ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Licensee:	Pacific Gas and Electric Company
Facility:	Diablo Canyon Nuclear Power Plant, Units 1 and 2
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Inspector(s):	D. L. Proulx, Senior Resident Inspector D. B. Allen, Resident Inspector D. G. Acker, Project Inspector T. R. Meadows, Licensing Examiner W. M. McNeill, Reactor Inspector B. J. Olson, Project Inspector
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Attachment :	Supplemental Information

EXECUTIVE SUMMARY

Diablo Canyon Nuclear Power Plant, Units 1 and 2
NRC Inspection Report 50-275/98-07; 50-323/98-07

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection.

Operations

- Shutdown and startup evolutions were conducted in a professional manner, in accordance with procedures and a focus on safety (Section O1.1).

- A noncited violation, per Section VII.B.I of the NRC Enforcement Policy, was identified for failure to provide a procedure appropriate to the circumstances for switching of power supplies between the units. The switching of the power supply without clearly understanding the outcome resulted in unexpected alarms, loss of power to equipment required by Technical Specifications, and unnecessary disruption in both control rooms. The immediate response of the Unit 1 control room operators was very good, with timely and appropriate response to each alarm (Section O1.2).

- A violation was identified for failure to provide a midloop procedure appropriate to the circumstances that required proper stowage of a nonseismically qualified hoist. The hoist was left in an unstowed condition above the operating residual heat removal pump during a reduced inventory condition (Section O1.3).

- Several significant operator evolutions were performed well. Shutdown and startup evolutions were conducted well, in a professional manner, in accordance with procedures, and with a focus on safety. Licensee preparations and implementation, including the operations pre-evolution briefings for early midloop operations, were conservative and reflected a focus on safety. The reflood of the emergency core cooling systems evolution was well coordinated and controlled, with each participant aware of their responsibilities. The pre-evolution briefing was comprehensive, with emphasis on safe, cautious performance and the necessity for good communications (Sections 01.3 and O1.4).

- A violation was identified for failure to restore the "High Flux at Shutdown" annunciator when the required number of fuel assemblies was installed in the core. The responsibility to perform the actions was not clearly assigned prior to the evolution (Section O1.5).

- The control of refueling activities lacked clear procedural guidance and management expectations. The lack of procedural guidance for performing signal-to-noise ratio calculation, the lack of acceptance criteria in the procedure for fuel assembly clearances, and the confusing procedure format were weaknesses in the procedure. The method used to calculate inverse count rate ratio and the method used to perform

the post core load verification were inconsistent with the methods described in the procedure. The lack of separate signatures in the controlled copy of the procedure for verifying that the signal-to-noise ratio was greater than two was an example of poor documentation of procedurally required activities (Section O1.5).

- A violation was identified for several examples of failure to properly implement the clearance procedure. However, the number and significance of clearance errors in Refueling Outage 2R8 were improved from the previous outage. Several significant errors were not found or prevented by the clearance process and resulted in the potential for work to be performed without the required isolation from sources of energy to allow safe work (Section O1.6).

- A violation was identified for failure to translate the design of the reactor vessel refueling level indication system into abnormal operating procedures. The licensee exhibited good attention to detail in identifying this issue during simulator training. Documented corrective actions at the end of the inspection period for this violation failed to address deficiencies in the procedure preparation and approval process (Section O3.1).

- The training provided for Unit 2 outage preparation was implemented well and provided valuable lessons learned and necessary procedural changes. The inspectors noted, in particular, that the simulator training was professional, well executed, and identified a vulnerability in the abnormal operating procedures (Section O5.1).

<u>Maintenance</u>

- Maintenance personnel did not exercise appropriate care during penetration seal work and stepped on a valve, that when repositioned, challenged operators by causing a leak in the chemical and volume control system (Section O2.1).

- A number of maintenance activities were observed and were performed in accordance with the procedural requirements. Good coordination between technical maintenance, mechanical maintenance, and radiation protection was observed in performing several maintenance tasks concurrently on the containment spray pump, thereby reducing the time the pump was inoperable due to maintenance (Section M1.1).

- The inspectors observed a number of surveillance tests and found that the surveillances observed were performed in a cautious manner with self-checking and proper communications employed. The procedures governing the surveillance tests were technically adequate and personnel performing the surveillances demonstrated an adequate level of knowledge. The inspectors noted that test results appeared to have been appropriately dispositioned (Section M1.2).

- The containment cleanup and closeout activities were appropriately controlled, and the material condition of containment areas was satisfactory for restart of Unit 2 (Section M1.3).

- The license's approach to the inspection of part length control rod drive mechanism welds was sound and aggressive. The inspectors found the ultrasonic testing showed the seven motor tubes' upper and lower transition welds were free of the type of defect found at Prairie Island on the G-9 motor tube (Section M2.1).

- A noncited violation was identified for failure to provide a procedure appropriate to the circumstances for ground buggy installation. The improper ground buggy installation had the potential to have caused significant damage to safety-related equipment and injure workers (Section M4.1).

- The inspectors concluded that the corrective actions for Violation 50-275;323/96014-03 were sufficiently directed towards ensuring that control board action request stickers were removed when the work was complete, but did not appear to fully address the need to closely control these deficiency tags. The inspectors found six additional deficiencies concerning control board action requests. Therefore, the licensee's programs to ensure that the control board action requests stickers reflected the licensee's tracking list and the up-to-date plant configuration warranted further licensee attention (Section M8.1).

<u>Engineering</u>

- The inspectors concluded that the design change package and associated safety evaluation for replacement of the Unit 2 recirculation sump screens was comprehensive, and the conclusions were reasonable. The design change was effective in improving the containment sump's ability to screen out debris that could block safety injection flow paths (Section E2.1).

<u>Plant Support</u>

- Licensee management's efforts to keep exposures as low as reasonably achievable during Refueling Outage 2R8 appeared to be successful in that total outage exposure was improved from previous outages. The licensee's cleanup of the reactor coolant system following shutdown of Unit 2, and the use of mock-up training for several outage tasks contributed to the lower exposure (Section R1.1).

Report Details

Summary of Plant Status

Unit 1 began this inspection period at 100 percent power. On March 21 1998, reactor power was reduced to 50 percent to conduct main feedwater pump control and stop valve testing, and Unit 1 was returned to 100 percent power later that day. Unit 1 continued to operate at essentially 100 percent power until the end of this inspection period.

Unit 2 began this inspection period in Mode 4 (Hot Shutdown) for Refueling Outage 2R8. On March 24, on completion of outage activities as of that point, Unit 2 entered Mode 2 (Startup). On March 25, Unit 2 entered Mode 1 (Power Operation). Later on March 25, the turbine tripped prior to synchronizing to the grid, and Unit 2 was returned to Mode 3 (Hot Standby) to investigate a turbine lube oil system problem. After completing repairs in the lube oil system, the reactor was returned to Mode 2 on March 27 and entered Mode 1 later that day. On March 28, Unit 2 was synchronized to the grid, ending Refueling Outage 2R8. Unit 2 was at 30 percent power at the end of this inspection period.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors observed control room operations and toured the plant on a frequent basis, including frequent backshift inspections. In general, the performance of plant operators was professional and reflected a focus on safety. Operators continued to perform well, utilizing three way communications and self-checking techniques. Operator response to alarms were observed to be prompt and appropriate to the circumstances. Operations shift management were frequently present in the control room and were aware of plant conditions. All crew members interviewed by the inspectors were aware of plant conditions and system configurations. Limiting conditions for operations were properly entered when required. During this period the inspectors observed several crews for each unit, including backshifts. The inspectors noted increase operations department management presence in the control room during the Unit 2 shutdown. The inspectors observed portions of reactor startup activities between March 25-March 28. The inspectors noted that these startup activities were conducted in a professional manner, in accordance with procedures, and with a focus on safety.

