APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-275/94-07
50-323/94-07

Operating Licenses: DPR-80
DPR-82

Licensee: Pacific Gas and Electric Company
Nuclear Power Generation, B14A
77 Beale Street, Room 1451
San Francisco, California 94177

Facility Name: Diablo Canyon Units 1 and 2

Inspection At: Diablo Canyon Site, San Luis Obispo County, California

Inspection Conducted: February 17 through March 19, 1994

Inspectors: M. Miller, Senior Resident Inspector
M. Tschiltz, Resident Inspector

Accompanying Personnel: J. Winton, Inspector Intern, NRR

Approved By: D. F. Kirsch 4/1/94
D. Kirsch, Chief — Date Signed
Reactor Projects Branch 1

Inspection Summary

Areas Inspected (Units 1 and 2): Routine, announced, resident inspection of onsite followup of events, operational safety verification, plant maintenance, surveillance observations, quality assurance, followup on corrective actions for violations, other followup, and in-office review of licensee event reports.

Results (Units 1 and 2):

Strengths:

- A request for enforcement discretion appeared to have been well implemented.
- Nuclear Quality Services appeared to have performed a well focused and appropriately critical assessment of the major licensee organizations.

Weaknesses:

- Operators did not properly vent the reactor vessel head, resulting in an RCS level change of about six feet during the RCS draindown evolution.

Summary of Inspection Findings:

- Violation 50-275/94-07-01 was opened (Section 2.1).
- Violations 50-323/93-16-01, 50-323/93-16-02 and 50-275/93-22-01 were closed (Section 7).
- Followup Items 50-275/92-31-03 and Unresolved Item 50-275/93-34-03, were closed (Section 8).
- Licensee Event Reports 50-275/94-01, Revision 0 and 50-275/94-03, Revision 0 were closed (Section 9).

Attachments:

- Attachment 1 - Persons Contacted and Exit Meeting
- Attachment 2 - Acronyms

DETAILS

1 PLANT STATUS

1.1 Unit 1

The Unit operated at 100 percent of rated thermal power until March 11, when the Unit was shut down for a scheduled refueling outage. The Unit is currently in Mode 6.

1.2 Unit 2

The Unit operated at 100 percent power except for February 24, 1994 to February 26, 1994 and March 6, 1994 to March 7, 1994, during which power was reduced to 50 percent to clean condensers.

2 OPERATIONAL EVENTS (93702)

2.1 Failure to Properly Vent the Reactor Vessel Head During Unit 1 Reactor Coolant System (RCS) Draindown

On March 14, 1994, during routine draindown of the Unit 1 RCS from a full system, to the level just below the reactor vessel flange, in preparation for refueling, an abrupt change in RCS level occurred. When an indicated level of 111.5 feet was reached, both trains of level indication rapidly increased to an indication of 117.9 feet. Draining operations were halted to determine the cause.

Operators determined that the step in the drain down procedure to vent the reactor vessel head by cross-tieing the level indication reference legs had been performed improperly. This step cross-ties the reactor vessel head volume with the top of the pressurizer to provide a common reference leg, thus assuring a consistent level reading between the two channels of reactor vessel level indication, and level equalization between the reactor vessel and the RCS loops. The operators had proceeded to cross-tie the level indication between the pressurizer and the reactor vessel head by performing the required valve manipulations concurrent with performing RCS draining operations. The control room staff had not recognized the need to complete the cross-tie evolution before proceeding with the reactor vessel draindown. Completion of the cross-tie evolution not only placed the narrow range level indication in service, but also provided a common reference leg for the reactor vessel level indication channels, and a vent path for the reactor vessel head.

As a result, when the required valve line up to cross tie the vents was completed in containment, an abrupt increase in the indicated level was observed and water levels were allowed to equalize between the reactor vessel and the pressurizer, causing the indicated level transient. The event had no impact on decay heat removal. Draining operations continued after the root cause of the level change had been determined.

The procedure, OP A-2:II, Revision 11 (XPR), "Reactor Vessel - Draining the RCS to the Vessel Flange - With Fuel in the Vessel", Step 6.4, stated "Place

the NR RVRLIS and the vent cross-tie in service as follows:...NOTE: The following step aligns the PRT nitrogen supply to the both the Pressurizer and the Reactor Vessel Head through the cross-tie." The following step, 6.4.2, implemented alignment of the cross-tie, and noted that the Narrow Range Reactor Vessel Refueling Level Indication System (NR RVRLIS) was not required at that RCS water level.

