ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-275, 50-323
License Nos.:	DPR-80, DPR-82
Report No.:	50-275/96-20; 50-323/96-20
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:	August 18 through September 28, 1996
Inspectors:	M. Tschiltz, Senior Resident Inspector S. Boynton, Resident Inspector D. Corporandy, Project Inspector D. Allen, Reactor Inspector, Region IV
Approved By:	R. Huey, Acting Chief, Project Branch E Division of Reactor Projects
ATTACHMENT:	Partial List of Persons Contacted List of Inspection Procedures Used List of Items Opened, Closed, and Discussed List of Acronyms

EXECUTIVE SUMMARY

Diablo Canyon Nuclear Power Plant, Units 1 and 2
NRC Inspection Report 50-275/96-20; 50-323/96-20

Operations

- Management demonstrated conservative decision making when both units were curtailed during periods when wildland fires had the potential to jeopardize two of the three offsite power distribution lines (Section O1.1).

- Investigative and corrective actions to address deficiencies with the controls established for the moveable incore detector system keys during containment entries were not timely or comprehensive (Section R1.1).

Maintenance

- Inadequate work planning and procedures were identified as significant contributors to the deformation of spent fuel pool cooling piping during the application of a freeze seal. A noncited violation was identified (Section M1.1.1).

- Maintenance activities associated with charging pump relief valve weld replacement were well coordinated and properly performed (Section M1.1.2)

- Maintenance workers did not replace tripper cams and tripper arm assembly that had abnormal wear during overhaul of a motor-operated damper actuator. Failure to replace the tripper cams resulted in the inability to place the valve in the manual mode during subsequent testing (Section M1.1.3).

Engineering

- Engineering aggressively resolved equipment concerns associated with recently installed 4160V breaker auxiliary switches and with the chemical and volume control system (CVCS). This was demonstrated by the accomplishment of several design changes made to improve system reliability and reduce unnecessary operation of engineered safety feature equipment (Sections E1.1 and E1.2).

- Routine system engineer system walkdowns failed to identify and evaluate 12 CVCS system valves with evidence of packing leakage (Section E1.2).

Plant Support

- Procedural requirements for establishing a fire watch were not met prior to commencing welding in the turbine building. A noncited violation was identified (Section F1.1).

- A superseded revision of a chemistry procedure was utilized when drawing a primary sample. The procedure for sampling, which had been revised in April, was used on a daily basis and was required to be verified to be the latest revision every

30 days. Similar problems were noted with "issued-for-use" documents in NRC Inspection Report 50-275/96-06; 50-323/96-06. A violation was identified (Section R3.1).

- There has been a noted improvement in the general housekeeping and radiological work practices observed in the Fuel Handling Building areas designated for work on radioactive components (Section R8.1).

Report Details

Summary of Plant Status

Unit 1 began this inspection period at 100 percent power. On August 19, the unit was curtailed to 70 percent power due to a wildland fire that threatened two of the three 500 kV transmission lines from the plant. The unit returned to 100 percent power on August 20. The unit was briefly curtailed to 70 percent power on August 21, when there was, again, a perceived threat to the 500 kV lines from the wildland fire. Following the curtailment, the unit returned to and remained at 100 percent power for the balance of the inspection period.

Unit 2 began this inspection period in power ascension at 90 percent power, following a unit trip on August 10. Unit 2 returned to full power on August 18. On August 19, the unit was curtailed to 70 percent power due to a wildland fire that threatened two of the three 500 kV transmission lines from the plant. The unit returned to 100 percent power on August 20. The unit was briefly curtailed to 70 percent power on August 21, when there was, again, a perceived threat to the 500 kV lines from the wildland fire. Following the curtailment, the unit returned to and remained at 100 percent power for the balance of the inspection period.

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.

Wildland fires that burned in areas in the vicinity of two of the three 500 kV distribution lines created an increased potential for the loss of these lines. Based upon the load carrying capacity of the 500 kV line not threatened by the fire and the potential for grid instability, plant management made the decision to curtail both units to 70 percent power. These management actions were viewed as proactive since they limited the potential for unplanned plant transients and challenges to plant systems and operators.

O2 Operational Status of Facilities and Equipment

O2.1 Effluent and Environmental Radiation Meteorological Monitoring

a. Inspection Scope (71750)

Radiation stack monitor recorder traces were reviewed and the operability of the plant's meteorological indicators was audited for the period from August 12-31, 1996.

b. Observations and Findings

Primary meteorological instrumentation for the plant was out of service for the period reviewed within the inspection scope because of a planned replacement of the electronic instrumentation. The replacement was completed, and the primary meteorological instrumentation was returned to service on September 5, 1996. During the period that the primary meteorological tower was unavailable, the backup meteorological instrumentation was operable in accordance with Technical Specification (TS) 3.3.3.4, which requires one of the two channels to be operable.

