ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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License Nos.:	DPR-80 DPR-82
Report No.:	50-275/97-07 50-323/97-07
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:	April 28 through May 2, 1997
Inspector:	W. P. Ang, Senior Reactor Inspector, Engineering Branch
Approved By:	C. A. VanDenburgh, Chief, Engineering Branch Division of Reactor Safety

ATTACHMENT: Supplemental Information

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EXECUTIVE SUMMARY

Diablo Canyon Nuclear Power Plant, Units 1 and 2
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Engineering

- The inspector verified that the licensee had performed the corrective actions that they had committed to perform for the main steam safety valve setpoint problems. The licensee had replaced the discs in the Unit 1 valves and intended to replace the discs in the Unit 2 valves during the next refueling outage in an effort to eliminate disc-to-seat sticking that was believed to be contributing to the valve setpoint discrepancies (Section E8.1).

- The licensee identified and reported as-found out-of-tolerance pressurizer safety valve setpoints that were violations of Technical Specification 3.4.2.2. The licensee had attempted several reasonable corrective actions to preclude recurrence of the problem. The violations were identified as a noncited violation (Section E8.2).

- Although the licensee had implemented adequate corrective actions for an incorrectly sized spring pack in an auxiliary feedwater system motor-operated isolation valve, the licensee did not implement adequate and prompt corrective action for the problem when it was first identified in May 1994. The inadequate licensee corrective action was identified as a noncited violation (Section E8.3).

Report Details

III. Engineering

E8 Miscellaneous Engineering Issues

E8.1 Main Steam Safety Valve Open Items

The inspector reviewed the following seven open items involving various aspects of the Main Steam Safety Valves. The details and inspection effort is discussed in the following sections:

(Closed) LER 50-275/94-300: Licensee Event Report 1-94-03, Revision 1, Main Steam Safety Valve As-found Setpoint Exceeds Technical Specifications.

(Closed) LER 50-323/94-301: Licensee Event Report 2-94-07, Revision 0, Main Steam Safety Valve's Exceeded Technical Specifications Tolerance for Lift Settings.

(Closed) LER 50-275/96-205: Licensee Event Report 1-96-03, Revision 0, Technical Specifications not Met During Main Steam Safety Valve Testing.

(Closed) LER 50-275/96-243: Licensee Event Report 1-96-13, Revision 0, Main Steam Safety Valve's Set Outside Technical Specifications.

(Closed) LER 50-323/96-244: Licensee Event Report 2-96-07, Revision 0, Technical Specifications not Met Due to High Initial Main Steam Safety Valve Lift Setpoints.

(Closed) Violation 50-275/96-180 (01014): Violation Involving Augmented Testing Issued in Inspection Report 96-012.

(Closed) Violation 50-275/96-180 (02014): Violation for Main Steam Safety Valve's Out-of-Tolerance High Issued in Inspection Report 96-012.

a. Background

Licensee Event Report 1-94-03, Revision 1

On February 9, 1994, the licensee tested all 20 of Unit 1 main steam safety valves using Trevitest equipment. One valve was found to be between 1 and 3 percent below the Technical Specifications required setpoint. Five valves tested between 1 and 3 percent high. Five valves tested greater than 3 percent high. Two valves did not lift due to test equipment limitations. Following additional testing and/or adjustment, all of the valves lifted within the Technical Specifications 3.7.1.1 tolerances. On February 11, 1994, a licensee analysis of the test data determined that the main steam safety valves were capable of performing within the final safety analysis report update analyzed condition.

On March 5, 1994, the licensee tested all 20 of Unit 2 main steam safety valves using Trevitest equipment. One valve tested between 1 and 3 percent low. One valve tested greater than 3 percent low. Seven valves tested between 1 and 3 percent high. Six valves tested greater than 3 percent high. Following additional testing and/or adjustment, all of the valves lifted within Technical Specifications 3.7.1.1 tolerances. The licensee analyzed the Unit 2 test data and concluded that the main steam safety valves were capable of performing within the final safety analysis report update analyzed condition.

The licensee reset each valve that was found to be out-of-tolerance. Additional valve testing was performed at the nuclear steam supply system vendor (Westinghouse) facility using live steam. In an effort to prevent recurrence of the problem, the licensee: (1) purchased a new lift-assist test device, correlation tested the device to live steam, and planned to use the device for future surveillance testing; (2) submitted a Technical Specifications change request for relaxation of the main steam safety valve as-found setpoint tolerance; and (3) continued to participate with industry (American Society of Mechanical Engineers) and the Nuclear Steam Supply System vendor test group regarding main steam safety valve testing.

Licensee Event Report 2-94-07, Revision 0

On October 8, 1994, the licensee found that 12 of Unit 2 main steam safety valves that had been tested at the Westinghouse facility using live steam during the previous refueling outage lifted at pressures greater than the +1 percent setpoint tolerance of Technical Specification 3.7.1.1. The testing also identified that 2 of the 12 out-of-tolerance main steam safety valves were outside the +3 percent test failure criteria specified in ASME OM-1.

As corrective action, the licensee reset each valve that was found to be out-of-tolerance. Licensee corrective actions to prevent recurrence were the same as those provided for Licensee Event Report 1-94-03, Revision 1, discussed above.