Operators had assumed that, since the NR RVRLIS was not yet required, draining operations allowed by Step 6.5, could continue in parallel with performance of Step 6.4, which placed the NR RVRLIS and vent cross-tie in service.

The above failure to properly perform Procedure OP A-2:II, which resulted in incorrect level indication and an abrupt change in RCS level, is considered a violation of Technical Specification 6.8.1., which requires that procedures be implemented governing the activities recommended in Appendix A of Regulatory Guide 1.33, Revision 2, including procedures involving draining and refilling of the recirculation loops, and the reactor vessel. (Violation 50-275/94-07-01).

2.2 Offset of Main Steam Safety Valve Lift Setpoints and Notice of Enforcement Discretion

On March 9, 1994, the licensee was notified by a vendor (Fermanite) that the method used to test main steam safety valve (MSSV) lift setpoints during power operations was potentially inaccurate. The vendor identified that the seat lift area coefficient used to set the valves may have been incorrect, resulting in the MSSVs lift setpoints being set about two percent higher than the intended setpoint. At 6:00 p.m. on March 12, 1994, after evaluation and analysis, the licensee determined that the issue was applicable to Diablo Canyon. Because TS 3.7.1.1 requires MSSVs to be set within plus or minus one percent of the design setpoint, all MSSVs were declared inoperable. At that time, the licensee requested that the NRC grant enforcement discretion; to not enforce TS 3.7.1.1, which would have required plant shutdown within 6 hours, while the licensee obtained test equipment and performed lift setpoint testing.

The NRC granted the request, documented in an NRC letter dated March 15, 1994. The licensee's request was well focused and provided all required information to evaluate enforcement discretion. The licensee completed testing and resetting of the valves, returning them to operable status on March 15, 1994.

2.3 Main Steam Safety Valve Drift

A separate issue concerning the MSSVs involved unexpected random drift of the lift setpoint. During several days in February, 1994, routine surveillance testing showed that most of the 20 Unit 1 Main Steam Safety Valves (MSSVs) had experienced apparently random lift setpoint drift of up to 5 percent. The licensee reset each MSSV lift setpoint to within plus or minus one percent of the required lift setpoint after each lift test, as required by TS. Although some setpoint drift had been identified in the past, drift was typically no greater than 3 percent of the lift setpoint. As a result of the findings on Unit 1, the licensee then tested the Unit 2 MSSVs, and found that similar

drift had occurred. On March 4, 1994, testing of the Unit 2 valves was completed.

Safety Significance To address past, less severe MSSV setpoint drift, the licensee had performed analysis which concluded that drift of up to plus or minus three percent of the MSSV lift setpoint had no effect on safety. The licensee then performed similar analysis of the as-found condition of the MSSVs, and concluded that the as-found Unit 1 and 2 setpoints would not have had an adverse effect on any design basis events. The inspector concluded that the licensee's analyses provided adequate assurance the observed drift in setpoints would have minimal effect on any design basis events.

2.4 Failure of a Steam Generator Level Transmitter Seal Resulting in Non-Conservative Protection System Bistable Response

On March 11, 1994, operators observed Steam Generator Level Transmitter LT-519 fail high without causing the expected resultant reactor protection system bistable trip. Operators promptly tripped the associated reactor trip bistable. The subsequent investigation and repair work determined that a conduit seal associated with providing the required Equipment Qualification for the level transmitter had failed. Both the conduit seal and the transmitter were manufactured by Rosemount. The licensee determined that a short circuit had occurred in the seal and initiated a Quality Evaluation to track the root cause investigation. The seal was sent to the vendor (Rosemount) for failure analysis. Neither licensee nor Rosemount investigations have identified the cause of the failure to date, nor have similar failures been identified in industry. The licensee's investigation and analysis are continuing.

The failure of the bistable to trip was determined to have been a result of exceeding the protection set comparator design maximum voltage of 15 volts. The short circuit in the conduit seal resulted in the current path bypassing the current limiter in the transmitter. As a result, the comparator was subject to a voltage spike over the 15 volts design maximum rather than typical voltage of less than 6 volts. When the comparator was subjected to a voltage spike of over it's design maximum, a non-conservative response, failure of the bistable to trip, occurred. The vendor manual for the over ranged component in the comparator, an operational amplifier, also stated that valid responses were not expected over the 15 volt design maximum.