Radiation monitor recorder traces were reviewed for Radiation Monitors RM-28R, the particulate sampler; RM-14R, the noble gas activity monitor; and RM-24R, the iodine sampler. Measured radiation activity levels for the monitors remained at less than 1 percent of the radioactivity release limit during the period covered within the scope of the inspection. Licensee Equipment Control Guideline 39-4R7.4B, Revision 7, which requires that either the primary or redundant radiation monitor be operable at all times, was satisfied for the inspected period.

c. Conclusions

Effluent and environmental radiation monitors and meteorological monitors met the operability requirements of the applicable equipment control guidelines and TS for the period reviewed by the inspector. Measured radiation levels for noble gas, particulate, and iodine were significantly below alarm levels during the period, indicating that there were no apparent uncontrolled releases of radiation during the period.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) VIO 50-323/96002-01: failure to perform TS required channel checks of incore thermocouples. Licensee Procedure STP I-1D improperly allowed the use of the plant process computer (PPC) to perform the TS required monthly channel checks of the incore thermocouples. As a result, the requirements of TS 4.3.3.6 were not being met by the performance of Procedure STP I-1D.

The licensee researched the use of the PPC in performing other TS required surveillances and found that the PPC was also utilized to perform channel checks of the subcooled margin monitor. In response to these findings, the licensee revised Procedure STP I-1D to preclude the use of the PPC to perform the channel checks of the postaccident monitoring system instrumentation. Subsequently, the licensee established a new procedure, STP R-27A, for conducting the monthly channel checks of the postaccident monitoring panel incore thermocouples and removed that requirement from Procedure STP I-1D. A review of Procedure STP R-27A found that it adequately met the requirements of TS 4.3.3.6.

Because Procedure STP I-1D allowed the use of either the PPC or the local postaccident monitoring panel display, the licensee could not determine when channel checks of the incore thermocouples had, or had not, been performed adequately. Consequently, the licensee concluded that the requirements of TS 3.3.3.6 had not been met since initial operation of Units 1 and 2. As a result, the licensee issued licensee event report (LER) 50-275/84-050-00. Based upon this review, LER 50-275/84-050-00 is closed.

08.2 (Closed) VIO 50-275/95006-01: four examples of failure to follow procedural requirements. The violation identified four separate instances where operations personnel failed to follow procedures. The licensee determined that the violations had several different causes including: inattention to detail, inadequate self-verification, and failure to follow procedure. One of the four examples was subsequently determined to not be a violation based upon additional information that was not provided to the inspector during the initial inspection.

The inspector reviewed the licensee's corrective actions for the remaining three examples and determined that, in each instance, actions had been taken to prevent recurrence of the problem. Corrective actions included: revision to Operating Procedure C-7C:III, "Condensate Polishing System Transferring Resin Beds," issuance of an Operations Shift Order that reiterated the requirements for operation of sealed valves, and improvement and clarification of self-verification criteria. The inspector concluded that the licensee's corrective actions were appropriate.

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:

- C0146264 Unit 2 Control Room Ventilation System filter train blank flange installation
- PEP R-3E Replacement of Unit 2 moveable incore Detector A

b. Observations and Findings

The inspectors found the work performed under these activities to be accomplished in accordance with procedures. All work observed was performed with the work package present and in active use. Technicians were experienced and knowledgeable of their assigned tasks. The inspectors observed system engineers

monitoring job progress and that quality control personnel were present when required by the procedure. When applicable, appropriate radiation control measures were in place.

In addition, selected maintenance observations are discussed below.

M1.1.1 Spent Fuel Pool (SFP) Cooling Piping Freeze Seal

a. Inspection Scope (62707)

On August 14, after establishing a freeze seal on a spent fuel pool cooling system pipe, the licensee identified that the pipe had deformed in the vicinity of the freeze seal location. The inspector reviewed Procedure MP M-54.3, Revision 7, "Freeze Sealing of Piping," NMAC NP-6384-D, "Freeze Sealing (Plugging) of Piping," and Work Order (WO) C0145900. Actions taken to establish the freeze seal were discussed with both the engineering personnel investigating the event and the director of mechanical maintenance.

b. Observations and Findings

Under WO C0145900, mechanical maintenance technicians applied a freeze seal to the SFP demineralizer/filter outlet piping, upstream of Valve SFS-1-19. The freeze seal was required to establish conditions needed to repack the valve. As required by Procedure MP M-54.3, the work planner completed Attachment 8.1 to provide direction on the placement of the freeze seal jacket. A sketch of the pipe was provided in accordance with Step 5.0 of the attachment and showed the freeze seal area centered between two pipe hangers. According to the work planner, the length was based upon a request from the mechanical maintenance foreman to have an area cleaned and tested that was approximately three times the length of the jacket. The work planner was not aware that multiple jackets were to be used for this freeze seal at the time he completed Attachment 8.1.

To ensure the adequacy of the freeze seal, the maintenance personnel utilized three separate CO_2 jackets side-by-side to form what they believed would be a single, long ice plug in the pipe. Neither engineering nor the freeze jacket vendor were consulted on the acceptability of this arrangement. The as-installed configuration was not indicated in Attachment 8.1 and was not annotated in the WO. In utilizing three separate jackets, three distinct ice plugs formed. As the ice plugs grew, water trapped between the plugs was pressurized and, consequently, the yield strength of the pipe was exceeded. This resulted in observable bulging of the pipe at points between the jackets.