Licensee Event Report 1-96-03, Revision 0

On March 26, 1996, with Unit 2 in Mode 1 at 100 percent power, the licensee identified during testing of the main steam safety valves, using the AVK acoustic sensors, that one main steam safety valve on Steam Generator 2-1 had a lift pressure that was higher than Technical Specifications tolerances. After testing all valves on March 27, 1996, the licensee identified that a total of 2 of 20 main steam safety valves had setpoints higher than the Technical Specifications tolerances. Following additional testing and required adjustments, all of the valves were left with measured setpoints within the +/- 1 percent Technical Specifications tolerance.

On April 2, 1996, the licensee commenced augmented testing of the Unit 1 main steam safety valves using the AVK acoustic sensors as committed to in License Amendments 108 and 107. The first of three main steam safety valves tested on

Steam Generator 1-1 had setpoints higher than the Technical Specifications tolerances. As previously agreed with the NRC, the testing results were discussed prior to expansion of testing. By April 11, 1996, 6 of 10 main steam safety valves setpoints had tested higher than the Technical Specifications tolerances on Steam Generators 1-1 and 1-2.

On April 12, 1996, at 4:50 p.m., the licensee determined that the Steam Generator 1-2 main steam safety valve high lift pressures could have resulted in steam generator pressures greater than 110 percent of the steam generator design pressure, which placed Unit 1 outside the design basis. At 5:30 p.m., the licensee made a 1-hour, nonemergency report to the NRC. On April 19, 1996, additional testing on the Steam Generator 1-3 main steam safety valve's confirmed that Steam Generator 1-3 was also outside the design basis.

On April 11, 1996, at the conclusion of testing on Steam Generator 1-2, one main steam safety valve was left at +1.4 percent, and on April 14, 1996, at the conclusion of testing on steam generator 1-3, one main steam safety valve was left at +1.2 percent. The valves were previously set on steam at the Westinghouse test facility during the previous refueling outage. Due to the lack of AVK correlation factors for these valves, and because the AVK test method could add as much as a 1.4 percent error to valve set pressures, the licensee did not reset the valves. The licensee felt that the Unit 1 valves without AVK correlation factors were within the Technical Specification's +/- 3 percent as-found requirement and may or may not have been within the Technical Specifications +/- 1 percent setpoint requirement. The licensee discussed their conclusion with the NRC in a conference call on April 12, 1996. As a consequence of a subsequent conference call with the NRC, the licensee decided to reset the two valves. The two valves were reset with measured setpoints within the +/- 1 percent Technical Specifications tolerances on April 21, 1996, using the AVK test equipment.

At the conclusion of testing of Unit 1 main steam safety valves on April 21, 1996, 12 of 20 main steam safety valves had been found out of Technical Specifications tolerances. Following testing and any required adjustments, all main steam safety valves on Units 1 and 2 were left with measured setpoints within the +/- 1 percent Technical Specifications tolerance.

As noted above, Units 1 and 2 main steam safety valves were retested, adjusted as required, and left with measured setpoints within the Technical Specifications required tolerance of +/- 1 percent. The licensee committed to perform the following corrective actions to prevent recurrence.

1) Testing of all 20 of Unit 2 main steam safety valves was performed at the Westinghouse test facility during Refueling Outage 2R7 to develop valve-specific correlation factors and to further confirm the correlation between the AVK test equipment and live steam testing.

2) All main steam safety valves on either unit will be tested upon return to Mode 3 (Hot Standby) from any future entry into Mode 5 (Cold Shutdown). Testing will continue until valve discs are replaced.

3) Testing of all main steam safety valves will be performed on an augmented test schedule starting 3 months after reaching full power on Unit 2 (completion of Refueling Outage 2R7) and 3 months from the end of the initial augmented testing of Unit 1 (April 21, 1996) with a 3-month interval between tests. Testing will continue until valve discs are replaced.

<u>NRC Inspection Report 50-275;-323/96-12 and Notice of Violations 50-275/96-180 (01014) and (02014)</u>

The NRC performed a special inspection of licensee activities associated with main steam safety valve testing on May 13-28, 1996. The results of the inspection were documented in NRC Inspection Report 50-275;-323/96-12. The inspection included a review of main steam safety valve testing problems, including those reported in the licensee event report's discussed above. On July 1, 1996, the NRC held a predecisional enforcement conference with the licensee to discuss apparent violations identified during the May 13-28, 1996, NRC inspection.

On July 10, 1996, the NRC issued two violations (01014 and 02014) resulting from the May 1996 inspection (NRC Inspection Report 50-275;-323/96-12). Specifically, the violations were for inadequate corrective action resulting from Unit 1, Main Steam Lead 1 out-of-tolerance lift setpoints, and for violation of Technical Specifications main steam safety valve lift setpoint tolerances identified by the licensee during testing on April 11 and 14, 1996.

On August 9, 1996, in letter DCL-96-167, the licensee agreed with both violations and stated the corrective actions for the violations. For the first violation,

1) The licensee conducted performance counseling with the individual who made several of the key decisions during the test program.