O1.2 Transferring Power Supply For Instrument Distribution Panel PYNM

a. Inspection Scope (71707)

On February 10, the inspectors observed control room activities while systems and equipment with unit cross-ties were being realigned to Unit 1 prior to the Unit 2 refueling outage (2R8).

b. Observations and Findings

While transferring the power supply for Panel PYNM (a 208/120 volt instrument distribution panel) from Unit 2 to Unit 1, a number of unexpected alarms were received in both Unit 1 and Unit 2 control rooms. The operator performing the switching had not expected the alarms and had informed the Unit 2 shift foreman that there would be no alarms in the control room during the switching. The drawing the operator had referenced did not have sufficient information to identify the effects of the transfer. Had the correct drawings been used, each alarm would have been anticipated or actions taken to prevent the alarms from occurring. Action Request A0452798 was initiated to document this occurrence

The alarms received included: fire detection, component cooling surge tank pressure high and low, seismic trip undervoltage, post accident sample room radiation monitor failure, and low flow to Radiation Monitors RM-11 and RM-12. The seismic trip system provided an input to reactor trip at Diablo Canyon. Radiation Monitor RM-11 was the containment air particulate monitor and RM-12 was the containment radioactive gas monitor. Technical Specification 3.4.6.1, "Reactor Coolant System Leakage Detection Systems" requires that with both Radiation Monitors RM-11 and RM-12 inoperable, the containment fan cooler collection monitoring system and the containment structure sumps and reactor cavity sump level and flow monitoring systems must be operable.

The inspector observed the Unit 1 control room operators respond to these alarms. The control operator immediately evaluated and prioritized the alarms by importance. The Unit 1 shift foreman questioned the Unit 2 shift foreman and learned that the alarms could have been caused by the switching of power to Panel PYNM, and relayed this information to the control operator. Follow up actions to return equipment and alarms to normal was appropriate and timely. Both units entered Technical Specifications action statements for loss of Radiation Monitors RM-11 and RM-12.

The switching of power was performed under the direction of an Operations Section Policy D-4, "Preoutage System Alignments For Systems With Unit Crossties," Revision 1. There was not a procedure available to perform this switching. For six other activities addressed by the policy, specific procedures were referenced to perform the transfer. The policy document was not a procedure and provided no guidance as to the expected results. The procedures referenced in the policy did not provide the necessary directions.

As corrective actions, the licensee committed to revise Operations Section Policy D-4 and the applicable procedures referenced in the policy such that clear procedural guidance would be provided for future power switching operations. In addition, the licensee placed a permanent operator aide near Panel PYNM to alert the operators of the multiple inputs to the panel.

Failure to provide a procedure appropriate to the circumstances for switching power supplies is a violation of 10 CFR Part 50, Appendix B, Criterion V. However, this

nonwillful, self-revealing, and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-275;323/98007-01).

c. Conclusions

A noncited violation was identified for failure to provide a procedure appropriate to the circumstances for switching of power supplies between the units. The switching of the power supply without clearly understanding the outcome resulted in unexpected alarms, loss of power to equipment required by Technical Specifications, and unnecessary disruption in both control rooms. The immediate response of the Unit 1 control room operators was very good, with timely and appropriate response to each alarm.

O1.3 Midloop Operations (Unit 2)

a. Inspection Scope (71707)

The inspectors verified the prerequisites and witnessed the performance of midloop operations, when Unit 2 was in a reduced inventory condition. In addition, the inspectors witnessed training in preparation for this evolution. The inspectors performed these inspections on February 19 and March 9, 1998, each time the licensee entered the reduced inventory condition.

b. Observations and Findings

Early Midloop

On February 19, 1998, prior to entering midloop operations early in the outage with a high decay heat load, the inspectors verified the prerequisites for the evolution. This included plant tours to verify that personnel and equipment were staged to permit venting of the residual heat removal pumps in the event of pump vortexing. The inspectors also ensured that no ongoing evolutions that could affect midloop operations were in progress. The inspectors concluded that the prerequisites for entering hot midloop were satisfied.

The inspectors observed the operations briefing for entering midloop. The briefing was organized, detailed, and focused on safety. Licensee management was present and emphasized expectations with respect to safe operations.

On February 19, with Unit 2 in Mode 5, the inspectors observed the control room crew drain the reactor vessel water level down to 107 feet, midlevel of the reactor vessel hotlegs, in preparation for steam generator maintenance. The evolution was conducted without incident, in a very professional manner. The inspectors monitored the following special reactor vessel refueling level indications and other special instrumentation:

- Reactor Vessel Refueling Level Indicating System wide and narrow range
- LT-954 & LT-953 normal level indicators

- LI-561 normal level indicator
- Chart recorders wide and narrow range indicators
- Chart recorders for pressure relief tank pressure and volume control tank level

The inspectors compared these indications throughout the evolution and they appeared to be accurate. All ranges and trends were consistent. The inspectors also witnessed the maintenance of midloop conditions and the refill following completion of installation of the steam generator nozzle dams. These activities were also conducted satisfactorily.

Late Midloop

On March 10, 1997, following completion of the steam generator tube inspection and tube plugging, the licensee reentered midloop operations to remove the steam generator nozzle dams. The inspectors reverified the prerequisites were met. The inspectors provided continuous coverage during the entire evolution and ensured that the precautions, limitations, and prerequisites remained in effect. The inspectors observed the control room crew drain the reactor vessel down to 107 feet, which was completed without incident. The inspectors monitored instrumentation, which appeared to be tracking satisfactorily.

With the reactor at midloop conditions, the inspectors toured the facility to assess the material condition of the systems used for this evolution. In the room for residual heat removal Pump 2-2, the operating pump, the inspectors noted that the overhead trolley hoist for the pump was not in its normally stowed position and was above the residual heat removal pump and piping. The inspectors were concerned because the overhead trolley hoist in the pump room was not seismically qualified, and could possibly impact the operating residual heat removal pump, instrument tubing, or system piping while in a reduced inventory condition. The inspectors contacted the shift supervisor, who directed maintenance personnel to stow and lock the hoist in its proper position.

Engineering determined that the failure to stow the overhead hoist was not a significant concern because it was unlikely for a seismic event to result in the hoist chains or hoist to impact residual heat removal components in such a way that the system could not perform its intended safety function. The licensee did not document the inspectors' concern on an action request until March 25, after the inspector continued to question the licensee on the cause and safety significance of this problem.

Procedure MP M-10.2, "Residual Heat Removal Pump Motor and Impeller Handling," Revision 5, Section 7.6.9, required the hoist and trolleys in the residual heat removal pump rooms to be placed in their normally stowed positions with the chains secured in place following maintenance. Although the licensee believed that this procedure was properly implemented, the hoist was not in its proper location when required.

On March 25, 1998, the licensee began investigating this problem. The licensee noted that although Section 7.6.9 of Procedure MP M-10.2 was not signed off as completed,

maintenance personnel stated that they had stowed the hoist as required. The licensee believed that subsequent to mechanical maintenance personnel properly stowing the hoist, other personnel removed the restraining bracket from the chain and moved the hoist back over the pump. At the end of the inspection period, the licensee had not determined the cause of the improper hoist location.

The inspectors discussed this issue with licensee engineering personnel. The licensee noted that although licensee procedures required plant walkdowns to identify seismic concerns prior to operational mode changes, no such requirement existed for entry into midloop operations in the prerequisites of the midloop procedure. Procedure OP A:2-III, "Reactor Vessel - Draining to Half Loop/Half Loop Operations with Fuel in the Vessel," Revision 13, was not appropriate to the circumstances. Specifically, Procedure OP A:2-III did not provide for verification that no seismic concerns existed prior to entry into reduced inventory conditions. As a consequence, the hoist and trolley for residual heat removal Pump 2-2 was not in its seismically approved storage position and the chains were not in the storage racks when residual heat removal Pump 2-2 was being used for decay heat removal. The failure to provide a midloop procedure appropriate to the circumstances is a violation of 10 CFR Part 50, Appendix B, Criterion V (50-323/98007-02).

c. <u>Conclusions</u>

A violation was identified for failure to provide a midloop procedure appropriate to the circumstances that specified proper stowage of a nonseismically qualified hoist. The hoist was left in an unstowed condition above the operating residual heat removal pump during a reduced inventory condition. Licensee preparations and implementation, including the operations pre-evolution briefing for early midloop operations, were conservative and reflected a focus on safety.