Both Procedure MP M-54.3 and NMAC NP-6384-D provide guidelines for the minimum spacing between freeze seals to protect against overpressurization of the water volume between the seals. Neither document provided guidance on the use of multiple jackets for establishment of a single freeze seal. For the SFP piping

application, the minimum distance between the seals should have been 3 feet. The actual distances between the freeze seals was less than 1 foot.

To preclude recurrence of this problem, the licensee has revised Procedure MP M-54.3 to explicitly prevent the use of multiple jackets for a single freeze seal application. The associated training module on freeze seals was also revised to reflect this precaution. The licensee assessed the continued operability of the deformed piping and concluded that the deformation caused by the freeze seals did not significantly reduce the pipe strength and that the remaining strength was adequate for the application. The affected section of pipe has been scheduled for replacement in October 1996.

c. Conclusions

The use of multiple freeze jackets resulted in the overpressurization and deformation of the SFP cooling piping. A violation was identified in that Procedure MP M-54.3 did not provide adequate guidance or controls over the attempted application. This licensee identified and corrected violation is being treated as a noncited violation consistent with Section VII.B.1 of the NRC Enforcement Policy (NCV 50-275/96020-01).

M1.1.2 Positive Displacement (PD) Pump Discharge Relief Valve (CVCS-2-RV-8116) Inlet Weld Replacement

a. Inspection Scope (62707)

The inspectors observed portions of work activities under WO C0145272, "Replace welds on inlet piping to PD Pump Discharge Relief Valve CVCS-2-RV-8116," and reviewed the related clearance and history of associated action requests (ARs).

b. Observations and Findings

On September 11 the inspectors reviewed the WO and found the level of detail of the instructions was adequate for the tasks being performed. The instructions for the work contained the applicable requirements for prejob briefing, control of foreign material, cleanliness, clearance, permits, laydown area, and use of tools and materials. The maintenance personnel were observed to be correctly performing the instructions in the sequence listed.

The related clearance was reviewed and the boundaries were found to adequately protect both equipment and personnel without unnecessarily impacting the operability of related equipment. The clearance tags were hung at the specified locations in accordance with administrative requirements.

The radiological controls for the job appeared appropriate. Required radiological surveys had been performed and personnel at the job site wore the required protective clothing. Potentially contaminated areas were correctly posted, and appropriate actions were observed to minimize the spread of contamination.

The tools and material at the job site were inspected. Welding rods were properly tagged and controlled and they corresponded with the work instructions and material controls. The test equipment was calibrated and tagged, and the personnel were knowledgeable of its proper use.

The combustible material permit and welding permit were posted and adequate personnel coverage was provided for the work, including radiological protection, supervision, and a monitor for foreign material exclusion. Involved personnel were knowledgeable of the necessary work practices and the history of the equipment problems.

A comparison of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," with the licensee's program for control of sensitized stainless steel indicated that the licensee's program was consistent with the Regulatory Guide. The completed work documentation was reviewed and was noted to have been properly completed. In addition, the training record of the welder performing the work documented that the individual was qualified to perform the welding.

c. Conclusions

The observed portions of the work were noted to have been accomplished in accordance with applicable procedures, and personnel involved with the maintenance were knowledgeable of procedural requirements for the work as well as the reason for performing the design change.

M1.1.3 Overhaul of Motor Operated Damper VAC-1-MOD-8

a. Inspection Scope (62707)

The inspector reviewed the following work documents and procedures:

- WO R0159930: VAC-1-MOD-8 Damper Overhaul

- MP E-53.10M, Rev 10A, "Limitorque SMB-00 and SB-00 Valve Operator Maintenance"

- MP E-53.10A, Rev 19, "Preventive Maintenance of Limitorque Motor Operators"

b. Observations and Findings

Technical Maintenance (TM) personnel were observed during the installation of the actuator worm shaft and the clutch tripper assembly for the Limitorque SMB-00 actuator that operates Damper VAC-1-MOD-8. This portion of the work was being repeated since, when attempting to operate the actuator following the initial overhaul, it could not be placed in the manual mode. Discussion with the cognizant engineer identified that this was the first time that the actuator had been overhauled since the plant had started operation.

Actuator Disassembly

During actuator disassembly it was noted that the tripper lever spacer had not been installed. Correspondence referred to by the licensee from Limitorque indicated that the installation of the tripper lever spacer was not required as long as proper alignment could be obtained. In addition, during the actuator disassembly, an additional spacer was noted to have been previously installed that was not specified in the actuator diagrams. The additional spacer had been installed between the inner tripper cam and the bearing spacer. After technicians consulted with engineering, neither the tripper lever spacer nor the additional spacer were used in the reassembly of the actuator.

Mixed Lubricants

Upon disassembly, technicians noted a mixture of Beacon 325 and Mobil 28 grease in the actuator limit switch gearbox. Prior to reassembly, the gearbox was cleaned and greased using Mobil 28 lubricant. Discussion with the cognizant engineer indicated that the Beacon 325 grease had most likely been used by the valve manufacturer since it had not been utilized by the licensee. The mixing of grease by addition of Mobil 28 lubricant had most likely been performed during the licensee's prior maintenance activities. The combination of the two types of lubricants, although not desired, was evaluated as not to have impacted valve operability since the grease was noted to remain in a liquid state. Although the combination of the greases resulted in a thinner more liquid substance, it was considered acceptable since the actuator was not required to be environmentally qualified.