2) The licensee formalized its augmented test program into Surveillance Test Procedure STP M-77A, "Augmented Test Program For Main Steam Safety Valves," Revision 0. Procedure STP M-77A and Maintenance Procedure M-4.18, "Verification of Lift Point Using Ultra Star Assist Device for the Main Steam Safety Valves," Revision 14, were sent to the NRC in letter DCL-96-156, dated July 16, 1996.

3) Surveillance Test Procedure STP M-77A included provisions for scope expansion. It also stated that main steam safety valve testing shall be considered "in progress" until all of the valves, including any expansions, have been tested. The procedure stated that management's expectation, due to the importance of the testing, was that the testing should continue 12 hours per day, 7 days a week, unless test results indicated a design basis

concern. If a design basis concern was indicated, testing will be completed within 24 hours of the identification of the concern.

4) Surveillance Test Procedure STP M-77A required that:

   a) Operations and management be informed of out-of-tolerance conditions.

   b) Prompt operability assessments will be generated as necessary.

   c) Time limits have been set on the determination of operability of each steam lead and test expansion to other steam leads.

5) A project manager was appointed to specifically manage the main steam safety valve augmented testing program.

For the second violation,

1) Main Steam Safety Valves 223 and 14 were retested on April 21, 1996. The valves were left with settings within +/- 1 percent of setpoint.

2) Valve STP M-77A was written and approved to formalize the test program. It explicitly stated the acceptance criteria for as-left settings using the AVK test methodology.

3) The principal individuals involved with the in-situ testing of Main Steam Safety Valves 223 and 14 participated in the preparation of Valve STP M-77A and in the preparation or reviews of the response to the violation. Therefore, they were cognizant of the lessons-learned.

<u>Licensee Event Report 1-96-13, Revision 0</u>

On April 11, 1996, for Unit 1, and August 7 and 8, 1996, for Unit 2, with each unit in Mode 1 (Power Operation) at 100 percent power, the licensee determined that Technical Specification 3.7.1.1 was not met when the main steam safety valves were found to be set outside the +/- 1 percent tolerance. The main steam safety valves were set using AVK test equipment with inaccurate mean seat area correction factors. This condition was discovered on August 10, 1996, when Units 1 and 2 tripped as a result of a major western grid disturbance (Licensee Event Report 1-96-012-00). One Unit 1 main steam safety valve and two Unit 2 main steam safety valves lifted low during the unit trips. The licensee initiated investigations including post-trip verification of setpoints using AVK and Trevitest systems. The licensee investigation determined that inaccurate mean seat area correction factors were applied to the main steam safety valves during recent AVK testing. Following additional testing with Trevitest and any required adjustments, all main steam safety valves were left within Technical Specifications tolerance.

Licensee Event Report 2-96-07 Revision 0

On August 7, 1996, with Unit 2 in Mode 1 (Power Operation) at 100 percent power, the licensee determined that Technical Specification 3.7.1.1 was not met when the initial lift on one main steam safety valve did not meet Technical Specifications tolerance. On August 8, 1996, upon testing completion, the initial lifts of 7 of 20 main steam safety valves had not met the Technical Specifications tolerance. Following testing and any required adjustments, all main steam safety valves were left within the Technical Specifications tolerance of +/- 1 percent. Based on a preliminary analysis, a 1-hour nonemergency report was made to the NRC at 7:27 p.m. in accordance with 10 CFR 50.72(b)(1)(ii)(B) due to a concern that the design basis may have been violated.

The licensee subsequently determined the mean seat areas used to determine the lift points were inaccurate. Licensee Event Report 1-96-013-00, dated September 9, 1996, discussed the effects of inaccurate mean seat area correction factors on past main steam safety valve operability, including the effects on this event. The licensee discussed the as-found condition of the valves with the valve vendor and Westinghouse, and determined that it was unnecessary to include an additional 3 percent accumulation into the analysis of main steam safety valves that had an initial lift 5 percent greater than the Technical Specifications setpoint. A reanalysis using the vendor recommended mean seat area correction factor without accumulation showed that the lift setpoints did not result in an outside design basis condition. The cause of the high initial lifts was believed to be related to a sticking phenomenon between the main steam safety valve nozzle and disc seating surfaces.

b. Followup Inspection

The inspector reviewed the licensee event reports and inspection reports noted above. The inspector reviewed Surveillance Test Procedure STP M-77A, "Augmented Test Program For MSSVs," Revision 0, and Maintenance Procedure M-4.18, "Verification of Lift Point Using Ultra Star Assist Device for the Main Steam Safety Valves," Revision 14. The inspector reviewed the test data for the reported conditions. The inspector also discussed the main steam safety valve problems and licensee corrective actions noted above with the licensee.

The inspector noted that the main steam safety valve problems and corrective actions discussed in Licensee Event Reports 1-94-03, Revision 1, 2-94-07, Revision 0, and 1-96-03, Revision 0, were reviewed and discussed in NRC Inspection Report 50-275;-323/96-012. As noted above, a main steam safety valve setpoint Technical Specifications violation was identified during that inspection.