The licensee determined that, because redundant and diverse reactor protection trains were available, the SSPS safety function would have been satisfied under design basis conditions. The inspector agreed, with the exception of the concerns noted below.

The inspector expressed concern regarding a common mode failure vulnerability of the existing SSPS comparators to fail in a non-conservative manner. The licensee pointed out that an NRC safety analysis had been performed to evaluate the common mode failure of the protection set functions of the planned Westinghouse Eagle 21 SSPS upgrade, to be installed in both units. From a functional standpoint, the Eagle 21 failure would bound the concern for a common mode comparator failure identified by the inspector; that is, the

failure of all trains of a trip function at the 7100 analog Hagan rack protection sets. The NRC safety analysis associated with NRC License Amendments 84 (Unit 1) and 83 (Unit 2) concluded that the analysis of protective function diversity to the Eagle 21 upgrade provided reasonable assurance that adequate diversity exists in diverse functions to mitigate events.

Since the SSPS in both units will be replaced with the Westinghouse Eagle 21 solid state protection system upgrade, the licensee investigated the effect of a similar voltage spike on the new system. Both analyses of the design and testing using the maintenance training installation determined that Eagle 21 could withstand up to 32 volts and continue to respond in a conservative fashion (by causing a high bistable trip). Analysis and discussions with Westinghouse determined that up to 120 Volts could be applied to the current loop without causing the Eagle 21 upgrade system to fail in a non-conservative manner. The licensee's responses resolved the inspector's concern in this area.

The inspector expressed concern that this failure of the bistable to trip when subjected to a voltage spike may additionally be a generic concern for Westinghouse plants which may not have the same level of diversity of reactor protection functions as Diablo Canyon. The licensee stated that Westinghouse had been informed both directly and by industry information issued by the licensee, and was evaluating the failure for applicability to other Westinghouse plants. The inspector had no further concerns in this area.

2.5 Conclusions

For the issues associated with MSSVs, the inspector reviewed the licensee's safety analyses and operability evaluation. The licensee's conclusions appeared valid; that no adverse effects on any design basis events would have occurred due to the MSSV setpoints.

For the non-conservative response of the level transmitter bistable, the inspector reviewed the NRC staff safety evaluation, and, based in part on that evaluation, concluded that the licensee's assessment appeared valid. The concerns for Westinghouse generic vulnerabilities have been transmitted by the licensee to Westinghouse and to the industry.

3 OPERATIONAL SAFETY VERIFICATION (71707)

3.1 Operator Logs

The inspector noted that the log entries for entry into the Technical Specification action statements associated with the MSSV Notice of Enforcement Discretion discussed above were incorrect. The error involved the documentation of time of entry into the action statements, which was several hours too early. The licensee stated that the operator recording the action statements listed the time that the operability evaluation was initiated, rather than when the evaluation was concluded. The licensee promptly corrected all log errors.

Based on the low safety significance of the errors, the inspector considered the licensee's action adequate. The inspector will continue to monitor the licensee's performance in this area as part of routine inspection activities.

3.2 Conclusion

Examples of minor log errors were identified by the inspector and were promptly corrected by the licensee. The licensee's response was appropriate.

4 PLANT MAINTENANCE (62703)

During the inspection period, the inspector observed and reviewed selected documentation associated with maintenance and problem investigation activities listed below to verify compliance with regulatory requirements, compliance with administrative and maintenance procedures, required quality assurance/quality control department involvement, proper use of safety tags, proper equipment alignment and use of jumpers, personnel qualifications, and proper retesting.