VAC-1-MOD-8 was one of two actuators that had not been overhauled since actuator maintenance was turned over to the maintenance department prior to the plant commencing commercial operation. The cognizant engineer indicated that all other motor-operated valve and damper actuators had been overhauled by the maintenance department since that time, and the other nonoverhauled actuator had been satisfactorily inspected. Based upon the information provided, there does not appear to be a concern for the use of mixed grease in other actuators.

Clutch Tripper Assembly Installation

During installation of the worm shaft, the previously installed tripper cams and tripper fingers were inspected and noted to be worn. One of the tripper cams had been slightly deformed, apparently by contact with the tripper fingers. The tripper fingers were also noted to be rounded, thus reducing the contact area on the tripper adjustment arm. The inspector concluded that a more thorough inspection of the tripper fingers and tripper cams would have indicated the need for replacement during initial overhaul of the actuator. The cognizant engineer agreed that the parts should have been replaced. Following replacement of the tripper cams and the tripper lever assembly, the actuator was verified to satisfactorily operate in the hand mode.

c. Conclusions

The initial reassembly of VAC-1-MOD-8 without replacing the worn components is considered to be a poor maintenance work practice. In addition, the presence of mixed lubricants in the limit switch gearbox is indicative of a prior weakness in the licensee's motor-operated actuator maintenance activities.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

Selected surveillance tests required to be performed by the 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 the following surveillances:

- STP I-9-L922, Revision 2, Refueling Water Storage Tank 1-1 Level Channel LT-922 Calibration

- STP P-RHR-12, Revision 4, "Routine Surveillance Test of RHR Pump 1-2"

b. Observations and Findings

The inspectors found that the surveillance reviewed and/or observed were 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 also noted that test results were appropriately dispositioned.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) VIO 50-275/95016-02: NI audio count rate secured contrary to TS and procedural requirements. During operational tests of Nuclear Instrument Channels NI-31 and NI-32, TM personnel failed to follow procedures and, as a result, the audible count rate provided by these instruments was secured for approximately 4 hours during a period required by TS. Evaluating these procedural violations, the licensee determined that the first example was caused by personnel error in that the technicians, recognizing a discrepancy in a procedural step, did not obtain an on-the-spot-change (OTSC) to allow them to perform the step as desired. The second example was most likely caused by inadequate self-verification.

The licensee subsequently issued an OTSC to Procedure STP I-4A to: (1) allow the audio count rate setting on the instrument to be adjusted, as necessary, to produce a discernable change in the audio count rate, and (2) add an independent verification requirement during the audio count rate channel restoration. Additionally, the technicians were counseled on the need to process an OTSC when work being performed is not clearly specified by the applicable procedure. In addition, a general tailboard was conducted with TM personnel regarding the issues. Prior to Refueling Outage 2R7, Procedure STP I-4A was revised to enhance equipment turnover requirements contained in the "Return to Service" section. The inspector verified the completion of these corrective actions and reviewed the revised portions of Procedure STP I-4A. The licensee's corrective actions appeared to be reasonable and sufficient to preclude recurrence of the violation.

M8.2 (Closed) VIO 50-275/95016-03: failure to follow fire protection requirements for fire doors. During housekeeping activities in the Emergency Diesel Generator (EDG) 1-1 room, personnel blocked open a fire door without informing the fire protection specialist or the shift foreman. The licensee determined that the root cause of the violation was inadequate guidance in the governing procedure for fire system impairment. Specifically, Procedure OM8.ID2 did not clearly define when a fire door is considered blocked or impaired. Consequently, the personnel performing the housekeeping activities did not believe that the their activities impaired the function of the fire door. Based upon this finding, the licensee revised Procedure OM8.ID2 to specify when a fire door is considered impaired. The licensee also provided information to plant personnel on this event, including the definition of fire door impairment, in a March edition of the plant's Nuclear News letter. The inspector verified the completion of the licensee's corrective actions and determined that the actions were appropriate to preclude recurrence of the violation.

III. Engineering

E1 Conduct of Engineering

E1.1 4160V Vital AC System Review

a. Inspection Scope (71707, 37551)

The inspector conducted a review of the design, maintenance, and operation of the on-site 4160 V vital AC power system. This review included the following documents:

- Updated Final Safety Analysis Report (UFSAR), Chapter 8.3, Onsite Power Systems
- Design Criteria Memorandum S-63, 4160 V System
- DCP H-049059, Rev. 0, 4 kV Switchgear and Cable Spreading Room Air Flow
  DCP E-047079, Rev. 0, Setting of Secondary Undervoltage Relays
  DCP E-049237, Rev. 1, Tap Adjustment on the Standby Startup Transformer
  DCP C-043902, Rev. 1, 4 kV Switchgear Embedment Plate Welds
  DCP M-047098, Rev. 0, 4 kV Switchgear Door Fasteners
- TP TD-9607, Providing 125 VDC Power From SD21 to SD22 Vital Loads
- Quality Evaluations: Q0007408, Q0010602
- 4160 V System Surveillance Procedures
- 4160 V System Operating Procedures