The inspector also noted that an NRC special inspection followed up the August 10, 1996, western grid disturbance and Diablo Canyon main steam safety valve early lift event. The results of the inspection were documented in NRC Inspection Report 50-275;-323/96-16. The licensee reported the main steam safety valve early lift problems associated with that event, and a similar problem identified on August 8, 1996, in Licensee Event Reports 1-96-13

and 2-96-07. The licensee attributed the cause of the early lifts to the use of incorrect mean seat area correction factors during previous setting of the valves and the previously noted disc sticking problem. The early lift and incorrect correction factor main steam safety valve problems were identified as Unresolved Item 50-275/96016-07 in NRC Inspection Report 50-275;-323/96-16. The unresolved item was followed up by the NRC and the inspection was documented in NRC Inspection Report 50-275;-323/96-21. The use of the incorrect correction factors and the consequent incorrect setting of the main steam safety valves were identified as Noncited Violation 50-275/96021-03 during that inspection.

The inspector discussed, with the licensee, the additional corrective actions for the main steam safety valve setpoint problems. The licensee informed the inspector that the Unit 1 main steam safety valve discs had been replaced during the ongoing refueling outage. The licensee informed the inspector that the replacement discs were lapped, pre-oxidized and heat treated Inconel X-750 material. The licensee chose the material in an effort to correct the sticking problem between the discs and the seats of the main steam safety valves. The licensee informed the inspector that the discs of the Unit 2 main steam safety valves would be similarly replaced during the February 1998 Refueling Outage 2R8.

During the inspection period, the licensee held a telephone conference call with the Office of Nuclear Reactor Regulation to discuss a modification to the main steam safety valve augmented testing schedule. The licensee reported that recent main steam safety valve testing had provided satisfactory results. The licensee informed the NRC that they intended to only test two Unit 1 main steam safety valve leads in lieu of four leads within 30 days of power operations. The licensee stated that all four steam leads would be tested after 90 days of power operations. The licensee considered that any sticking of the new replacement valve discs could be better evaluated after 90 days of exposure to system temperature and pressure conditions.

As noted above, the problems and licensee corrective actions noted in Licensee Event Reports 1-94-03, Revision 1; 2-94-07, Revision 0; 1-96-03, Revision 0; 1-96-13, Revision 0; 2-96-07, Revision 0, and Violations 96-180 (01014) and (02014) had been the subject of several NRC inspections. The inspector determined that the licensee had performed the testing corrective actions that they had committed to perform for the problems experienced. The inspector also noted that the licensee had replaced the valve discs in the Unit 1 valves and intended to replace the discs in the Unit 2 valves during the next refueling outage. The licensee's augmented testing of the main steam safety valves will continue to be monitored by the NRC. Based on the violations identified and the licensee's corrective actions, these open items are closed.

E8.2 <u>Pressurizer Safety Valve Open Items</u>

The inspector reviewed the following four open items involving various aspects of the pressurizer safety valves. The details and inspection effort is discussed in the following sections:

(Closed) LER 50-275/94-129: Licensee Event Report 1-94-09, Revision 0, Pressurizer Safety Valve Setpoint Spread.

(Closed) LER 50-275/95-101: Licensee Event Report 1-94-09, Revision 1, Pressurizer Safety Valve Setpoint Spread.

(Closed) LER 50-275/95-275: Licensee Event Report 1-94-09, Revision 2, Pressurizer Safety Valve Setpoint Spread.

(Closed) LER 50-275/95-277: Licensee Event Report 1-95-16, Revision 0, Technical Specifications not Met During Pressurizer Safety Valve testing.

a. Background

Licensee Event Report 1-94-09, Revisions 0, 1, and 2

On March 28, 1994, with Unit 1 in Mode 6, the licensee determined that Technical Specification 3.4.2.1 and 3.4.2.2 were not met when the setpoints for three Unit 1 pressurizer safety valves were found to be outside the 2485 psig plus or minus 1 percent tolerance during testing conducted at an offsite test facility. One valve was tested high at +2.70 percent and two valves were tested low at -1.81 and -2.70 percent. The valves were subsequently reset and tested satisfactorily. The licensee subsequently determined that the out-of-tolerance setpoints did not result in a condition outside the design basis of the plant.

The licensee determined that the deviations in set pressure resulted from the interaction of the spring, the upper spring washer, and the nose of the adjusting bolt as they pivot. The licensee developed and tested a prototype valve with a modified upper spring washer in order to reduce the minute buckling and pivoting that takes place in a standard valve arrangement. The valve with the modified washer was comparison tested with the standard Crosby valve. Testing was performed under identical environmental conditions for both valves. No adjustments were made to either valve type throughout the tests. The standard Crosby valve demonstrated a standard deviation of 22.5 psi. The magnitude of this value was nearly equal to the Technical Specifications tolerance of 24.8 psi. The prototype modified Crosby valve demonstrated a standard deviation of 11.4 psi.

The licensee intended to modify the pressurizer safety valves using the configuration of the tested prototype valve. However, on September 7, 1995, the licensee received a request for additional information regarding an NRC concern associated with certification testing of the modified valve configuration in accordance with the requirements of NUREG 0730, Item II.D.I. Due to the unavailability of a domestic facility with the capability to perform the recommended valve testing, the licensee deferred installing pressurizer safety valves with the modified design.