O1.4 <u>Reflood of the Unit 2 Emergency Core Cooling Systems following Core Offload Outage Period</u>

a. <u>Inspection Scope (71707)</u>

On March 2, the inspectors observed control room activities while preparations were made to refill the residual heat removal and other emergency core cooling systems. On March 3, the inspectors observed operations venting emergency core cooling systems using Procedure OP A 2:VII, "Core Offload Window Systems Restoration," Revision 6.

b. <u>Observations and Findings</u>

The pretask briefing performed by the shift foreman covered the precautions and limitations, prerequisites, major steps, individual responsibilities and communications between organizations. The personnel attending the briefing included the operators assigned to perform the venting, operators in the control room, radiation protection personnel assisting with the venting, and technical maintenance personnel assigned to

backfill and place into service the reactor vessel refueling level instrumentation system. Special attention was given in the briefing to maintaining vessel level to avoid cavitation of the residual heat removal pump. The required positions of the newly installed throttle valves also received additional attention and were verified to be addressed by the procedure.

The control room operators closely monitored reactor vessel level indications and residual heat removal system parameters and were cognizant of the field activities during the fill and vent. Good two-part and three-part communications, self checking and peer checking was observed in the control room throughout the evolution. Venting of high point vents within controlled surface contamination areas was observed. The operators and radiation protection personnel worked well together to contain the vented liquid and demonstrated good radiological practices in entering and exiting controlled areas and monitoring for contamination.

c. Conclusions

The reflood of the emergency core cooling systems evolution was well coordinated and controlled, with the participants aware of their responsibilities. The pre-evolution briefing was comprehensive, with emphasis on safe, cautious performance and the necessity for good communications.

O1.5 Refueling Activities

a. Inspection Scope (60710)

On March 4 and 5, the inspectors observed refueling activities in the control room, fuel building and containment. These activities included handling and movement of the fuel assemblies from the spent fuel pool to the upender and from the upender to the final core location, control room monitoring of required parameters, and reactor engineering calculations of inverse count rate ratio and monitoring of fuel location for accountability requirements. The inspectors reviewed Operating Procedure OP B-8DS2, "Core Loading Sequence," Revision 20, which contained the procedural requirements for these activities.

b. Observations and Findings

The inspectors observed the refueling senior reactor operator directing the fuel handling operations in containment. He was observed to maintain good supervision over the activities, maintaining communications with the control room and the personnel in the fuel building, giving directions to the crane operator and other observers, verifying the correct core location as specified in the fuel movement tracking sheets, monitoring the load on the manipulator crane as the fuel assembly was raised or lowered, and confirming the proper indicating lights and Z-Z tape position. The fuel assemblies were inserted off index as described in the procedure. The inspectors observed the insertion of an assembly against the baffle and into a three-sided box, which resulted in a rapid

load fluctuation when the assembly bottom nozzle hung on the baffle. The senior reactor operator immediately stopped the loading and performed the steps required in the procedure to protect the assembly and achieve a successful installation. Reactor engineers were monitoring and video taping the fuel movement with cameras installed on the upper internals guide pins.

Fuel Handling

The inspectors observed the movement of the fuel assemblies in the fuel handling building. This activity was coordinated with the senior reactor operator to ensure an assembly was not raised out of the spent fuel racks until the upender was unloaded in the containment. The crane operators were observed to follow the procedural requirements, including observing limitations on crane motion, load, and speeds. The fuel assembly location and identification was independently verified against the fuel movement tracking sheets prior to being removed from the racks. Reactor engineers used an underwater camera to monitor and record the fuel assemblies as they were moved to the upender. The inspectors also observed that the foreign material exclusion area controls around the reactor cavity and the spent fuel pool were effective.

Control Room Activities to Support Fuel Load

The inspectors observed control room activities in support of fuel load. On March 5, while inspecting the switch lineup on the nuclear instrumentation panels, the inspectors noted that the "High Flux at Shutdown" alarm switch for source range Channel N-32 was in the blocked position, rather than the normal position required for the current step in the core loading procedure. This was immediately brought to the attention of the control operator and reactor engineer in the control room. Operations placed the switch in the normal position and documented on Action Request A0455551 that OP B-8DS2, Attachment 9.8, "Fuel Movement Tracking Sheet," required the "High Flux at Shutdown" alarm for N-32 be restored at Step 21B and this action had been overlooked. Approximately eight additional assemblies had been loaded after Step 21B and before the alarm was returned to normal. During this time, both channels of source range nuclear instruments were operable and monitored continuously by the control operator and reactor engineer. In addition, one channel of source range was audible in both the control room and containment, and the "High Flux at Shutdown" alarm for Channel N-31 was in normal.

The failure to restore the "High Flux at Shutdown" alarm at the specified step in the core loading procedure, Step 21B, is a violation of Technical Specification 6.8.1.a. (50-323/98007-03).

A contributing cause of the failure to restore the "High Flux at Shutdown" alarm appeared to be the format of the fuel movement tracking sheets in that: the applicable step contained three actions, the actions were to be performed by two different groups, the step did not have a separate step number (step numbering was used to designate the number of assemblies loaded), and the place for date, time, and initials was covered

by the text of the step. Another contributing cause was that the responsibility to perform the actions was not clearly assigned prior to the evolution.

Reactor Engineering Activities During Fuel Load

Step 21B of OP B-8DS2, Attachment 9.8, contained several actions, including verifying that the signal-to-noise ratio for source Range N-32 was greater than 2. The procedure provided no directions for determining signal to noise ratio. The completion of this activity was not documented in the procedure by a separate signature or initials, although the reactor engineer stated that he had performed the verification. The step was not signed until after the missed action of restoring the alarm was brought to the control operator's attention.

Step 6.17 of OP B-8DS2 stated, in part, that a calculated prediction of critical assemblies was required to be performed after the first 13 assemblies had been loaded adjacent to a detector. The reactor engineers used a computer program to calculate inverse count rate ratio (ICRR). This program also calculated the predicted ICRR if 12 additional assemblies were loaded based on extrapolation of the most recent ICRR data points. If this predicted ICRR were greater than zero, it indicated core load could continue. This method satisfied the intent of performing ICRR, but appeared inconsistent with the procedure in that the number of assemblies predicted to cause criticality was not calculated.

Step 6.25 of OP B-8DS2 stated that a fuel accountability audit and a foreign materials scan at the completion of core loading be performed, prior to placement of the upper internals in the vessel. This was to consist of either producing a video tape of the core and reviewing it for foreign objects, coupled with a verification of proper loading from the tape, or visual (camera and monitor) verification with two signatures to document the inspection. Inspection and verification of nozzle-to-nozzle and nozzle-to-baffle clearances was also required. Reactor engineering satisfied the fuel accountability requirement by reviewing the video tapes made in the spent fuel pool and in the reactor cavity during fuel loading. After the core was loaded, a video tape was made with a camera scan to inspect for foreign material and clearances between the assemblies and between the assemblies and the baffle plate. Although the methods used were technically adequate, they did not appear to be entirely consistent with the procedure (although vague) in that the tape of the core produced during the post core load scan was not used to verify proper loading, nor was a visual (camera and monitor) verification with two signatures to document the inspection performed. The procedure did not provide guidance nor acceptance criteria for the size of the gaps between the assemblies or between each assembly and the baffle. The inspectors also noted minor documentation discrepancies in the closeout of the paperwork.

c. Conclusions

A violation was identified for failure to restore the "High Flux at Shutdown" annunciator when the required number of fuel assemblies was installed in the core. The

responsibility to perform the actions was not clearly assigned prior to the evolution. The lack of formality in performance of control room refueling activities contributed to the failure to restore this alarm.

The control of safety activities lacked clear procedural guidance and clear management expectations. The lack of procedural guidance for performing signal to noise ratio calculation, the lack of acceptance criteria in the procedure for assembly to assembly and assembly to baffle clearances, and the confusing procedure format which contributed to the missed action of restoring the "High Flux at Shutdown," were weaknesses in the procedure. The method used to calculate inverse count rate ratio and the method used to perform the post core load verification appeared to be inconsistent with the methods described in the procedure, although the procedure was vague. The lack of separate signatures in the controlled copy of the procedure for verifying that the signal-to-noise ratio was greater than 2 was an example of poor documentation of procedurally required activities. The necessary coordination between the fuel handlers in containment, the fuel handlers in the refueling building, the reactor engineer, and the control operator's observation of plant conditions were otherwise performed well.