The inspector also conducted a detailed system walkdown of the 4160 V vital switchgear and interviewed the system engineer. This review did not specifically address the electrical portions of the EDGs.

b. Observations and Findings

The 4160 V vital AC power system is designed to provide reliable power to various safety-related components. Major loads on the 4160 V buses include emergency core cooling system pumps, component cooling water pumps, auxiliary saltwater pumps, auxiliary feedwater pumps, and the 480 V vital buses. Normal power to the buses is fed from the unit main generator or 500 kV transmission system via the unit auxiliary transformer. Backup power is supplied from the offsite 230 kV

transmission system via the standby startup transformer and from the EDGs. Each of the three 4160 V buses has a dedicated EDG that will automatically supply power to the bus in the event of a loss of voltage.

Plant TS for the 4160 V system were found to be consistent with the requirements of the Updated Final Safety Analysis Report (UFSAR). Procedures STP M-13A, STP M-13F, STP M-13G, STP M-13H, and STP M-75 adequately addressed the requirements of TS 4.8.1.1.1. Operating procedures were also found to be of sufficient detail to provide adequate guidance to operators in the startup, operation, and shutdown of the 4160 V system.

Since the start of commercial operation, the licensee has identified several design deficiencies in the 4160 V system and has proactively pursued their resolution. Examples include the replacement of the underrated 250 MVA GE Magne-Blast circuit breakers with 350 MVA SF_6 breakers, improvement of the seismic withstand capability of the 4160 V switchgear, and a modification to provide annunciation when the breaker closing spring fails to charge. During and following the installation of the new 350 MVA SF_6 breakers, several problems were revealed in the interface between the new breaker and the stationary auxiliary switch. These problems were aggressively pursued by the licensee with corrective actions that were both thorough and technically justified. A review of the associated design change packages for these modifications found that they were technically complete and that they adequately addressed any impact on the licensing basis of the system.

The inspector walked down the 4160 V switchgear in both units, including the ventilation lineup for room cooling. No extraneous materials affecting fire loading or seismic interaction were noted in any of the switchgear rooms. Switchgear deficiencies were properly identified and tagged. No deficiencies were noted that had not already been identified.

The system engineer was very knowledgeable on both system requirements and component design basis. He has been assigned to the 4160 V system for 6 years and is also the system engineer for the electrical portions of the EDGs. He had played an active role in each of the design changes to the system and was able to discuss the technical details of each. The system engineer walked down the system on a nominal monthly basis and maintained a quarterly system report on the status of design changes and resolution of deficiencies.

c. Conclusions

Current design and testing of the 4160 V vital AC power system is in conformance with the UFSAR and plant TS. The licensee has been proactive in identifying and correcting degraded conditions and system design deficiencies. Engineering, operations, and maintenance staffs have demonstrated the ability to coordinate efforts in the implementation of design changes and problem resolution.

E1.2 CVCS System Review

a. Inspection Scope (71707, 37551)

The inspectors reviewed documentation related to the CVCS, including:

- ARs: A0403181,A0394714, A0394405, A0398021, A0384085, A0384084, A0402973, A0410262, A0122862, A0314487, A0326480 and A0393237
- System Engineer Quarterly Reports
- Quality Evaluations Q0011894, Q0011791, Q0011639
- Nonconformance Report N0001955
- UFSAR Section 6.3, Emergency Core Cooling System; Section 9.3.4, Chemical and Volume Control System; and Table 6.2-39, Containment Piping Penetrations and Valving
- Plant Staff Review Committee (PSRC) TS Interpretation 96-08, Revision 0
- Surveillance Procedures STP M-54, Verification of RCP Seal Injection Flows By Resistance Measurements, Revision 18
- Temporary Modification/Plant Jumpers 94-44, 96-14, 96-28

The inspectors walked down portions of the system in both Units 1 and 2 and observed equipment operation, valve alignments, AR tags, and overall material condition of the equipment.

b. Observations and Findings

Selected ARs on the CVCS system were reviewed with the system engineer. The system engineer was knowledgeable of the status of these items and of the equipment history for his system. At the time of the review there was a total, for both Units 1 and 2, of approximately 300 outstanding ARs on the CVCS system, some dating back to 1989. Of the older ARs that were reviewed, a majority had been assigned as low priority items. Based on a limited sample, the prioritization of these ARs appeared appropriate due to the minor nature of the problems.

CVCS TS Requirements

Applicable TS were reviewed. During the review, the inspector noted that the PSRC had approved an interpretation of TS 3.4.6.2 e. that addressed

acceptable CONTROLLED LEAKAGE. The TS states that the reactor coolant system leakage shall be limited to 40 gpm CONTROLLED LEAKAGE at a system pressure of 2235 ± 20 psig.