Licensee Event Report 1-95-16, Revision 0

During the Unit 1 Refueling Outage 1R6 in 1994, the licensee tested all three pressurizer safety valves at the Westinghouse Service Center. The valves were then returned to the plant, installed, and declared operable without any additional adjustment of the setpoints until the safety valves were checked at the Westinghouse Service Center in October 1995. On October 10, 1995, with Unit 1 in Mode 6, the licensee determined that Technical Specification 3.4.2.2 was violated during the previous cycle because the setpoints for two out of three Unit 1 pressurizer safety valves were found to be outside the 2485 psig, plus or minus 1 percent tolerance. The two valves tested high at +1.97 percent.

As noted in the discussion of Licensee Event Report 1-94-09 above, the licensee believed that the set pressure deviations were a result of the interaction of the spring, the upper spring washer, and the nose of the adjusting bolt as they pivot. The licensee stated that they had deferred performance of a valve modification, which they believed would correct the problem, due to unavailability of a testing facility for the modified valve configuration.

b. Followup Inspection

The inspector reviewed the licensee event reports noted above. The inspector discussed the pressurizer safety valve problems and corrective actions noted above with the licensee. The licensee discussed an extensive history of actions taken to attempt to identify and resolve the pressurizer safety valve setpoint problems that had been experienced. The actions were predominantly documented in Nonconformance Report N0001328. The licensee actions included several initiatives aimed at refining the valve testing and setting methodology. The licensee studied and determined that control of ambient and loop seal temperatures resulted in closer setting and testing of the valves. Similarly, the licensee determined that the effect of setpoint spread could be minimized by adjusting the valve as close as possible to the medium set point. The licensee considered that the results of changes to the setting and testing methodology would be demonstrated during future testing of the valves during upcoming outages.

The licensee also informed the inspector that the additional actions they had performed to resolve the problem included a review of the nozzle loadings of the valves to correct any effects nozzle loads had on the valve setpoints. In addition, the licensee disassembled, inspected, and evaluated the valve internals to determine if other conditions, such as disc material, could be contributing to the problem. No other modifications to the valve internals, other than the previously discussed modification, were identified as possible corrective actions. Finally, the licensee also reviewed relaxation of the Technical Specifications as-found setpoint tolerance from +/- 1 percent to +/- 3 percent, but determined that the relaxed tolerance would not be within the design basis. The licensee was reviewing relaxation of the tolerance to a value less than +/- 3 percent, that would be within the design basis, to determine if the relaxation would be of benefit to as-found setpoint conditions that have been experienced.

As reported in Licensee Event Reports 1-94-09 and 1-95-16, the as-found out-of-tolerance pressurizer safety valve setpoints were violations of Technical Specification 3.4.2.2. The inspector determined that the licensee-identified violation of the Technical Specifications could not have been reasonably prevented by licensee corrective actions for a previous violation or a previous licensee finding that occurred within the past 2 years of this NRC inspection. The licensee had attempted several reasonable corrective actions to preclude recurrence of the problem. The inspector determined that the out-of-tolerance setpoint conditions were corrected, i.e., the valves were reset and tested satisfactorily. The violations did not appear to be willful in nature. The licensee identified and corrected violations are being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-275/9707-01).

E8.3 Battery Charger Printed Circuit Board Open Items

The inspector reviewed the following two open items involving problems on electrical circuit boards. The details and inspection effort is discussed in the following sections:

(Closed) Inspection Followup Item 50-275/9422-01: Licensee Identified Missing Solder on Printed Circuit Boards.

(Closed) Licensee Event Report 50-275/94-262: Licensee Event Report 94-18, Revision 0, Defects in Battery Charger Printed Circuit Boards.

a. Background

On July 21, 1994, during preparations for the performance of post-maintenance Surveillance Test Procedure M-16PI, "Continuity Testing of Train A/B Slave Relays K627, K628 and K635," licensed operating personnel observed that the vital dc switchgear bus voltage was above the maximum of 136 volts permitted by Surveillance Test Procedure M-16PI. Battery charger BTC 1 output voltage was noted to be 138.9 volts. The licensee determined that the high output voltage was caused by a defect in the voltage control module printed circuit board. Licensee examination of the printed circuit board identified a missing solder connection at a capacitor lead. The mechanical connection between the capacitor lead and the board had previously been adequate for proper battery charger operation for 3 years. Over a period of time, the electrical connection had degraded. This degraded connection resulted in the abnormal high voltage condition on the battery charger.

Although the charger was functional, the licensee placed the battery bank on the backup charger, BTC 121, within 2 hours and the Technical Specification 3/4.8.2, 14-day action statement was entered as a precautionary measure. The licensee then replaced the voltage control module and BTC 121 was returned to service. A preliminary engineering evaluation of the missing solder connection concluded that the connection would have been functional following a seismic event. However, the licensee decided to conservatively consider the connection nonfunctional

following a seismic event since actual seismic testing had not been performed on a printed circuit board with this type of a defective connection.

On July 29, 1994, the licensee's inspection of the spare warehouse battery charger printed circuit boards identified four circuit boards with soldering deficiencies. Three of the deficiencies were partially soldered connections. The fourth deficiency was a missing solder connection similar to that found in BTC 121. The licensee determined that although the partially soldered connections did not meet soldering quality standards, they were capable of providing an adequate electrical connection. As discussed above, a printed circuit board with connections that did not have any solder was conservatively considered as not being capable of performing its intended function following a seismic event.