O1.6 Clearance Related Errors During 2R8 Refueling Outage

a. Inspection Scope (71707)

The inspectors reviewed the licensee's self-assessment of operations clearance performance during 2R8, and 27 action requests initiated to document errors related to clearances.

b. Observations and Findings

Operations performed a self-assessment of their performance during 2R8 and concluded there was a reduction in clearance errors and a reduction in the severity of these errors compared to 1R8. The licensee categorized the errors by significance. A level 1 (high significance) error was one that caused or could have caused personnel injury or equipment damage or all clearance process barriers were breached. A level 2 (moderate significance) error was one where some clearance process barriers were breached, but others prevented placing personnel or equipment in jeopardy. A level 3 (low significance) error was an inconsequential error related to some phase of the clearance process. In comparing the performance in 2R8 to 1R8, the licensee determined there were 3 high significance errors in 2R8 as compared to 8 in 1R8, there were 15 moderate errors in 2R8 compared to 33 in 1R8, and there were 45 low significance errors in 2R8 compared to 73 in 1R8. During 2R8, none of the errors resulted in personnel injury or equipment damage.

The licensee identified some recurring themes in the clearance errors. Most of these themes had minimal significance. Tags becoming unsecured or becoming illegible but none of the cleared equipment were operated as a result, and minor administrative

errors such as missing a set of initials were considered minor. Errors of most significance were generally caused by lack of proper self verification and independent verification.

The most significant clearance errors during 2R8 were related to clearing tags to perform testing and resulted in tags being removed from the wrong components or allowing work to be performed without a clearance, as follows:

- On February 25, while attempting to remove a clearance tag from 52-2F-02R to allow testing of containment fan cooling Unit 21, an operator mistakenly removed a tag from 52-2F-01R and attempted to close the breaker, which jammed and did not close. The operator recognized his mistake while showing the technical maintenance technicians the jammed breaker (AR A0454479):

- On February 26, while processing a partial removal of a clearance to perform testing, a senior control operator marked the desired return to service position on two incorrect clearance points. As a result, an operator removed tags and closed two incorrect breakers. The problem was identified and the breakers reopened and retagged (AR A0454548).

- On March 17, after testing of a control rod drive mechanism fan for proper rotation identified the fan motor leads needed to be swapped, the power supply breaker was opened and technical maintenance swapped the leads without clearing the breaker (AR A0457998).

Other clearance errors had the potential to be cause equipment damage or personnel injury, but were identified outside the clearance process. For example,

- On March 4, while removing tags following work, a senior control operator discovered a caution tag hanging over the fuse holders for fuses that had been removed for Valve RCS-2-PCV-474. This tag should have removed the fuses for Valve RCS-2-PCV-472 (AR A0455485). This error was not identified in the clearance process.

- On February 17, Clearance 57169 was hung prior to the proper plant conditions being established, isolating reactor coolant pump seal injection while the reactor coolant system pressure was approximately 350 psig. Operating procedure OP A-6:II, "Reactor Coolant Pumps - Shutdown and Clearing," Precautions and Limitations 5.2, specified that seal injection should remain in service as long as the reactor coolant system is pressurized to prevent introduction of crud from the reactor coolant system into the seals. The control room operators, responding to the loss of seal injection, contacted the operator in the field, and the seal injection flow was reestablished (AR A0455670). This clearance error was not found in the clearance process, but by alarms in the control room.

- On March 3, workers detected voltage on a cathodic protection circuit prior to performing work. The clearance, 56570, did not clear an alternate source of power from Unit 1's power supply 52-PJ18-1-33 (AR A0455394).

Other clearance errors were considered moderately significant by the licensee because not all barriers were broken, such as the errors were identified during walkdown by maintenance prior to starting work. These errors indicated that improvement in self-verification in the operations organization required improvement, because more than one barrier failed. For example:

- On February 16, during the maintenance walkdown, Breaker 52-23J-08 was found closed with a man-on-line tag hanging requiring the breaker to be open (AR A0453258).

- On March 6, during a clearance walkdown prior to starting work a man-on-line tag was found hanging on the wrong fuse. The tag was hanging on the fuse holder for IYFW21, but it should have been hung on the fuse holder for IYFW22, which was the cross feed power from the relay scheme for IYFW21 (AR A0455669).

Several other less significant errors existed that were identified by the independent verifier, thus indicating a failure of only one barrier.

The inspectors noted that, fortuitously, none of these examples of clearance errors resulted in personnel injury or equipment damage; however, the potential existed. Multiple failures of operations personnel to properly implement the clearance procedure is a violation of Technical Specification 6.8.1.a (50-275;323/98007-04).

c. Conclusions

A violation was identified for several examples of failure to properly implement the clearance procedure. Several significant errors were not found or prevented by the clearance process and resulted in work being performed or had the potential for work to be performed without the required isolation from sources of energy to allow safe work. However, the number and significance of clearance errors in Refueling Outage 2R8 were improved over previous outages.

O2 Operational Status of Facilities and Equipment

O2.1 Chemical and Volume Control System Leak

a. Inspection Scope (93702)

On February 13, 1998, the inspectors responded to the control room and observed operator response to a leak from the chemical and volume control system.

b. Observations and Findings

On February 13, 1998, while Unit 2 was operating at 100 percent power, the control operator identified that volume control tank level was dropping at a rate of approximately 12 gallons per minute. Operators implemented Procedure OP AP-17, "Charging Line Leak at Power," Revision 19, and isolated the flowpath to the deborating demineralizer, but volume control tank level continued to drop at the same rate. Approximately 30 minutes later a nuclear equipment operator reported that the miscellaneous equipment drain tank level was rising. The shift foreman directed that the nuclear equipment operators check the flow sight glasses in the auxiliary building, which determined that Flow Indicator FI-276 had flow. The nuclear equipment operators verified the position of check valves that drain through Flow Indicator FI-276. A nuclear equipment operator found that Valve CVCS-2-66 (the centrifugal charging pump recirculation drain line isolation valve) was approximately 1-1/2 turns open. The valve was closed, which stabilized volume control tank level. The inspectors monitored the operator response to the event and considered the operator resolution of the lowering of volume control tank level to be timely and thorough.

Licensee investigation revealed that maintenance personnel performed penetration seal work in the overhead area above Valve CVCS-2-66, and that the likely cause of the event was that the handwheel for Valve CVCS-2-66 was bumped while using a ladder adjacent to the valve. The inspectors noted that NRC Inspection Report 50-275; 323/97019 discussed an event in which the root cause was similar. In the previous event, maintenance personnel stepped on a main steam isolation valve limit switch when working overhead that resulted in a reactor trip and safety injection.

In response to this event, licensee management issued a shift order to the crews describing the event and directing operators to brief maintenance personnel on sensitive equipment areas while working overhead. In addition, licensee management issued a memorandum to all employees that discussed the event and the need to exercise vigilance in ensuring that care is taken not to disturb equipment while working in the plant. The inspectors considered the licensee actions appropriate.

c. Conclusions

Maintenance personnel did not exercise appropriate care during penetration seal work and stepped on a valve handwheel. This challenged operators by causing a leak in the chemical and volume control system. Operator response to decreasing volume control tank level was timely and thorough.

O3 Operations Procedures and Documentation

O3.1 Midloop Abnormal Operating Procedures

a. Inspection Scope (71707)

The inspectors evaluated the licensee's response to Action Request A0451605, which identified an inadequate pressure rating of the reactor vessel refueling level indicating system.

b. Observations and Findings

In 1993, the licensee upgraded their strategies for responding to abnormal events during shutdown conditions. This item was done with the help of the vendor. These strategies were incorporated into Procedure OP AP SD-0, "Loss of, or Inadequate Decay Heat Removal," Revision 3.

New strategies for combating casualties during shutdown and midloop operations included the use of natural circulation cooling and reflux cooling. Each of these methods required pressurizing the reactor coolant system to up to 400 psig and rejecting heat through the secondary side of the steam generators.

However, on January 23, 1998, during simulator training in preparation for midloop operations, the licensee noted that the reactor vessel refueling level indication system, predominantly constructed from tygon and nylobraid tubing, was designed for a normal operating pressure of 57 psig. The burst pressure of the reactor vessel refueling level indicating system was 150 psig for temperatures expected to be encountered during these abnormal procedures. Therefore, although Procedure OP AP SD-0 directed the operators to pressurize the reactor coolant system to 400 psig in the event of a loss of decay heat removal, the reactor vessel refueling level indication system was not designed nor qualified for this pressure. This condition existed for both Units 1 and 2, and was of immediate concern for Unit 2 because of the forthcoming refueling outage. The licensee initiated Action Request A0451605 to enter this item into the corrective action system.