Specifically, the inspector witnessed portions of the following maintenance activities:

Unit 1

- Diesel fuel oil suction cross-tie piping replacement (both units)
- Diesel fuel oil discharge cross-tie piping replacement (both units)
- Replacement of the main annunciator system with a temporary annunciator system in preparation for the Unit 1 annunciator system upgrade

Unit 2

- Temporary diagnostic modification (jumper) to Inverter IY 2-2
- Repair of Steam Generator Level Transmitter LT-519

4.1 Lack of Documentation of Seismic Interaction Analysis for Temporary Modification

During a routine plant walkdown, the inspector observed that a temporary modification (jumper) to add monitoring capability to the IY 2-2 inverter had been placed within a foot of the safety related inverter IY 2-2, and a jumper (No. 93060) had been installed to allow monitoring of the inverter performance. Since the monitoring equipment, an oscilloscope, had not been physically restrained, the inspector raised the question of possible seismic interaction between the monitoring equipment and the inverter, a seismic target. The jumper safety analysis required by 10 CFR 50.59 did not include a seismic interaction analysis. The inspector identified three aspects of potential seismic interaction which did not appear to have been systematically

addressed in the safety analysis: 1) the force transferred to the frame of the inverter from a collusion with the monitoring device; 2) potential interaction with devices directly mounted inside the location on the panel which the monitoring device could contact; and, 3) the potential weakening of the inverter frame structure resulting from a gap created by wiring which was routed under the side panel of the inverter, resulting in a fraction of an inch separation of the side panel from the lower foundation panel.

Discussions with the licensee's onsite design engineering group addressed the inspector's concerns. The potential for seismic interaction was not a concern since no relays or electrical devices were in direct contact with the panel which may come in contact with the monitoring device. The frame structure of the inverter was expected to have sufficient design margin to withstand contact with the monitoring device during a design basis seismic event with no damage. The routing of the wiring under the side panel, causing a gap, did not significantly weaken the inverter's structural integrity. Based on further detailed discussions, the inspector determined that the licensee's conclusions were valid.

Interviews with members of the Plant Staff Review Committee (PSRC), revealed that the evaluation of seismic interaction had been discussed during PSRC evaluation of the jumper, and the PSRC had concluded that no interaction was likely, based on engineering judgement and familiarity with seismic requirements.

The licensee subsequently modified the jumper procedure checklist to include a requirement for documentation of applicable seismic interaction analysis.

4.2 Installation of Temporary Annunciator During Main Annunciator System Replacement

On March 14, the inspector observed the licensee transition from the Unit 1 main annunciator to a temporary annunciator to support replacement of the main annunciator. The inspector questioned the licensee regarding the procedure that would be followed if the temporary system failed. The licensee stated that, upon loss of the temporary annunciator, although the annunciator response procedure required only that the annunciator be returned to service, and that important parameters be monitored, the licensee would also stop all important operations, and closely monitor selected plant parameters while the annunciator system was returned to service.

About 3 hours after the transition to the temporary system, the temporary system malfunctioned. Licensee troubleshooting found that a cable connector for one of the temporary annunciator computers was loose. During the troubleshooting while the system was inoperable, the licensee stopped all evolutions which could result in changes to existing plant conditions. Additional operators were stationed to monitor plant parameters such as RHR pump performance, RCS temperature, RCS level, and other significant indicators. The annunciator was returned to service within 30 minutes, all connectors for the temporary annunciator system were verified, and access to areas adjacent to the temporary annunciator components were restricted from routine personnel traffic.

4.3 Conclusions

The licensee's actions to document the seismic safety analysis, and to respond to the annunciator system failure were examined and appeared appropriate.

5 SURVEILLANCE OBSERVATIONS (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 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.

Specifically, portions of the following surveillances were observed by the inspector during this inspection period:

Unit 1

- Analog Channel Operational Test Nuclear Source Range

Unit 2

- ASW System Flow Monitoring
- Routine Surveillance Test of Auxiliary Saltwater Pumps

Conclusions

The inspector concluded that the surveillance tests appeared to have been performed in an acceptable manner.

6 Review of Quality Activities (40500)

6.1 Quality Performance Assessment Report

The inspector reviewed the recently issued Quality Performance Assessment Report issued by Nuclear Quality Services to determine if the report was appropriately probing and critical of problem areas and potential problem areas in various licensee organizations. In this report, the Nuclear Quality Services organization evaluated safety performance of the various licensee organizations. Current problem areas, trends and corrective action for problem areas were reviewed. The report concluded that both site and corporate office engineering organizations, as well as the Nuclear Quality organizations, were in need of additional focus in complying with procedural requirements and in implementing problem ownership. The other organizations evaluated, including Operations and Maintenance, were considered to have adequate identification and correction of problems, although problems in personnel methods were noted.