Since printed circuit board connection soldering deficiencies had been identified in an operating battery charger and on four spare printed circuit boards in the warehouse, the licensee decided to inspect all installed battery charger control module printed circuit boards.

- On August 26, 1994, the licensee found a filtering module for BTC 21 that had inadequate solder coverage on an isolation transformer lead. The control module was replaced. The licensee determined that the electrical and mechanical connection was sound and that battery charger operation was not affected by this condition.

- On August 27, 1994, the licensee found a filtering module for BTC 221 with an unsoldered diode lead. In addition, an amplifier module was found with excessive solder between the base and collector of a transistor. The charger output was not affected by these conditions. The licensee replaced both modules. Upon initial startup of BTC 221 for post-maintenance testing, there was no output voltage. A licensee investigation identified a deficient solder connection on one of the two modules that had just been replaced. An initial licensee visual inspection of the connection determined the connection to be good, but a more detailed visual inspection found that the lead was not making electrical contact. The licensee replaced the board and the charger was successfully tested and returned to service.

- On August 31, 1994, the licensee found a partially soldered connection on the control module for BTC 22. The licensee replaced the module. The licensee evaluation of the condition concluded that the partially soldered connection had no effect on charger operation.

On August 31, 1994, the licensee's technical review group for the identified problems determined that the number of deficiencies found on the control module printed circuit boards could have prevented the fulfillment of a safety function (i.e., maintain the reactor in a safe shutdown condition). On August 31, 1994, at 5:45 p.m. PDT, the licensee made a 4-hour, nonemergency report in accordance with 10 CFR 50.72(b)(2)(iii)(A). On September 30, 1994, the licensee submitted Licensee Event Report 1-94-18-00 to provide a written report of the problem.

The licensee performed the following corrective actions to resolve the identified conditions.

- The affected battery was switched to the installed spare battery charger and electrical maintenance investigative and corrective actions were initiated.
- An inspection plan for all safety-related battery charger printed circuit boards was implemented.
- A review of the effectiveness of commercial grade dedication processes for other vendor printed circuit boards utilized in safety-related applications was performed and found satisfactory based upon a warehouse stock sampling inspection and a problem report history search.
- All installed battery charger printed circuit boards were inspected and any deficiencies had been repaired.

In addition, the licensee performed the following corrective actions to prevent recurrence.

- The receipt inspection criteria was enhanced to provide specific attention to soldered connections on printed circuit boards for all future procurement of battery charger printed circuit boards.
- The existing stock of battery charger printed circuit boards were re-inspected to enhanced criteria specific to soldered connections and any deficiencies were immediately placed on hold for repair.

Prior to completion of licensee evaluations of the battery charger circuit board problems, the NRC resident inspectors performed a followup inspection of the problem and noted that the boards were commercial-grade items that the licensee dedicated for use in safety-related applications. The inspectors considered the commercial-grade dedication activities for the boards were potentially less than adequate and determined that further followup inspection of the completed licensee evaluations was warranted. The inspectors identified this as a followup item (50-275/9422-01).

b. <u>Followup Inspection</u>

The inspector reviewed and discussed the reported problem and corrective actions with the licensee. The licensee discussed Nonconformance Report N0001842, which the licensee issued as the main corrective action document for the problem. The nonconformance report, and associated referenced action requests were reviewed by the inspector. The inspector also reviewed Quality Control Procedure QCP 10.1, "Receipt Inspection Program," Revision 1.

The inspector determined that the licensee, as reported, expanded the inspection of battery charger circuit boards to identify any other defective or questionable solder connections. The licensee also performed a sampling inspection of circuit boards in the warehouse and determined that the identified solder condition could be isolated to one manufacturer. The licensee inspected all circuit boards from that manufacturer. The inspector determined that the licensee's receiving inspection procedure had a requirement for inspecting received items for physical damage and workmanship. The licensee informed the inspector that, in addition to the procedural requirements, a standard clause for receiving inspection of circuit boards from the manufacturer in question was added to the procurement document. Standard Clause 4534, "Receiving Instruction - PC board," provided the inspection attributes and acceptance criteria for the inspection.

The inspector was informed by the licensee their current practice was to perform the specified inspection on all circuit boards. The inspector questioned the licensee regarding the application of the inspection by specifying it in the receiving inspection procedure. The licensee acknowledged that the procedure would be changed to incorporate the circuit board solder receiving inspection requirements in the procedure for all circuit boards.

The inspector concluded that the licensee had appropriately identified and reported the circuit board solder problem. The inspector concluded that the licensee had performed reasonable corrective actions; therefore, these items are closed.