The licensee evaluated several options to address this condition. Options that were initially considered included allowing the system to rupture and making up with safety injection or entering containment and isolating the level indication system following a loss of decay heat removal. The licensee determined that these solutions were undesirable because each involved a loss of level indication during midloop operations and would further complicate an event. The licensee chose to upgrade the system to withstand pressures of up to 250 psig by replacing the tubing with stainless steel and installing valves with a high pressure rating. To accomplish this task, the licensee initiated Design Change Package DCP J-50422 to upgrade the system. In addition, Procedure OP AP SD-0 was revised to limit pressurization of the reactor coolant system

to 250 psig. This modification was completed on February 17, and the system was placed in service for midloop operations on February 19. The inspectors noted that the upgraded reactor vessel refueling level indicated system performed satisfactorily during both entries into midloop operations for Unit 2.

The inspectors noted that the apparent cause of this issue was the failure of operations procedure preparers and reviewers to recognize that the reactor vessel refueling level indication system was not qualified for the pressures specified in the proposed strategies for mitigation of a loss of decay heat removal. Both the preparer and the reviewer of Procedure OP AP SD-0 were operations personnel that were not knowledgeable of the system's design basis. The licensee's quality assurance plan only required the licensee to make a determination if a cross-disciplinary review was required for any new procedures. The licensee determined that such a review was not required for Procedure OP AP SD-0. Therefore, Procedure OP AP SD-05 was not reviewed by design or system engineers to ensure that the systems affected by the new procedure were designed to withstand the referenced conditions. This condition existed for approximately 4 years and encompassed several refueling outages for both units when the conditions that could potentially use Procedure OP AP SD-0 were in place. This issue was mitigated by the fact that a safety injection pump was available during midloop operations and could have adequately made up coolant inventory to the core in the event of a reactor vessel refueling level indication system rupture.

The inspectors reviewed the proposed corrective actions for AR A0451605 and noted that these actions included upgrading the reactor vessel refueling level indication system, revising procedures and drawings to reflect the modification, and performing a maintenance preventable functional failure evaluation. These corrective actions were proposed by February 27. At the end of the inspection period (March 28), no further corrective actions were proposed.

Based on the above discussion of the apparent cause of the problem, the inspectors concluded that corrective actions did not address the failure of the procedure preparation and approval process to identify the need to upgrade the reactor vessel refueling level indication system or the need to choose another mitigation strategy. Because of the incompleteness of the corrective actions, the exercise of discretion was not considered warranted.

The failure to properly translate the design of the reactor vessel refueling level indication system into Procedure AP SD-0 is a violation of 10 CFR Part 50, Appendix B, Criterion III (50-275;323/98007-05).

c. <u>Conclusions</u>

A violation was identified for failure to translate the design of the reactor vessel refueling level indication system into abnormal operating procedures. The licensee exhibited good attention to detail in identifying this issue during simulator training. Corrective actions for this violation failed to address deficiencies in the procedure preparation and

approval process.

O5 Operator Training and Qualification

O5.1 Unit 2 Outage Training

a. Inspection Scope (71707)

The inspectors observed training in preparation for the shutdown and anticipated hot midloop evolutions.

b. Observations and Findings

From February 10 to 13, 1998, the inspectors observed both classroom training for the Unit 2 shutdown and hot drain down to midloop evolutions. This training included both classroom and simulator scenario sessions for the two expanded crews that were anticipated to perform these evolutions. System engineers that would be performing tests during the evolutions were involved in order to anticipate any impact on plant activities that the tests might have. The inspectors noted valuable lessons learned and procedural changes that were identified during the training and that, in particular, the simulator training was professional and well executed. The licensed operators and shift engineers viewed their training to be valuable and would ensure successful performance of the planned shutdown evolutions. The inspectors noted the actual observed evolutions were implemented without major incident. The inspectors also determined that the involvement of the test engineers with the crew training was a valuable contribution to the outage and was an improvement over the training provided in the past. Evidence of the value of this training was exhibited by the identification that the reactor vessel refueling level indication system was not designed for pressures required in the abnormal operating procedures.

c. Conclusions

The training provided for Unit 2 outage preparation was implemented well and provided valuable lessons learned and necessary procedural changes. The inspectors noted, in particular, that the simulator training was professional, well executed, and identified a vulnerability in the abnormal operating procedures.

II. Maintenance

M1 Conduct of Maintenance

M1.1 Maintenance Observations

a. Inspection Scope (62707)

The inspectors observed portions of the following work activities:

- MP M-7.53, Reactor Coolant Pump Motor Ten (10) Year Inspection for Reactor Coolant Pump 2-1
- Replace Flow Element FE-974 in the Safety Injection to Loop 1 Cold Leg, Work Order C0153639
- Replace Flow Element FE-975 in the Safety Injection to Loop 2 Cold Leg, Work Order C0153640
- Replace Existing Grating at Residual Heat Removal Sump for 2R8, Work Order C0154660
- Install Ground Buggy in Cubicle for Component Cooling Water Pump 1-1., MP E-7.11B, Revision 17, for Clearance CL 00057211,
- Preventative Maintenance on Auxiliary Feedwater Pump 1-2, Work Orders R0161465 and R0170439
- Remove and Replace Valve FW-2-183, Work Order C0144622
- Remove and Replace Gasket on Piping Flange Downstream of Containment Spray Pump 1-2 Vent Valve CS-1-24, Work Order C0156657
- Sample Containment Spray Pump 1-2 Pump Bearing Oil, Work Order R0178536
- Sample Containment Spray Pump 1-2 Motor Bearing Oil, Work Order R0170178
- Perform Motor Preventative Maintenance on Containment Spray Pump 1-2, Work Order R0162541

b. Observations and Findings

On March 2, the inspectors observed portions of the 10-year inspection of the reactor coolant Pump 2-1 motor. The technicians had the work package at the job site and were performing the steps as written. The torque wrench used was calibrated and had the proper range as specified in the procedure. An alternating pattern was used to tighten the bolts, performing three passes with increasing torque values.

The inspectors observed the replacement of Flow Elements FE-974 and -975 in the safety injection lines to the Loop 1 and Loop 2 cold legs. A radiological catch bag, tube and bottle were setup on each line to contain the contaminated water released when the flanges were unbolted. This arrangement was effective in controlling the spread of contamination. Scaffolding to support the work was properly erected and had the required inspection tags attached. A nuclear quality services inspector was observed verifying the gasket, all-thread, nuts, bolts, lubricant and new orifice plates were as specified in the work package. The nuclear quality services inspector also verified the

orientation of the new orifice plate was correct for the direction of flow and matched the orientation of the old orifice plate.

The inspectors observed the modifications made to the Unit 2 residual heat removal containment recirculation sump and inspected the final configuration after the new screens were installed. Cutting, grinding, and welding on the structure were properly controlled, with the required posting of permits for open flame and combustibles, and fire watches in place. The modifications appeared to be effective in closing all the paths for small debris to enter the sump. New flashing installed around the grating was effective in closing the gaps between the grating and the sump housing. The design change is described in Section E2.1 of this report. During the inspection of the inside of the sump, the inspectors noted no debris or peeling paint inside, the condition of the sump housing appeared sound, with sufficient structural support, a secondary internal screen that covered the inlets to the suction piping, and no visible unscreened path into the sump.

On March 3, the inspectors observed the installation of a ground buggy in the cubicle for component cooling water Pump 1-1 for maintenance. The maintenance personnel installing the ground buggy were knowledgeable of the equipment, and the correct method and indications of proper alignment of the buggy in the cubicle. They were aware of the required safety precautions, using flash suits, ensuring all unnecessary personnel were out of the room, ensuring the doors were posted to prevent entry, and using a voltage tester to confirm the load side of the breaker cubicle was deenergized. They inspected the ground buggy to verify correct configuration and current rating. An operator was also present to verify the correct breaker cubicle, correct ground buggy configuration (line side or load side), and correct current rating. Both the operator and the maintenance personnel had their procedures at the job site and used them. The ground buggy was installed without difficulty.

The inspectors observed the removal and replacement of flanged butterfly Valve FW-2-183, auxiliary feedwater Pump 2-3 suction isolation valve. The alignment of the auxiliary feedwater pump to its motor was monitored during the loosening of the bolts and removal of the valve to ensure the suction pipe was not exerting excessive load on the pump suction. The alignment was rechecked following installation of the new valve. The maintenance personnel had the procedure at the job site and used it correctly. The torque wrench was within its calibration. The clearance for the work, C0144622, was verified by the inspector to be hung and to adequately protect the equipment and personnel.