The licensee's interpretation of the TS is that its purpose is to ensure that the CONTROLLED LEAKAGE is less than 40 gpm under postloss of cooiant accident conditions. The basis for the interpretation is supported by the TS bases, which indicate that the limit for controlled leakage is to ensure the safety injection flow will be greater than that assumed in the analysis in the event of a loss of coolant accident. Therefore, in situations where the CONTROLLED LEAKAGE is calculated to be less than 40 gpm, with charging aligned in the post accident mode, but flow measurements taken at 2235 psig indicate the flow rate is greater than 40 gpm, the licensee considers the test results acceptable. The inspector questioned the validity of the TS interpretation in that it appeared to deviate from the surveillance as currently written in the TS to ensure CONTROLLED LEAKAGE at 2235 ± 20 psig is less than 40 gpm. The most recent surveillance results for Units 1 and 2 indicated that the CONTROLLED LEAKAGE was less than 40 gpm. The evaluation of the licensee's interpretation of TS 3.4.6.2.e is being considered as an inspection followup item (IFI 50-275/323 96020-02).

CVCS System Walkdown

A walkdown of portions of the system was performed for both Units 1 and 2 and found the valve alignment to be correct, including accessible containment isolation valves. The operating pumps had adequate oil levels and cooling flows. The overall material condition of the equipment was good, with the exception of dry boric acid indications on numerous valves. Although most of these valves were tagged and tracked on a master AR, 12 valves were identified as having indication of dry boric acid leaks and were neither tagged nor included in the master AR. After this concern was raised with the licensee, the valves with evidence of boric acid leakage were added to the master AR.

Various documents were reviewed that identified the containment isolation valves in the CVCS system. The documents were consistent with the plant drawings and with each other, with the exception of two minor editorial errors in AD13.DC1, Attachment 7.10, "Containment Isolation Valves." These deficiencies were identified to the licensee for correction.

The jumper log was reviewed for temporary modifications to the CVCS system. Three temporary pressure instruments were noted to have been installed to improve monitoring of the CONTROLLED LEAKAGE during surveillance testing and to allow monitoring of the differential pressure across the reactor coolant letdown filters 1-2 and 2-2 when Filters 1-1 or 2-1 are isolated. The temporary letdown filter differential pressure gauges were installed in April 1994 and May 1996 for Units 1 and 2, respectively. The

use of a temporary jumper for over 2 years, in lieu of installing a design change, appeared to be a protracted length of time to utilize a temporary jumper. However, the inspector noted that design changes had been initiated, and the licensee had scheduled replacement of the temporary gauges with permanent installations within the next month.

Centrifugal Charging (CC) Pump and Positive Displacement (PD) Pump Issues

The licensee has taken several positive steps to ensure that the material condition of the CC pumps and the PD pumps is properly maintained, as demonstrated by the following actions:

a. The licensee had previously identified erosion of the restricting orifice in the recirculation flow path for each CC pump. Following identification, recirculation line flow testing was performed in order to determine if the pump recirculation flow rates were within allowable limits. Testing indicated that CC pump recirculation flow had increased but not to the point of causing flow rates to be outside of allowable limits. The testing appeared to adequately assess the impact of the degradation on flow rates for the existing conditions. The licensee has scheduled replacement of the orifices during the next refueling outage for each unit.

b. In response to industry problems described in NRC Information Notices 94-76: Recent Failures of Charging/Safety Injection Pump Shafts, 80-38: Cracking in Charging Pump Cladding; and 94-63: Boric Acid Corrosion of Charging Pump Casing Caused By Cladding Cracks, the licensee has replaced one cc pump on each unit with pumps that have stainless steel casings and internal assemblies.

c. Following identification of an indication on a pipe weld associated with the Unit 1 PD pump, the licensee attempted to perform an ultrasonic examination of the Unit 2 piping. Due to the inaccessibility of the weld for ultrasonic examination and the potential concern for a similar problem with the Unit 2 weld, the licensee replaced the socket weld in question with a butt weld as a precautionary measure.

c. Conclusions

The licensee's initiatives to improve system reliability were noteworthy. Significant effort had been put forth to improve the reliability of the system and facilitate running the PD pump to provide normal charging flow and limit operation of the CC pumps during normal plant operation. In addition, the system engineer was very knowledgeable of the system and the status of outstanding deficiencies. One weakness was noted in that a number of boric acid leaks were noted that had not been identified by the licensee and entered into their tracking and evaluation AR.

ICES Engineering Support of Facilities and Equipment

Review of UFSAR Commitments

A recent discovery of a licensee operating their facility in a manner contrary to the UFSAR description highlighted the need for a special focused review that compares plant practices, procedures, and/or parameters to the UFSAR description. During the inspection period, the inspectors reviewed the applicable sections of the UFSAR that related to the inspection areas discussed in this report. There were no inconsistencies noted between the wording of the UFSAR and the plant practices, procedures, and/or parameters observed by the inspectors.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) LER 50-275/94006-00: CC pump outside of design basis due to throttling of component cooling water (CCW) to subcomponents. The LER was written to report the licensee's determination that CCW flow to the CC pumps had been reduced by throttling to the point where the CCW flow rate may not have been high enough to adequately cool the CC pump subcomponent heat exchangers in the event of an accident to maintain postaccident CC pump operability.