E8.4 <u>(Closed) IFI 50-275/9427-01: Undersize Springpack for Auxiliary Feedwater Motor-Operated Valve LCV-107</u>

a. <u>Background</u>

During the May 1994 Unit 1 Refueling Outage 1R6, the licensee replaced the actuators for Motor-Operated Valves 1-LCV-106 and 1-LCV-107 with new model actuators. These are the Unit 1 turbine-driven auxiliary feedwater pump discharge level control valves. During setup for installation of valve 1-LCV-106, the licensee identified that the valve had low thrust values due to an incorrect spring pack in the actuator. On May 4, 1994, the actuator vendor (Limitorque) confirmed that four actuators supplied to the licensee on Purchase Order 063868 may not have conformed to the purchase specification. They were potentially configured with 0101-099 (extra light) belleville spring packs instead of the 0101-091 spring packs specified. In addition, limitorque informed the licensee that size 1.5 torque switch limiter plates were supplied with the actuators instead of the specified size 3.5. The licensee had installed two of the actuators on motor-operated valves 1-LCV-106 and 1-LCV-107, and two were spare actuators.

The licensee repaired Valve 1-LCV-106 and installed the correct parts. The two spare actuators were disassembled on May 5, 1994, and the incorrect size spring packs were also found to he installed on the both spare actuators. Further licensee discussions with Limitorque determined that Limitorque had not recognized the changes in the purchase order. The licensee had ordered 2-foot-pound motors with

5-foot-pound actuators. The licensee intended to use the 5-foot-pound motors that were in stock with the ordered 5-foot-pound actuators. The 2-foot-pound motors would be used as spare motors. Limitorque incorrectly supplied the specified 2-foot-pound motors with the corresponding extra light spring pack and size 1.5 limiter plate in the actuators.

The licensee documented the evaluations performed for the reported problem by Nonconformance Report N0001866. On May 4, 1994, the licensee performed a prompt operability assessment for valve 1-LCV-107. The licensee reviewed the "VOTES" test traces for valve 1-LCV-107 and compared the traces with those for similar valve with known proper spring packs (i.e., valves 1-LCV-108 and 1-LCV-109). The licensee determined that the performance of all three valves were as expected and the valve 1-LCV-107 performance closely resembled those of valves with an actuator with a 0101-091 proper spring pack. The thrust of valve 1-LCV-107 was within the setpoint window and within the limiter plate capabilities. Nuclear Engineering Services concluded that the observed acceptable performance could not have been obtained with the improper spring pack. The licensee determined that valve 1-LCV-107 was fully functional with no degraded condition. However, the prompt operability assessment also stated that the spring pack be tested at the next maintenance opportunity to ascertain the spring pack size. Unit 1 completed Refueling Outage 1R6 on May 7, 1994, and resumed full power operation on May 27, 1994.

On May 27, 1994, the site review group evaluated the condition for reportability in accordance with 10 CFR Parts 50.72, 50.73, and 21. The review group determined the condition to not be reportable, but recommended that the "VOTES" testing of valve 1-LCV-107 be increased to coincide with the surveillance test of the valve. On November 9, 1994, the licensee performed "VOTES" testing on valve 1-LCV-107. The actuator output thrust was found to be 1626 pounds, which was less than the minimum required thrust of 2656 pounds. The licensee reviewed the test data and determined that the motor loading was consistent with an extra light spring pack, rather than the initially believed light spring pack.

On November 9, 1994, the licensee initiated Action Request 0358065 to evaluate the as-found thrust of valve 1-LCV-107. The evaluation was completed on November 11, 1994. The licensee determined that the as-found thrust of valve 1-LCV-107 was still sufficient for the valve to be capable of performing its required safety function during the period of operation with the incorrect spring pack. The licensee replaced the spring pack and limiter plate for valve 1-LCV-107.

An NRC inspection was performed in November 1994 and was documented in Inspection Report 50-275;-323/94-27. During the inspection, the inspector noted the valve 1-LCV-107 problems and licensee evaluations discussed above. The inspector determined that further review of the licensee resolution of the issue, documented in Nonconformance Report DRI-94-EM-N054, was warranted.

b. Followup Inspection

The inspector discussed the above noted problems and licensee evaluations associated with valve 1-LCV-107 with the licensee. The inspector determined that the appropriate nonconformance report for the noted condition was DC1-94-EM-N054, which was also Nonconformance Report N0001866 in the licensee's current corrective actions system.

The inspector reviewed the licensee's 1-LCV-107 operability evaluation and calculations associated with Action Request 0358065. The inspector found that the licensee removed conservatisms in the minimum required thrust calculation and was able to show that the valve would be capable of performing its safety function with a 1345-pound thrust, which was within the 1626 as-found thrust of the valve with the extra light spring pack. The inspector determined that the licensee's calculations were reasonable.

The inspector discussed with the licensee the incorrect "VOTES" test results which was used initially by the licensee for determining that valve 1-LCV-107 was functional despite indications that an incorrect spring pack may have been installed in the actuator. The licensee documented the evaluations and analysis that were performed to determine the cause of the incorrect "VOTES" test results in Nonconformance Report N0001866. The licensee determined that the test of valve 1-LCV-107 was performed using a mini-C clamp calibrator and that the valve had a two-piece stem design. The licensee determined through the evaluations and analysis that the "VOTES" test equipment was position sensitive for the split-stem design and the proper placement of the mini-C clamp calibrator influenced the test results. The licensee concluded that the two-piece stem and the extremely tight working conditions for the placement of the mini-C clamp calibrator on the stem all combined to produce anomalous behavior of the calibrator.