On March 19, the inspectors observe performance of routine maintenance on containment spray Pump 1-2 by technical maintenance and mechanical maintenance. This work included obtaining samples of motor bearing oil and pump bearing oil, replacing a gasket on a pipe flange downstream of the pump vent valve, and routine cleaning, inspecting, and testing of the motor leads and junction boxes. A radiation protection technician assisted a mechanic in removing and replacing the pipe flange gasket. A catch bag, tubing, and bottle were erected prior to opening the flange to contain any liquid. Good radiological practices were used to survey the parts and

equipment and to minimize the contamination released by the work. The survey equipment and torque wrench were inspected and found to be within calibration. The new gasket material was verified to match that specified in the work package. The flange bolts were observed to be torqued to the value specified in Procedure MP M-54.1.

The oil samples were obtained as specified in Procedure MP M-56.7, "Lubricant Sampling." The oils used to refill the bearings were verified to agree with the lubricant charts contained in Procedures MP M-56.24 and E-56.1, and to agree with the nameplates mounted on the containment spray pump. The electricians cleaning and inspecting the motor leads and junction boxes were thorough, and checked the torque on the bolted connections using a torque wrench of the proper range and within its calibration frequency. One bolt that appeared to be corroded was removed, cleaned, inspected, and reinstalled after its condition was determined to be acceptable. The clearance for this work, CL0057967, was reviewed and found to be adequate to protect the work and each tag was hung in the correct location.

c. Conclusions

A number of maintenance activities were observed and were performed in accordance with the procedural requirements. Good coordination between technical maintenance, mechanical maintenance and radiation protection was observed in performing several maintenance tasks concurrently on the containment spray pump, thereby reducing the time the pump was inoperable due to maintenance.

M1.2 Surveillance Observations

a. Inspection Scope (61726)

Selected surveillance tests required to be performed by the Technical Specifications 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 Technical Specifications; and (4) test results satisfied acceptance criteria or were properly dispositioned.

The inspectors observed all or portions of the following surveillances:

- STP M-13F 4KV Bus F Non-SI Auto -Transfer Test, Revision 22
- STP M-15 Integrated Test of Engineered Safeguards and Diesel Generators, Revision 29
- STP R-6 Low Power Reload Physics Tests, Revision 7
- STP R-30 Reload Cycle Initial Criticality, Revision 12

- STP R-31 Rod Worth Measurements Using the Rod Swap Method, Revision 8
- STP M-120 Firewater Availability to Centrifugal Charging Pump Coolers, Revision 2

The inspectors also reviewed the results of the following surveillances:

- STP R-17 Estimated Critical Position; Revision 12A
- STP G-8C Operational Checkout of the Reactivity Computer, Revision 7
- STP R-7A Determination of Moderator Temperature Coefficient, Revision 10

b. Observations and Findings

On February 24, the inspectors observed the performance of surveillance test procedure (STP) STP M-120, "Firewater Availability to Centrifugal Charging Pump Coolers," Revision 2, on the Unit 1 centrifugal charging pumps. This test demonstrated the ability to flow cooling water through the hoses and manifold to provide cooling to either centrifugal charging pump using firewater. Firewater would be used in the event component cooling water was lost to both centrifugal charging pumps. The operators had the procedure and signed off the steps as they were performed. All of the hoses, fittings and manifold worked as design with no observable leakage. The test adequately demonstrated that the equipment could be properly installed and would provide cooling water to the centrifugal charging pump cooling piping. The operators performed self checking and independent verifications, by verifying the valve numbers on the tags matched the procedure prior to operating the valves. The operators were careful to cleanup the small amount of water that spilled during the disconnection of the hoses.

On March 15, the inspectors observed the pretest briefing and performance of surveillance test procedure STP M-13F, "4KV Bus F Non-SI Auto-Transfer Test." The pretest briefing was comprehensive, covering precautions and limitations, prerequisites, communications and responsibilities, major test evolutions, and expected results. The performance of the test was well coordinated, with clear communications between the test personnel and control operators, including 3-part communications. During the test, all equipment operated as expected and the test results satisfied the acceptance criteria.

On March 17, the inspectors observed the pretest briefing and performance of surveillance test procedure STP M-15, "Integrated Test of Engineered Safeguards and Diesel Generators." The pretest briefing was comprehensive, covering precautions and limitations, prerequisites, communications and responsibilities, major test evolutions, and expected results. Several other surveillance requirements were satisfied concurrently with this test, including STP V-11, "Containment Isolation Phase B

Valves FCV-355, FCV-356, FCV-357, FCV-363, FCV-749, and FCV-750." The inspectors observed good communications and coordination between test participants, which was necessary to capture the starting time of the many different pumps and fans, and to coordinate the recording of data from control board meters. During the test, the newly installed load tap changer for the startup transformer was used per the operations procedure to transfer the vital buses from startup to auxiliary power.

On March 24, the inspectors observed the preparations for and achievement of initial criticality and performance of low power physics testing. A pretask briefing was performed by the shift foreman and reactor engineer. Management oversight was provided during the briefing and evolution by the engineering services manager. The briefing covered responsibilities and communications, Technical Specification requirements, including the special test exceptions and related surveillances, precautions and limitations, prerequisites, and brief description of the physics tests. The prerequisites were verified by the inspectors. Operations limited other plant activities that could cause distractions to the control room or unnecessary alarms during the physics testing.

Communications between the reactor engineers and the control operators were clear with 3-part communications consistently used. The reactor engineers were knowledgeable of the test requirements and test equipment. They continually evaluated the data against expected results and made conservative decisions in implementing the procedures. The approach to criticality was performed slowly and cautiously, with criticality achieved well within the allowed deviation from the estimated critical conditions. The control operators maintained the plant stable to ensure good data for the reactor physics tests. The test results met both the acceptance criteria and review criteria.

c. Conclusions

The inspectors observed a number of surveillance tests and found that the surveillances observed were performed in a cautious manner with self-checking and proper communications employed. The procedures governing the surveillance tests were technically adequate and personnel performing the surveillances demonstrated an adequate level of knowledge. The inspectors noted that test results appeared to have been appropriately dispositioned.

M1.3 Containment Inspection Prior to Closeout for Containment Integrity

a. Inspection Scope (62707)

The inspectors toured containment during the period when containment work activities were finishing and efforts to clean up in preparation for containment closure were in progress.

b. Observations and Findings

The cleanup effort was well organized with coordination between completion of the final work activities, surveying and decontamination of radiologically controlled areas, removal of equipment, tools, and supplies, as well as cleaning to meet housekeeping standards occurring simultaneously. Loose insulation buckles had been a concern in the previous outage and were addressed early in the cleanup schedule. The inspectors found no unfastened, broken or missing insulation buckles during their tour. The reactor coolant pump oil collection system was inspected and found to be properly assembled. The various sections were bolted or fastened into place. The joints between the sections were sealed with approved sealant. The oil collection tank was inspected and found to be clean and empty of oil. The inspectors noted several small bolts and nuts and other debris. These were identified to the licensee and removed. The inside of several mechanical panels were inspected and found to be in good materiel condition with no housekeeping problems. Radiologically contaminated material was properly separated from noncontaminated material and appropriately labeled and posted.

c. Conclusions

The containment cleanup and closeout activities were appropriately controlled, and the material condition of containment areas was satisfactory for restart of Unit 2.

M2 Maintenance and Materiel Condition of Facilities and Equipment

M2.1 Review of Part Length Control Rod Drive Mechanisms Transition Welds (Unit 2)

a. Inspection Scope (62707)

The inspectors reviewed the ultrasonic examination of the transition welds on the part length control rod dive mechanisms to find if the same problem found at Prairie Island Unit 2 existed at Diablo Canyon.

b. Observations and Findings

Prairie Island Unit 2 shut down on January 24, 1998, because of reactor coolant system pressure boundary leakage. The licensee estimated the leak rate to be 0.2 gallons per minute as measured by mass balance calculations and radiation monitors. On February 27, 1998, Northern States Power reported to the NRC that one of four part length control rod drive mechanisms, G-9, had a leak in the lower transition weld of the motor tube.