In summary, the concern for appropriate implementation of personnel methods, following existing instructions, and ownership of problems, appeared to be the major concerns. The report also made note of minor equipment management problems, and concluded that few problems of great significance had occurred.

6.2 Nuclear Safety Oversight Committee Meeting

On March 16, 1994, the inspector observed portions of a meeting of the Nuclear Safety Oversight Committee (NSOC) in order to evaluate NSOC involvement and focus on plant safety and management performance. Recent issues were discussed, such as operational events, concerns identified by NRC inspections, quality assurance audits, and Independent Safety Evaluation Group evaluations. Discussion focused on root causes of the above issues, as well as commonalities of problems and corrective actions for the above issues. The discussions observed appeared to have been appropriately responsive to safety and management issues.

6.3 Conclusions

The samples of the licensee's quality functions, which were inspected as noted above, appeared to have been performed in a probing, critical, and well directed manner.

7 FOLLOWUP ON CORRECTIVE ACTIONS FOR VIOLATIONS (92702)

7.1. (Closed) Violation 50-323/93-16-01: Reactivity Increase While Shutdown, Due to Boron Dilution by Mixed Bed Demineralizer Return To Service

This violation involved a situation wherein the licensee allowed a boron dilution of the RCS to occur while shut down, as a result of not properly implementing corrective actions for lessons learned from industry experience. By a letter dated August 30, 1994, the licensee responded to the issue by performing corrective actions which included training on lessons learned for Operations personnel, correction of procedures to implement industry standard corrective actions (such as placing administrative tags on demineralizers regarding boron concentration, saturating demineralizer beds with boron prior to placing them in operation, checking boron concentrations of unused demineralizer vessels) and other procedure changes. The licensee also reviewed industry experience in addition to boron dilution concerns to determine if other areas of industry experience had been properly reviewed and applied to the licensee's operations. The inspectors examined the licensee's corrective actions and implementation, and concluded that the actions appeared to have been appropriate and acceptably implemented.

7.2 (Closed) Violation 50-323/93-16-02: Reactivity Increase At Power Due to Cation Demineralizer Bed Return to Service

This violation involved a situation wherein the licensee allowed a boron dilution of the RCS to occur while at power, soon after a startup, as a result of not properly implementing corrective actions for lessons learned from industry experience. By a letter dated August 30, 1994, the licensee

responded to the issue by performing corrective actions, which included training on maintaining a questioning attitude, the importance of consulting all affected groups before taking an action, documenting intended actions, correction of procedures to specify conditions, such as boron concentrations, that existed when the demineralizer bed was last in service, and other procedure changes. As discussed above, the licensee also reviewed industry experience in addition to boron dilution concerns to determine if other areas of industry experience had been properly reviewed and applied to the licensee's operations. The inspectors examined the licensee's corrective actions and implementation, and concluded that the actions appeared to have been appropriate and acceptably implemented.

7.3 (Closed) Violation 50-275/93-22-01: Inadequate Control of Plant Lubricants

This issue involved the addition of incorrect types of oil to safety related pumps. The most recent occurrence involved a mixture of oil to an ASW pump. The licensee's corrective actions included tighter control of plant lubricants by issuing lubricants at an issue control point, training of operators and maintenance staff regarding labeling and control of lubricants, color coding of lubricant containers, and reduction and elimination of several small lubricant storage bottles used throughout the plant. The inspector examined the licensee's corrective actions and concluded that the actions appeared appropriate and acceptably implemented.

8 FOLLOWUP OF OPEN ITEMS (93702)

8.1 (Closed) Followup Item 275/92-31-03: Lack of ASME Required Drain Lines for Relief Valve Tailpipes

The inspector identified that several of the safety related relief valve tailpipes which direct the valve discharge in a vertical direction did not have drains as required by ASME code. The licensee had pursued a decision by the ASME code committee to accept the valves without drains, as installed, but later decided to discontinue pursuit of a code committee ruling, and initiated an NCR to install drains in the tailpipes. Mr. M. Angus, Manager, Nuclear Engineering Services, made a formal commitment to the NRC to install drains in all relief valve tailpipes for which ASME code commitments were applicable. The inspectors considered that this resolved the concern in a satisfactory manner.