The licensee reviewed all other calibrations that had been performed using the same mini-C clamp calibrator and identified no other anomalies. The licensee reviewed data for all other similar valves in both units that were calibrated with other mini-C clamps. The data for valve 2-LCV-106 was determined to be questionable and the valve was tested. The licensee found lower than expected thrust for the valve and reset the actuator.

The licensee revised electrical Maintenance Procedure MP E-53.10V, "MOV Switch Setting and Diagnostic Testing," to change the calibration method for all future "VOTES" test of split stem design valves. The inspector reviewed Procedure MP E-53.10V, Revision 16, and confirmed the change to the calibration method required by the procedure.

The inspector concluded that the licensee had ultimately implemented adequate corrective actions for the spring pack problems identified in valve 1-LCV-107. However, the inspector also concluded that the licensee did not perform adequate and prompt corrective actions for the incorrectly sized spring pack that was supplied for valve 1-LCV-107 in May 1994. The inspector determined that the

inadequate corrective action was a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." The inspector determined that the incorrect spring pack and thrust washer for valve 1-LCV-107 had been replaced. The licensee identified violation did not appear to be willful in nature. The licensee identified and corrected violations are being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-275/9707-02).

V. Management Meetings

X1 Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of the inspection on May 2, 1997. The licensee acknowledged the findings presented.

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

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

S. Allen, Supervisor, Valve Engineering
S. Chesnut, Supervisor, Primary Systems Engineering
D. Cook, Supervisor, Materials Procurement Design Engineering
D. Elsalaymeh, Senior Engineer, Materials Procurement Design Engineering
T. Fetterman, Director, Instrumentation and Control Engineering
T. Grebel, Director, Regulatory Services
M. Jacobson, Supervisor, Nuclear Quality Services
S. Ketelsen, Senior Engineer, Regulatory Services
R. Kinports, Procurement Specialist, Materials Procurement Design Engineering
D. Kornberg, Procurement Design Engineer, Materials Procurement Design Engineering
D. Miklush, Manager, Engineering Services
D. Oatley, Manager, Maintenance Services
L. Pulley, Senior Engineer, Design Engineering Services
D. Taggart, Director, Nuclear Quality Services
R. Thierry, Acting Director, Licensing and Design Basis
D. Vosburg, Director, Nuclear Steam Supply Systems Engineering

NRC

M. Tschiltz, Senior Resident Inspector

INSPECTION PROCEDURES USED

IP 92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92903	Followup - Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-275/9707-01	NCV	Pressurizer Safety Valve As-found Setpoint Exceeds Technical Specifications, Section E8.2
50-275/9707-02	NCV	Inadequate Corrective Actions - LCV 107 Actuator installed and Operated With Undersize Springpack, Section E8.4

Closed

50-275/95-275	LER	Licensee Event Report 94-09, Revision 2, Pressurizer Safety Valve Setpoint Spread, Section E8.2
50-275/9422-01	IFI	Licensee Identified Missing Solder on Printed Circuit Boards, Section E8.3
50-275/9427-01	IFI	Undersize Springpack for AFW MOV LCV-107, Section E8.4
50-275/94-129	LER	Licensee Event Report 94-09, Revision 0, Pressurizer Safety Valve Setpoint Spread, Section E8.2
50-275/94-262	LER	Licensee Event Report 94-18, Revision 0, Defects in Battery Charger Printed Circuit Boards, Section E8.3
50-275/94-300	LER	Licensee Event Report 94-03, Revision 1, Main Steam Safety Valve As-found Setpoint Exceeds Technical Specifications, Section E8.1
50-275/95-101	LER	Licensee Event Report 94-09, Revision 1, Pressurizer Safety Valve Setpoint Spread, Section E8.2
50-275/95-277	LER	Licensee Event Report 95-16, Revision 0, Technical Specifications not Met During Pressurizer Safety Valve testing, Section E8.2
50-275/96-180 01014	VIO	Violation Involving Augmented Testing Issued in Inspection Report 96-012, Section E8.1
50-275/96-180 02014	VIO	Violation for Main Steam Safety Valve's Out-of-Tolerance High Issued in Inspection Report 96-012, Section E8.1
50-275/96-205	LER	Licensee Event Report 96-03, Revision 0, Technical Specifications not Met During Main Steam Safety Valve Testing, Section E8.1
50-275/96-243	LER	Licensee Event Report 96-13, Revision 0, Main Steam Safety Valve's Set Outside Technical Specifications, Section E8.1
50-323/94-301	LER	Licensee Event Report 94-07, Revision 0, Main Steam Safety Valve's Exceeded Technical Specifications Tolerance for Lift Settings, Section E8.1
50-323/96-244	LER	Licensee Event Report 96-07, Revision 0, Technical Specifications not Met Due to High Initial Main Steam Safety Valve Lift Setpoints, Section E8.1
50-275/9707-01	NCV	Pressurizer Safety Valve As-found Setpoint Exceeds Technical Specifications, Section E8.2

50-275/9707-02	NCV	Inadequate Corrective Actions - LCV 107 Actuator installed and Operated With Undersize Springpack, Section E8.4