The part length control rod drive mechanism had, by design, a section of the pressure boundary motor tube made of 403 stainless steel. This design by the vendor allowed motor coils to be located outside the pressure boundary because magnetic fields can penetrate 403 stainless steel. The rest of the motor tube assembly consisted of 304

stainless steel. To weld the two different stainless steels together, the manufacturer layered or buttered the 403 stainless with 309 stainless steel. A final weld consisted of 308 stainless steel and, normally, the stator shroud assembly covered these welds during operation. At Prairie Island, the G-9 motor tube had a manufacturing defect in the buttering weld. The manufacturing defect was a circumferential crack with a high temperature oxide layer and no evidence of operational extension. The crack was almost 360 degrees, initiated on the inside diameter and opened to the outside diameter for about ½ inch. Northern States Power concluded that this was a hot crack induced during manufacturing welding with no sign of propagation during service.

Diablo Canyon had eight part length control rod drive mechanisms and was similar to Prairie Island. The licensee modified one part length control rod drive mechanism at Diablo Canyon to be a head vent. The vendor ultrasonically inspected the remaining seven for the Diablo Canyon licensee. The vendor completed a hot cell inspection of G-9 plus other motor tubes from Prairie Island and used the same basic technique at Diablo Canyon. Diablo Canyon motor tubes contained the same heat of buttering material as G-9. One motor tube (M-6) had been weld repaired during manufacturing similar to the manufacturer repair of G-9. Just as the Prairie Island motor tubes, Diablo Canyon welds were both penetrant and radiographically tested after buttering and also final welding.

The ultrasonic testing of seven of the eight motor tubes done by the vendor was a remote automated inspection with a customized clamp and track for the search unit and bubbler to supply the water couplant. The vendor did a ½ vee inspection with a 1/4 inch, 4 MHZ search unit using a 45 degree refracted longitudinal wave for the lower transition weld where the thickness was approximately .400 inches. A 45 degree 2.25 MHZ, shear wave inspection done on the lower transition weld was not as informative as the 45 degree refracted longitudinal wave inspection. The upper weld where the thickness was approximately .490 inches, used a 60 degree, refracted longitudinal wave. The technique and personnel were Electric Power Research Institute (EPRI) qualified for intergranular stress corrosion cracking ultrasonic testing, and were in accordance with Procedure DC-800-001, and Field Change Notice Number 1. Also the technique and equipment was reported to have been used for the "Performance Demonstration Initiative."

The vendor calibrated the system on two mockups of the motor tube as calibration blocks, one for the lower transition weld and one for the upper transition weld. Each calibration block had 0.030 notches on the inside diameter placed at the location of the buttering weld. These notches gave some measure of the minimum detection limit of the system. The vendor used A, B, and C-scan displays to analyze the results. The scans showed low intensity noise reflectors from the grain structure of the weld metal, but did not show any reflectors like that found in G-9.

The inspectors reviewed all the ultrasonic data collected on the lower and upper transition welds of the seven motor tubes inspected and the manufacturing records. The inspectors reviewed the calibration records of the lower transition welds. The

inspectors witnessed the calibration and inspection of four upper transition welds. The tapered geometry and an outside diameter offset (.010 inches) on the motor tube in the areas in question presented technique challenges. The inspectors verified the inspection of the areas in question. The calibration notches of 0.30 inches were discernable and the inspectors found no apparent defects.

c. Conclusions

The license's approach to the inspection of part length control rod welds was sound and aggressive. The inspectors found the ultrasonic testing showed the seven motor tubes' upper and lower transition welds were free of the same kind of defect found at Prairie Island on the G-9 motor tube.

M4 Maintenance Staff Knowledge and Performance

M4.1 Ground Buggy Installation (Unit 2)

a. Inspection Scope

The inspectors reviewed the circumstances surrounding a failed attempt to install a ground buggy in safety-related 4160 vac Cubicle 52HG5.

b. Observations and Findings

On February 24, 1998, two contract technicians attempted to install a ground buggy on the deenergized line side of a diesel generator load in safety-related 4160 vac Cubicle 52HG5 in Unit 2 in accordance with Procedure MP E-57.11B, "Protective Grounding," Revision 17. The bus side of Cubicle 52HG5 remained energized with 4160 vac to provide power to other loads. Unit 2 was defueled at the time. When the technicians installed the ground buggy, they inserted the buggy into the cubicle until it was mechanically stopped. However, the buggy was hitting an obstruction and was not fully inserted into the cubicle.

The technicians began raising the ground buggy to mate with the cubicle line side stabs, however, because the ground buggy was not fully installed, it began to tilt forward. The technicians attempted to stop the elevator motor, which continued to run. The technicians then pulled the power plug to the motor, stopping the lift.

The ground buggy was found to have partially raised and tilted forward, opening the switchgear shutters and exposing the cubicle bussing. Because of the forward tilt of the buggy, the ground buggy stabs were almost touching the energized bus side stabs, which would have hard grounded the energized 4160 vac bus, challenged switchgear protective devices, and could have caused significant damage to the switchgear and injury to the technicians. The licensee successfully lowered the ground buggy from its improper position, and after inspection of the cubicle, installed another ground buggy.

The licensee found that the wheel runners on the original ground buggy were misaligned, which allowed the buggy to jam inside the cubicle prior to being fully inserted. The licensee issued instructions to inspect and repair all 4160 vac ground buggies for alignment and added this instruction to preventative maintenance requirements for the ground buggies.

The inspectors noted that Procedure MP E-57.11B and other associated licensee procedures did not provide any instructions on how to rack in ground buggies or how to verify proper installation prior to lifting. Licensee personnel stated that detailed ground buggy installation instructions had been provided after a ground buggy error in 1995 caused failure of the Unit 1 Auxiliary Transformer 1-1, as discussed in Inspection Report 50-275; 323/95-017. Subsequent to that time, specific instructions for ground buggy installation had been removed from Procedure MP E-57.11B, because the installation was considered by the licensee to be within the skill of the craft.

However, the inspectors noted that the improper installation was accomplished by contract personnel. The licensee stated that the two contract technicians were qualified by licensee training procedures to install ground buggies. The licensee provided the training records to the inspectors. The inspectors noted that these two contract employees had been certified as having the required training during previous outages and had received no additional training for this outage. The licensee stated that the site program for training of personnel for craft work, including ground buggy installation, did not require periodic retraining for permanent or contract personnel. In addition, the licensee's craft procedures did not require that permanent personnel accompany contract personnel during performance of any work the contract personnel were qualified for.

The licensee issued additional instructions for installing ground buggies. The inspectors questioned whether it was appropriate to give contract personnel, who may have recently worked at other sites with different hardware, permanent qualification for Diablo Canyon. The licensee's evaluation of the event, Quality Evaluation Q0012011, stated that the corrective action for the installation problem would include a review of the policy for permanent qualification of contract personnel. This review will identify specific qualifications that will require remediation of contractor qualifications prior to performing the task, based on criteria such as safety significance and frequency of prior use. The inspectors considered that the licensee's corrective actions were adequate.

The inspectors considered that Procedure MP E-57.11B was not appropriate to the circumstance in that the licensee allowed permanent qualification of contract personnel, allowed these contract personnel to work on safety-related equipment without supervision, and these contract personnel did not have the necessary skill of the craft to successfully complete the work.

The inspectors reviewed the licensee's corrective actions for previous violations for failure of maintenance and operations personnel to follow procedures for removal of a ground buggy in 1995, as discussed in NRC Inspection Report 50-275; 323/95-017.

The inspectors considered that the February 24, 1998, installation problem did not result from inadequate corrective actions for the previous violations. In the February 1998 error, both operations and maintenance personnel had followed licensee procedures. In addition, the inspectors considered none of the previous violations concerned inadequate procedures or craft skills.

Therefore, since the improper installation of the ground buggy did not result from inadequate corrective actions from previous violations, this self-revealing and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-323/98007-06).

c. Conclusions

A noncited violation was identified for failure to provide a procedure appropriate to the circumstances for ground buggy installation. The improper ground buggy installation had the potential to have caused significant damage to safety-related equipment and injure workers.

