L. Grebel, Director, Regulatory Services
J. A. Gregerson, Engineer, Engineer, Engineering Services
D. J. Hampshire, Senior Engineer, Balance of Plant Engineering
M. J. Jacobson, Manager, Nuclear Quality Services
S. C. Ketelsen, Senior Engineer, Regulatory Services
S. D. LaForce, Engineer Regulatory Services
J. E. Molden, Manager, Operations Services
M. G. Mosher, Acting Director O&SP, Nuclear Quality Services
D. H. Oatley, Manager, Maintenance Services
H. J. Phillips, Director, Engineering Services
R. P. Powers, Manager, Vice President DCPP and Plant Manager
J. A. Shoulders, Director, Support Engineering
R. L. Thierry, Engineer, Nuclear Technical Services
D. A. Vosburg, Director, NSSS Engineering Services
R. A. Waltos, Director, Engineering Services
R. C. Whitsell, Director, Nuclear Quality Services

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering

IP 61726: Surveillance Observations

IP 62707: Maintenance Observations

IP 71707: Plant Operations

IP 71750: Plant Support

IP 92700: Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities

IP 92902: Followup - Maintenance

IP 92903: Followup - Engineering

IP 92904: Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-275/97010-01	VIO	Failure to follow procedures for alignment of 480V power supply to instrument panel PY-16
50-275;323/97010-02	VIO	Failure to verify CIV SI-1(2)-8964 closed at least once per 31 days
50-275/97010-03	NCV	Mode 3 entry with CCP inoperable due to total pump flow exceeding TS limit
50-275;323/9710-04	IFI	Uncontrolled use of Dow Corning RTV-732 sealant in containment

Closed

50-275/94023-00	LER	CCP flow greater than TS due to erosion of mini-flow restricting orifice
50-275;323/97010-03	NCV	Mode 3 entry with CCP inoperable due to total pump flow exceeding TS limit
50-275/9506-02	IFI	ASW vault drain calculation for medium energy line break did not address the limited capacity of the intake structure sump pumps
50-275/96006-04	IFI	Licensee long-term corrective actions to improve source range instrument reliability
50-275/95014-00	LER	Diesel Generators started and loaded as designed upon failure of auxiliary transformer 1-1 due to inadequate/ineffective procedures relating to the control of grounding devices
50-275;323/96021-06	URI	ASW TS interpretation more restrictive than TS
50-323/96023-05	VIO	Failure to initiate an AR to address loose fasteners
50-275/9103-04	VIO	Inattentiveness to Duties

Discussed

50-275;323/97-06-02	VIO	loose fasteners on 4kV breaker cubicle

LIST OF ACRONYMS USED

AR	action request
ASW	auxiliary saltwater
CCP	centrifugal charging pump
CCW	component cooling water
CFCU	containment fan cooler unit
CIV	containment isolation valve
DCPP	Diablo Canyon Power Plant
EDG	emergency diesel generator
FSAR	final safety analysis report
IFI	inspection followup item
LBIE	licensing basis impact evaluation
LER	Licensee Event Report
MFP	main feedwater pumps
NCV	noncited violation
NRR	Office of Nuclear Reactor Regulation
OP	Operations Procedure
OTSC	on-the-spot-change
PDR	Public Document Room
RCS	reactor coolant system
SI	safety injection
SR	surveillance requirement
TS	Technical Specification
TVD	test, vents, and drains
URI	unresolved item
1R8	Unit 1 eighth refueling