The original LER, which reported only the licensee's discovery of the problem and immediate corrective actions, was submitted to the NRC on July 22, 1994. Revision 1 to the LER was submitted on June 15, 1995, and addressed the root cause, safety significance and corrective actions for the event. As a part of the corrective actions, the licensee performed testing that confirmed the adequacy of the existing nonthrottled CCW flow to cool the CC pump subcomponents. Although the CCW flow to CC pump heat exchangers remains less than that recommended by the vendor, the licensee has consulted with the vendor and obtained concurrence that the existing flow is sufficient to ensure the CC pump remains operable during a design basis accident. Final review of this issue will be performed prior to the closure of Revision 1 of the LER.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 Control of the Moveable Incore Detector System (MIDS) Key

a. Inspection Scope (71750)

In conjunction with a maintenance observation associated with the replacement of the Unit 2 MIDS Detector A, the inspector reviewed Procedure RCP D-230, Revision 9, "Radiological Control for Containment Entry" and AR A0394318 to evaluate the radiological controls requirements for the work.

b. Observations and Findings

Step 5.3.1.b.2 of Procedure RCP D-230 requires the MIDS keys to be in the possession of the Radiation Protection (RP) Foreman during all containment entries. As these keys are controlled by the unit shift foreman (SFM), the RP Foreman signs for and takes possession of the keys prior to a containment entry. AR A0394318 was initiated in February 1996 to document the fact that another key controlled by the SFM would also operate the MIDS power switch. The SFM who initiated the AR recommended that the MIDS key lock be changed to preclude the possibility of inadvertent operation of the MIDS drives while personnel were in the containment. As an immediate corrective action, the SFM would subsequently issue both keys to the RP Foreman. However, this action was neither proceduralized nor documented in the AR, and no further action was taken at that time to determine if the MIDS key was duplicated in other applications.

On September 12, 1996, the SFM informed the RP department that there were two more keys, controlled by the control room assistant, that would fit in the MIDS power switch lock. The RP General Foreman updated AR A0394318 to reflect this discovery; however, he did not document any corrective actions taken in response. On September 19, the inspector questioned the shift supervisor on what actions had been taken to ensure adequate control of the MIDS key during containment entries. Operations personnel determined that the control room assistant's keys were initially turned over to the SFM and then were subsequently removed from the SFM's key locker and stored as spares. Following the inspector's inquiry, the licensee discovered that the Unit 2 MIDS power switch key was also duplicated for use in other applications.

On September 24, in response to the number of duplicate keys identified for the MIDS power switch, Procedure RCP D-230 was updated to require an administrative tagout to provide adequate control of the MIDS during containment entries. No occurrences were identified where the MIDS had been operated during personnel entries into the containment.

c. Conclusions

The scope of the licensee's investigation and corrective actions in response to the duplication of the MIDS power switch key was too narrow to adequately bound the problem and ensure positive controls over the MIDS during containment entries. This was considered a weakness in the licensee's corrective action program.

R3 RP&C Procedures and Documentation

R3.1 Primary Coolant Sample Procedure

a. Inspection Scope (71750)

On September 27, the inspector observed the drawing of a reactor coolant system daily sample at the Unit 1 primary sample sink. Procedure CAP E-1, Revision 11B, "Sampling of Primary Systems," was also reviewed.

b. Observations and Findings

The chemistry technician was knowledgeable of the procedure and demonstrated proper radiological controls while working in the sample sink. A sufficient volume of coolant was purged through the sample line to ensure a representative sample was drawn.

The technician utilized an "issued-for-use" copy of Procedure CAP E-1, located in the primary sample room, to draw the reactor coolant sample. An "issued-for-use" stamp was affixed to the cover page of the procedure, indicating that it had been verified to be current; however, the copy was that of Revision 11A and not 11B. The latest revision, 11B, had been implemented in April 1996. A comparison between the two revisions found only minor administrative changes that did not impact the intent of the procedure.

Procedure AD2.ID1, Revision 4, "Procedure Use and Adherence," requires "issued-for-use" procedures to be verified current. Step 5.1.1.a states that, when a procedure is taken from a controlled manual and is to be used in the performance of work, the cognizant supervisor or designated individual shall verify that it is the current revision immediately prior to starting work. Step 5.1.1.c states that "procedures in use longer than the "issued-for-use" interval shall be verified to be the current revision . . ." Both the verifier and the technician failed to identify and update the superseded revision of Procedure CAP E-1 in the Unit 1 primary sample room. The failure to verify and update "issued-for-use" copies of controlled procedures was also documented in NRC Inspection Report No. 50-275/96-06; 50-323/96-06 with regard to the axial flux difference limits curve posted at the control operator's station.

c. Conclusions

The failure to verify and update the "issued-for-use" copy of Procedure CAP E-1 in the primary sample room was determined to be a violation of Procedure AD2.ID1 (VIO 50-275/96020-03).

R8 Miscellaneous RP&C Issues

R8.1 Housekeeping in Radiological Work Areas

a. Inspection Scope (71750)

The inspector toured the radiologically controlled area and observed the housekeeping and radiological work practices in areas established to accomplish maintenance on contaminated equipment.

b. Observations and Findings

The inspector observed that the radiological conditions in the 140 foot elevation of the fuel handling building had improved from that noted in NRC Inspection Report 50-275/96-03; 50-323/96-03. Tools and protective clothing were noted to be appropriately stored. Radiological boundaries were properly maintained in that the areas were clearly marked and posted and there were no items laid across the boundaries. General cleanliness of the area had also been improved and the amount of radioactive material that was being stored in the area had been significantly reduced.

c. Conclusions

The general housekeeping and radiological work practices in the fuel handling building contaminated work areas had significantly improved.