8.2 (Open) Unresolved Item 50-275/93-34-03: Safety Significance of Lack of Rated Fire Barrier for Power Operator Relief Valve Conduits

During a postulated fire, equipment may be subject to spurious operation as a result of short circuits. Therefore, 10 CFR 50, Appendix R, Section III.G requires that equipment which may have spurious operation during a fire and, therefore, may prevent safe shutdown of a plant must be protected from short circuits by rated fire barriers. For Diablo Canyon, spurious operation of a PORV during a fire may prevent safe shutdown.

The licensee identified that, for several fire areas, the circuits for the Power Operated Relief Valves (PORVs) were not protected from spurious operation by rated fire barriers, but by dedicated conduit (conduit in which only one circuit is installed). The inspector was concerned that this configuration may not meet the requirements of 10 CFR 50, Appendix R, Section III.G, which requires rated fire barriers.

The licensee provided engineering evaluations which concluded that the conduit and other fire protection measures in the affected area provided adequate assurance that a fire would not cause a spurious operation of a PORV. Part of the engineering evaluation was based on acceptance by the NRC of similar configurations of dedicated conduit for other safe shutdown equipment, as well as identification of fire suppression and a light fire loading in the areas.

Conclusion

It was not apparent that adequate basis had been provided to allow an exemption from the requirements of 10 CFR 50, Appendix R. Accordingly, this issue will be further discussed with Regional and NRR management to achieve resolution.

9 IN-OFFICE REVIEW OF LICENSEE EVENT REPORTS (90712)

The following LERs were closed based on in-office review:

- 275/94-01, Revision 0 Inadequate Fire Barrier Penetration Seals Due to Lack of Damming Boards
- 275/94-03, Revision 0 Technical Specification 3.7.1.1 Not Met During Main Steam Safety Valve Surveillance Testing Due to Indeterminate Causes

ATTACHMENT 1

1 PERSONS CONTACTED

1.1 Licensee Personnel

*G. M. Rueger, Senior Vice President and General Manager, Nuclear Power Generation Business Unit
*J. D. Townsend, Vice President and Plant Manager, Diablo Canyon Operations
W. H. Fujimoto, Vice President, Nuclear Technical Services
*R. P. Powers, Manager, Nuclear Quality Services
*J. S. Bard, Director, Mechanical Maintenance
M. J. Angus, Manager, Technical Services
*W. G. Crockett, Manager, Technical and Support Services
*R. P. Flohaug, Supervisor, Performance and Assessment
*S. R. Fridley, Director, Operations
*R. D. Glynn, Senior Performance Assessment Engineer
*T. L. Grebel, Supervisor, Regulatory Compliance
*B. W. Giffin, Manager, Maintenance Services
J. J. Griffin, Group Leader, Onsite Engineering
C. R. Groff, Director, Plant Engineering
*J. A. Hays, Director, Onsite Quality Control
*J. R. Hinds, Director, Nuclear Safety Engineering
*K. A. Hubbard, Engineer, Regulatory Compliance
*J. E. Molden, Director, Instrumentation and Controls
*T. A. Moulia, Assistant to Vice President, Plant Management
*S. R. Ortore, Director, Electrical Maintenance
P. G. Sarafian, Senior Engineer, Nuclear Quality Services
*J. A. Shoulders, Director, Nuclear Engineering Services
*D. A. Taggart, Director, Onsite Quality Assurance

*Denotes those attending the exit interview.

1.2 NRC Personnel

*M. Miller, Senior Resident Inspector
*M. Tschiltz, Resident Inspector
J. Winton, Inspector Intern, NRR

In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

*Denotes personnel that attended the exit meeting.

2 EXIT MEETING

An exit meeting was conducted on March 29, 1994. During this meeting, the inspectors reviewed the scope and findings of the report. The licensee acknowledged the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

ATTACHMENT 2

ACRONYMS

ASW	Auxiliary Salt Water
LER	Licensee Event Report
M&TE	Measuring and Testing Equipment
MSSV	Main Steam Safety Valve
NCR	Nonconformance Report
NSOC	Nuclear Safety Oversight Committee
PORV	Power Operated Relief Valves
PSRC	Plant Staff Review Committee
PRT	Pressurizer Relief Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWP	Radiation Work Permit
RHR	Residual Heat Removal System
RVRLIS	Reactor Vessel Refueling Level Indication System
SFP	Spent Fuel Pool
SPR	Site Problem Report
SSPS	Solid State Protection System
UFSAR	Updated Final Safety Analysis Report