M8 Miscellaneous Maintenance Issues

M8.1 (Open) Violation 50-275;323/96014-03: failure to remove action request tags from the control boards. This violation involved five examples of failure to remove stickers from the control boards following correction of the deficiencies. The root cause of this violation was that no formal program existed to control these action request stickers. As corrective action, the licensee: (1) performed an audit of both Units 1 and 2 control boards to ensure that all control board action request stickers were in place and those that were resolved had been removed, (2) reprogrammed the plant's computer work management system such that the status of installation and removal of the control board action request stickers was tracked and such that the work order could not be closed out without removal of the stickers, (3) revised procedures for controlling control room action request stickers to clarify management's expectations, and (4) issued a memorandum that indicated its expectations for supervisory personnel and its intention to hold personnel accountable.

The inspectors reviewed documentation that verified that these items were completed. The inspectors noted that Procedure OP2.ID2 "Problem Identification and Resolution - Action Requests" was revised to state that personnel may enter "Y" in the action request sticker removal block in the licensee's plant computer to signify that the stickers were removed from the control boards. The inspectors concluded that this procedure revision did not fully address the corrective actions because it did not appear to proceduralize the committed actions such that personnel could be held accountable.

On March 19, 1998, the inspectors performed an audit of existing control board action request stickers to verify the effectiveness of corrective actions. The inspectors identified a total of six discrepancies between the licensee's existing list of control board action requests and the stickers on the panels, as listed below:

- The inspectors identified three control board action request stickers on the panels that were not being tracked in the licensee's computer data base as control board deficiencies. The inspectors noted that the existing process in the licensee's computer data base would remove these stickers.

- The inspectors identified one control board action request sticker that was left on the panels for completed work and was faded such that the problem described on the stickers was illegible.

- The inspectors identified that the computer status block for removal of the control board sticker for Action Request A0437900 was changed to "Y," despite the fact that the sticker was still in place and the problem was not corrected.

- The inspectors identified one instance in which an action request was open for a control board action request, the work was not complete, yet the sticker was removed.

The inspectors informed the technical maintenance foreman, who corrected the discrepancies in the licensee's control board action request tracking system.

The inspectors concluded that the corrective actions for Violation 50-275;323/96014-03 were specifically directed to ensuring that control board action request stickers were removed when the work was complete, but did not appear to fully address the need to closely control these deficiency tags. The licensee's programs to ensure that the control board action requests stickers reflected the licensee's tracking list and the up-to-date plant configuration warranted further licensee attention. The licensee implemented the control board action request program to inform operators of equipment that was deficient, and with inaccuracies in the program, the operators could be misled.

Because of the additional deficiencies identified with the program, Violation 50-275;323/96014-03 will remain open for further inspector review of licensee improvements to the control of control board action requests.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Residual Heat Removal Recirculation Containment Sump Grating/Screen Modifications

a. Inspection Scope (37551)

The inspectors reviewed Design Change Package DCN N-050317, Revision 0, and the related licensing basis impact evaluation and screens, Final Safety Analysis Report Update Change Request, and field changes. The inspectors also observed the modification work in progress and inspected the sump upon completion of the modification.

b. Observations and Findings

As the result of the potential to pass debris through the previously installed containment sump screen and grating that could cause blockage of flow through the safety injection lines, modifications were made in refueling outage 2R8 to reduce the size of the screen openings from 3/16 inch to 1/8 inch. This modification was made in addition to emergency core cooling system injection line modifications (DCP N-50286) that increased the minimum openings in the flow paths. These modifications removed the physical possibility of debris entering the emergency core cooling system that could block the throttle/runout valves in the safety injection flow paths.

The design change evaluation identified the applicable design bases and design inputs affected by the modification. The technical review portion of the evaluation confirmed the sump would meet its design function following implementation of the modification. This review addressed the impact of: performing the modification during the refueling outage with fuel in the vessel, on routine operation, on the high energy line break study, on pump available net positive suction head, on hydraulic design considerations, on the seismic interaction evaluation, on consistency with Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," material compatibility and consistency with licensing documents, in addition to other design considerations.

The 10 CFR 50.59 safety evaluation concluded that an unreviewed safety question was not involved, nor was a change to the Technical Specifications involved. The proposed Final Safety Analysis Report Update change and revision to Design Criteria Memorandum T-16, "Containment Function," were consistent with the modification. The design change package was prepared in accordance with the applicable Procedure, CF33.ID9, "Design Change Package Development." The inspectors identified no concerns with these reviews.

Following the completion of modification, the sump was inspected for openings in the structure or potential flow paths that could bypass the screens. No openings greater than that specified in the design were found. Potential openings around piping or supports that penetrated the sump structure were effectively closed. Joints between sections of grating and between the edges of the screen and the supporting structure were effective closed by the design and installation of the modification.

c. Conclusions

The inspectors concluded that the design change package and associated safety evaluation for replacement of the Unit 2 recirculation sump screens was comprehensive, and the conclusions were reasonable. The design change was effective in improving the containment sump's ability to screen out debris that could block safety injection flow paths.

E8 Miscellaneous Engineering Issues (92901)

E8.1 (Closed) Violation 96016-06: failure to perform prompt operability assessment. The inspectors identified that following documentation of degraded conditions in the diesel generator voltage regulator boards, the licensee failed to perform a prompt operability assessment. For corrective actions, the licensee: (1) replaced the voltage regulator boards, (2) established a daily action request review team that screens each new action request for operability issues, (3) trained nuclear technical services personnel in the license procedures for operability assessments, (4) revised Procedure OM7.ID12 "Prompt Operability Assessments" to require shift supervisor notification of all degraded conditions, (5) briefed all shift supervisors on the responsibility for maintaining operability, (6) issued an all employee letter emphasizing management expectations with respect to operability assessments, (7) initiated an engineering directors meeting to discuss emergent issues that may have operability concerns, and (8) performed a case study on lessons learned from the violation. The inspectors reviewed documentation that established that these items have been completed satisfactorily. This item is closed.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments

During this inspection period, the inspectors observed radiation protection controls. The inspectors noted that licensee personnel followed basic radiation practices such as proper wearing of dosimetry and observance of radiation protection boundaries. Licensee management's efforts to keep exposures as low as reasonably achievable during Refueling Outage 2R8 appeared to be successful in that total outage exposure was 147 person-rem, while the outage exposure goal was 160 person-rem. The licensee's cleanup of the reactor coolant system following shutdown of Unit 2, and the use of mock-up training for several outage tasks, contributed to the lower exposure. This was an improvement over previous refueling outages.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

During routine tours, the inspectors noted that the security officers were alert at their posts, security boundaries were being maintained properly, and screening processes at the Primary Access Point were performed well. During backshift inspections, the inspectors noted that the protected area was properly illuminated, especially in areas where temporary equipment was brought in.

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 April 7, 1998. 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

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

W. G. Crockett, Manager, Nuclear Quality Services
R. D. Gray, Director, Radiation Protection
T. L. Grebel, Director, Regulatory Services
S. A. Hiett, Director, Operations
D. B. Miklush, Manager, Engineering Services
J. E. Molden, Manager, Operations Services
D. H. Oatley, Manager, Maintenance Services
R. P. Powers, Vice President and Plant Manager
L. F. Womack, Vice President, Nuclear Technical Services

INSPECTION PROCEDURES (IP) USED

IP 37551	Onsite Engineering
IP 60710	Refueling Activities
IP 61726	Surveillance Observations
IP 62707	Maintenance Observation
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 92902	Followup - Maintenance
IP 92903	Followup - Engineering
IP 93702	Prompt Onsite Response to Events at Operating Power Reactors

ITEMS OPENED AND CLOSED

Opened

50-323/ 98007-02	VIO	Failure to provide appropriate procedure for nonseismic hoist stowage (Section O1.3)
50-323/ 98007-03	VIO	Failure to restore high flux alarm during core reload (Section O1.5)
50-275;323/ 98007-04	VIO	Multiple failures to implement clearance procedure (Section O1.6)
50-275;323/ 98007-05	VIO	Failure to implement design of level indicating system into abnormal procedures (Section O3.1)

Closed

50-275;323/ 96016-06	VIO	Failure to perform prompt operability assessment (Section E8.1)

Opened and Closed

50-275;323/ 98007-01	NCV	Failure to provide an appropriate procedure for switching power supplies (Section O1.2)
50-27;323/ 98007-06	NCV	Inadequate ground buggy installation procedure (Section M4.1)

Discussed

50-275;323/ 96014-03	VIO	Failure to remove action request tags from the control boards (Section M8.1).