F1 Conduct of Fire Protection Activities

F1.1 Fire Watch Performance

a. Inspection Scope (71750)

On September 16, during a tour of the Unit 1 turbine building, the inspector observed maintenance personnel performing welding on the service air supply to the oily water separator. The requirements of the welding and open flame permit associated with the work were evaluated to determine whether they were being met.

b. Observations and Findings

The Welding and Open Flame Permit had been properly approved by a Fire Protection Specialist, and required that a trained fire watch be stationed during the work with a portable fire extinguisher in the work area. Although several personnel were in the room housing the oily water separator during the welding, a fire watch was not readily identifiable. Additionally, the required portable fire extinguisher was outside the room, under the temporary work bench that had been set up for the job.

The portable extinguisher was, in fact, further from the work site than a permanent fire extinguisher that was mounted on the exterior wall of the room. The inspector noted that the permit had not been initialed to indicate that these requirements had been met prior to the start of work.

Procedure OM8.ID1, Revision 4, "Fire Loss Prevention," delineates the fire protection requirements during welding activities. Section 3.3.4 of Attachment 7.1 to Procedure OM8.ID1 states that the fire watch is responsible for being readily identifiable (e.g., wearing a red vest or readily identifiable hard hat, arm band, etc.). Section 4.3.7 states that "prior to the start of actual welding or open flame work the worker or the fire watch shall initial the right side of the [Welding and Open Flame Permit] after inspecting the area and confirming each of the requirements designated have been completed." The failure of the maintenance personnel to properly designate a fire watch and to verify that the requirements of the Welding and Open Flame Permit had been met prior to commencing work was considered a violation of Procedure OM8.ID1.

c. Conclusions

The failure to properly designate and identify a fire watch during welding activities and the failure to verify and initial that the fire protection requirements had been met prior to commencing work was a violation of Procedure OM8.ID1. This failure constitutes a violation of minor significance and is being treated as a noncited violation consistent with Section IV of the NRC Enforcement Policy (NCV 50-275/96020-04). The placement of the portable fire extinguisher outside the room where the welding was being performed was considered a poor work practice.

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 October 2, 1996. 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

R. P. Powers, Manager, Vice President DCPP and Plant Manager
J. R. Becker, Director, Operations
D. K. Cosgrove, Supervisor, Safety and Fire Protection
S. R. Fridley, Manager, Outage Services
W. A. Ginter, Engineer, Nuclear Steam Supply Systems Engineering
T. L. Grebel, Director, Regulatory Services
J. A. Hays, Director, Chemistry and Environmental Services
J. R. Hinds, Director, Nuclear Quality Services
S. C. Ketelsen, Supervisor, Nuclear Quality Services
D. B. Miklush, Manager, Engineering Services
J. E. Molden, Manager, Operations Services
M. N. Norem, Director, Mechanical Maintenance
D. A. Vosburg, Director, Nuclear Steam Supply Systems Engineering

INSPECTION PROCEDURES USED

IP 37551: Onsite Engineering
IP 61726: Surveillance Observations
IP 62707: Maintenance Observations
IP 71707: Plant Operations
IP 71750: Plant Support
IP 92901: Followup - Plant Operations
IP 92902: Followup - Maintenance
IP 92903: Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-275/96020-01	NCV	Inadequate work instructions and procedures for installation of a freeze seal on SFP piping
50-275/96020-02 50-323/96020-02	IFI	PSRC interpretation of TS 3.4.6.2 regarding controlled leakage
50-275/96020-03	VIO	Failure to use the latest revision of CAP E-1 primary sample procedure
50-275/96020-04	NCV	Failure to follow fire watch procedures

Closed

50-275/96020-01	NCV	Inadequate work instructions and procedures for installation of a freeze seal on SFP piping
50-323/96002-01	VIO	Failure to perform required monthly channel checks of in-core thermocouples
50-275/95006-01	VIO	Four examples of failure to follow procedure
50-275/95016-02	VIO	Nuclear Instrument audio count rate secured when required by TS
50-275/95016-03	VIO	Fire door blocked open without authorization
50-275/96020-04	NCV	Failure to follow fire watch procedures
50-275/84050-00	LER	Failure to meet TS 3.3.3.6 surveillance requirements
50-275/94006-00	LER	CC pump outside of design basis due to throttling of component cooling water to subcomponents

LIST OF ACRONYMS USED

AR	action request
CC	centrifugal charging
CCW	component cooling water
CVCS	chemical and volume control system
EDG	emergency diesel generator
LER	licensee event report
MIDS	moveable incore detector system
OTSC	on the spot change
PD	positive displacement
PDR	public document room
PPC	plant process computer
PSRC	Plant Staff Review Committee
RHR	residual heat removal
RP	radiation protection
SFM	shift foreman
SFP	spent fuel pool
TM	technical maintenance
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
WO	work order